WM'96
Conference Proceedings
"HLW, LLW, Mixed Wastes and Environmental Restoration --Working Towards A Cleaner Environment"

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PREFACE

The participation this conference was significantly reduced by concerns with the US budget process. We are convinced that this is a one time event and we will be back on track next year.
This year we made a major change in the processing of the papers. We required the authors to send us a full paper for review and possible revisions. This change occurred as a result of recommendations of a special quality improvement subcommittee of the Program Advisory Committee. The process worked better than we expected and we will retain the process for WM'97. Please advise us of your opinions and whether we are gaining improvements in the writing of the papers.
We are moving rapidly to utilize the Internet and will shortly have our own web server. We are applying for the name WMSYM.org and will make an announcement on http://basix.com/~wmsym/ when this occurs. We have the Call for Papers on net for free access and will have next years' proceedings on with authorization to each attendee qualifying for a CD ROM version. We will put the preliminary program for WM'97 on and you will be able to register on the Internet. Note that payment by VISA, Master Card or Discover Card will still require a phone call, fax or letter.

Session 01 -- PLENARY
Co-chairs: John D. Hurley, WSRC
Robert G. Holmes, British Nuclear Fuels
1-2
HIGH NOON FOR NUCLEAR POWER
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ABSTRACT
Spent nuclear fuel management is mired in delay, missed milestones and litigation. Unless the commercial nuclear industry and the government can break the log jam, Mr. Gallagher, believes, nuclear power plants will become too expensive to operate. He draws parallels between the spent fuel debate and "High Noon," the classic 1952 Western. In the movie, Gary Cooper plays a marshall struggling to persuade the citizens of a small Texas town to unite and save the community from a gang of outlaws. To save nuclear energy, the commercial and government sides must similarly unite to resolve spent nuclear fuel problems. Mr. Gallagher believes utilities and the DOE must take three steps to ensure nuclear power's continued viability. First, they must reestablish open, two-way dialogue. Second, they must find common ground on spent fuel issues. Finally, they

must reach consensus on a standard spent fuel canister system to foster a cost-effective spent fuel management solution. He discusses the Multi-Purpose Canister (MPC) and how it can resolve spent fuel issues. One way to break the current deadlock is to go forward with the transportation portion of the MPC system. This action, Mr. Gallagher believes, would move the industry toward a standard canister design without impinging on competition. In a time of budgetary restraint, a limited MPC program could be completed with very little funding, Mr. Gallagher asserts. Equally important, it would allow the industry to begin resolution of potentially sensitive spent fuel transportation issues.

REMARKS

Good morning.

I'm here to discuss spent nuclear fuel management.

I know. It's a touchy subject. . . the cause of debate, discord and division.

I'm not here to continue that debate. I'm here to end it.

Because whenever I hear endless arguments over spent nuclear fuel, I hear the clock ticking. . .

. . .ticking toward nuclear energy's high noon. Make no mistake. Nuclear power's high noon is near. And unless we can find ways to work together, America's nuclear power plants will shut down.

We can't let that happen. Time is running out. It's almost High Noon.

High Noon. . . that classic 1952 western starring Gary Cooper. He plays Will Kane, the courageous Marshall of Hadleyville, a small, dusty town somewhere in Texas. Kane must save Hadleyville from a group of desperadoes. Their leader, Frank Miller, is arriving on the noon train. And when he does, they aim to kill Kane and loot the town.

Fig. 1

Kane must convince the people of Hadleyville to band together. . .face the gang. . .and save the town. He constantly checks the time, ever mindful of the impending disaster at high noon.

The outlaws are armed and ready, impatiently waiting at the station. The High Noon train will bring their leader, and they'll bring destruction to Hadleyville.

While the outlaws prepare for their assault, the good people of Hadleyville argue. They argue over who's responsible for the trouble. And they argue over who should end it.

Sound familiar? Let's drop in on their debate. . .

Citizen #1: "Yes, we all know who Miller is, but we put him away once. And who saved him from hanging? The politicians up north! . . . "

Do you see the parallels between the citizens' struggle and our own? They want to blame the government. They think the government caused the problem, and the government ought to solve it. That's pretty close to the argument of some nuclear utilities in the spent fuel debate.

Here's another point of view. . .

Citizen #2: "Well, I say this. We been payin' good money right along for a Marshall and deputies. Now the first time there's any trouble, we're supposed to take care of it ourselves. Well, what we been payin' for all this time? I say we're not peace officers. This ain't our job. . . ."

That sounds mighty similar to payin' good money into the Nuclear Waste Trust Fund. Utilities, or more correctly, utility ratepayers, have been paying into the fund since it was established by the Nuclear Waste Policy Act of 1982. The fund pays for the government to take spent fuel off the utilities' property beginning in 1998. It also pays for the government to

permanently dispose of the fuel in a federal geologic repository - presumably at Yucca Mountain.

Yet DOE has fallen victim to declining budgets. The opening of the repository has been delayed, initially from 1998 to 2003. . . and, then again. . .to 2010. As the delays appeared, DOE's commitment to accept spent fuel in 1998 began to slip. Its once firm commitment has become a firm "maybe."

The utilities feel betrayed. And DOE is sympathetic. But current law prevents DOE from moving ahead on a specific interim storage site. And the law does not allow DOE to accept spent fuel at any site other than the repository. . .

Citizen #3: "I can't believe I've heard some of the things that have been said here. Y'all oughta be ashamed of yourselves! Sure, we paid this man, and he was the best Marshall this town ever had. It ain't his trouble, it's ours. I tell you, if we don't do what's right, we're gonna have plenty more trouble. . ."

DOE argues there will be plenty more trouble if it moves too quickly on interim storage. That could foster public opposition. And that opposition could sink Yucca Mountain.

Citizen #4: "I've been saying right along we need more deputies. . ."

And I think everyone, except the politicians up North in Congress, has said that we need to devote more resources to the spent nuclear fuel program. After all, the money is there. Twelve billion dollars has been paid into the Waste Trust Fund. But because of Congressional constraints on funding, the DOE doesn't have the authority or the money to fund interim storage or to continue the Multi-Purpose Canister.

And it's concerned that every dollar invested in these programs will divert scarce resources away from Yucca Mountain site characterization, the ultimate goal of the program. . . .

I understand these arguments. They make sense. They're all valid viewpoints in the debate. Unfortunately, the citizens of Hadleyville never understood that words are a poor substitute for action.

While the citizens argued, the outlaws loaded their weapons and got ready to destroy Hadleyville. Their debate and delay gave the outlaws the opening they needed.

We have to understand that our infighting and indecision give ammunition to the anti-nuclear gunslingers. Our arguments give the obstructionists weapons that they will surely use against us. Our division emboldens them. Our court cases provide a public forum to promote their agenda.

The spent fuel issue can shut down nuclear power.

As we continue to argue, time is running out for nuclear power as a viable energy option.

We can't let this happen.

To save the nuclear energy option, we must resolve our differences and move forward, together, to solve the spent fuel problem. Once and for all. And we must do it now.

Let me suggest three steps we can take. Today.

First, we must reestablish open communications between the utilities and the DOE on spent fuel management.

Until last year, communications were good. But that was before Yucca Mountain was delayed and MPC funding was cut. Somehow, the lines of communication seem to have been cut along with the budgets.

This is the heart of our spent fuel management difficulties. It's the reason we're fighting among ourselves instead of solving spent fuel problems.

We need to revitalize communication. Right now. Today. Tom Grumbly was absolutely right when he told The Washington Post. . .and I quote. . . "The only way we're going to get the public's trust is to put our cards on the table. Face up. All the time. Even if it's painful."

We have to maintain full, open and honest communications among ourselves, and with all our stakeholders.

Next, we should recognize the many areas where we already agree. This will reinforce our common purpose. And it will foster mutual understanding, so we can work productively to resolve our points of disagreement.

I think we can all agree that we should take care of our spent fuel problems now. . . instead of leaving them for our grandchildren to inherit.

Another area of agreement. We all recognize the need for a central, permanent spent fuel repository. And until we get one, we need a rational way of storing spent fuel in the interim.

And there's a final point we can all agree on. We need a spent fuel storage system that's safe and cost-effective. Because no matter whose budget the money goes through, it all comes from the same place. Our pockets. As ratepayers and taxpayers, we pay all the bills.

Finally, we must reach consensus on a standard spent fuel canister system. This should be an industry standard. DOE should own the design, so that we get the benefits of competition and private sector manufacturing.

This is an urgent issue.

Seventeen plants already have spent fuel storage systems in place or on order. Twelve more will need storage by the year 2000, and 70 plants will need it by 2010.

In the absence of a standard system, a variety of different canisters are being used. There are single-purpose, storage-only systems. There are dual purpose systems - for storage and transportation. There are metallic casks and concrete casks. There are large ones, small ones, short ones, tall ones. There's everything except your choice of colors. But wait for next year's models.

Variety is great for the family car, but not for safe, cost-effective spent fuel storage.

The DOE is worried about this proliferation of canisters. They have to figure out how to get the fuel out of all those different canisters and into one permanent repository. At a recent NRC hearing, Dan Dreyfus told the commission that a situation could arise "where dry storage (built by utilities) could be so site-specific that it could create a management problem."

The NRC is also worried. At that same hearing, NRC Chairman Shirley Jackson and Commissioner Rodgers expressed concerns about whether different transportation and storage canisters developed by private industry would comply with repository requirements.

And we should be worried.

Because we, as ratepayers and taxpayers are going to have to foot the bill to close the cycle on many of these half-way solutions.

That's why we need a standard solution. We need a solution that accommodates fuel from all of our nuclear plants. We need a solution that

integrates all the elements of the spent fuel program: Storage. Transportation. And permanent repository emplacement.
The solution exists.
It's the Multi-Purpose Canister.
I say this, of course, as head of the Westinghouse unit given responsibility for developing the MPC by DOE's M&O contractor for the high-level waste program.
But I also say it as a 30-year veteran of the nuclear field, and as a person who cares deeply about keeping the nuclear option open.
A significant benefit of the Multi-Purpose Canister is its ability to accommodate 90 percent of the utility spent fuel inventory. Today's storage and transportation canisters can accommodate only a fraction of that fuel.
Fig. 2
In addition, today's canisters won't be accepted at the repository. Once fuel is loaded into the MPC, it stays in the MPC - through storage, transport, and into the repository.
Radiation exposure is minimized, and so is low-level waste. Other canisters must eventually be scrapped as their fuel is transferred to the repository. That will create huge amounts of low-level waste. And huge costs. Because we'll have to pay to re-handle the fuel; we'll have to pay to dispose of the scrap canisters, and we'll have to pay for new repository canisters.
Fig. 3
So without the MPC, we'll pay more for spent fuel management. A lot more. That's what the obstructionists want. They'd like us to price nuclear power out of business.
We can't let that happen.
And we don't have to. This spring, the MPC design will be complete. Let's take a look at it.
Fig. 4
The Multi-Purpose Canister system includes:

- The canister with its basket assembly
- A transfer cask to move the fuel to on-site storage
- A concrete storage unit, and
- A transportation cask and rail car to transport the MPC off site.

Here's how its used. In the fuel pool, each spent nuclear fuel assembly fits into a guide tube constructed of 304 stainless steel. Spacer plates secure the guide tubes. The spacer plates are assembled with support rods and sleeves to form the basket, which ensures that the fuel remains subcritical. The MPC basket is constructed of XM-19, the same corrosion resistant stainless steel used in sulfuric acid processing plants.
The MPC shell is 316L stainless steel. A shield plug uses depleted uranium in a stainless steel shell to attenuate gamma radiation. A 1-1/4 inch thick 316L stainless steel inner closure plate forms the primary containment boundary, and a 1-3/4 inch 316L stainless steel outer closure plate provides a redundant seal.
During loading, the MPC canister is enclosed in a reusable transfer cask that provides neutron and gamma shielding, heat dissipation, and structural protection. The MPC can then be transferred to a concrete storage unit for on-site vertical storage, or it can be transferred to a reusable transportation cask for shipment off site.
The transportation cask consists of four layers: The inner shell is XM-19, surrounded by a layer of depleted uranium which provides radiation

shielding. This feature also provides a margin of strength and toughness in physical testing, which would not be present in an equivalent lead-steel or all-steel design. The third layer is an XM-19 structural shell. The fourth layer consists of neutron shielding and copper fins. Proven, cast-in-place cementious material is used to attenuate neutron radiation through the cask wall. Boron, added to this neutron shielding material, reduces secondary gamma radiation from neutron absorption. Effective shielding is balanced with heat transfer requirements through the use of copper ribs, which dramatically improve the flow of heat to the outside of the cask. A thin copper-lined stainless steel jacket encloses the cask.

For transportation off-site, the trunnions are removed and the cask is fitted with large polyurethane foam and honeycomb aluminum impact limiters and the personnel barrier. In this integrated system, the storage, transfer and transportation casks are reusable.

We need the MPC. We need to get this standard spent fuel solution out of the computer and into the nuclear power plants where it's needed.

Let me suggest a path forward.

The most expedient, cost-effective way to do this is for DOE to allow the transportation portion of the MPC program to go forward.

Let me explain the benefits.

First, it will move the industry toward a standard solution without infringing on competition.

Second, it will maintain the momentum of the MPC program, making a standard spent fuel storage system available as soon as possible for the utilities that need it.

Third, it will only cost about $10 million to license the MPC transportation components. That's about one percent of the projected Yucca Mountain expenditures through 2010. I think that's a small price to pay to help move the industry confidently ahead on spent fuel storage. In this time of budgetary restraint, it's the right thing to do.

Finally, a government-certified standard transportation cask will take an important weapon away from the obstructionists. Transportation has the potential to be one of the most contentious spent fuel management issues. People worry about nuclear waste moving through their communities, even though our transportation safety record is perfect.

That's why DOE and Westinghouse tackled transportation issues early for the Waste Isolation Pilot Plant, near Carlsbad, New Mexico. Well before the repository was ready to open, the TRUPACT II transportation casks were designed, tested and certified. The transportation routes were established, and emergency response personnel were trained. So transportation is not an issue as the repository clears its final regulatory hurdles on the way to a 1998 opening.

By moving forward with the transportation portion of the MPC, we can resolve one of the most contentious issues in spent fuel management.

The obstructionists, like the outlaws in High Noon, would like to use transportation and other spent fuel issues to hold nuclear power hostage. And we are giving them the opportunity. Because we are allowing debate and division to keep us from taking the responsible actions needed to solve these issues.

I believe we can learn a lot from this Hadleyville citizen...

Citizen #4: ". . .so if Miller comes back here today, it's our problem. It's our problem, because it's our town. We made it with our own hands out of nothing. And if we want to keep it decent, keep in growing, we've

got to think mighty clear here today. And we've got to have the courage to do what we think is right, no matter how hard it is. Alright."
We have to do what's right, no matter how hard it is. And we have to do it now. Time is running out. High Noon is nearly upon us.
In the movie, the townspeople never did stop arguing long enough to band together and back Marshall Kane. He had to face the gunmen alone. And yet he still saved the town. . . and got the girl.
But that's Hollywood, my friends. This . . . is the real world.
America's great accomplishments don't come through the courts. They come through cooperation, communication, and responsible action. They come by working together to solve tough problems.
Nuclear power can keep our country strong and secure. Our story can have a happy ending, too, but only if we keep the spent fuel issue from taking us hostage.
We must put our differences behind us. We must all become deputies in the struggle to save nuclear power. Today.
Right now, we can end our unproductive arguing, eliminate our excuses and move forward on a standardized spent fuel storage system.
Let's all become spent fuel action deputies. Let's saddle up together to put our spent fuel problems behind us. And let's move now. While there's still time.
Let's ride!

1-3
THE GLOBAL IMPLICATIONS OF OPENING THE WASTE ISOLATION PILOT PLANT

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ABSTRACT
A missing global link in the nuclear fuel cycle and safe radioactive waste management is the opening of the world's first permanent facility for the safe disposal of long-lived radioactive waste such as transuranic radioactive waste (TRUW), and spent nuclear fuel and other high-level radioactive wastes (HLW). Pursuant to current laws in the United States, both TRUW and HLW will be disposed of in deep geologic repositories, but at different locations. This paper provides our estimate of the situation and the global impact of opening a TRUW repository at the Waste Isolation Pilot Plant (WIPP) site in the state of New Mexico.
The WIPP project began in 1974 and it has a long-standing record of excellent science and safety. In October 1993, the United States Department of Energy established the Carlsbad Area Office (CAO) with a mission to review and integrate the safe management of all TRUW in the USA. By April 1994, the CAO had conducted the review and implemented a new mission which included an accelerated schedule for the opening of the WIPP in 1998 rather than in 2001. Subsequently, the CAO has set precedents at the WIPP with early regulator and stakeholder involvements in the regulatory process, and with the development of a detailed, well-structured, and defensible decision-making basis for the most cost-effective path to the timely opening of the WIPP in 1998.
At the end of August 1995, the WIPP is well into the certification/permitting process and on schedule to open in 1998. The timely opening of the WIPP repository would reduce risks and increase the protection of human health and the environment by removing existing TRUW

from surface-based and near-surface-based temporary storage facilities to a repository located at a depth of approximately 650 meters below the land surface. Moreover, the opening of the WIPP repository would be a global first-of-a-kind operational facility, and its continued safe operation in compliance with one of the strictest environmental radiation protection standards in the world should enhance public confidence in the safety of deep geological disposal of TRUW and HLW both in the USA and abroad.

INTRODUCTION

A missing global link in the nuclear fuel cycle and safe radioactive waste management is the opening of the world's first permanent facility for the safe disposal of long-lived radioactive waste such as transuranic radioactive waste (TRUW), and spent nuclear fuel and other high-level radioactive wastes (HLW). Pursuant to current laws in theUnited States, both TRUW and HLW will be disposed of in deep geologic repositories, but at different locations. This presentation provides our estimate of the situation and the global impact of opening a TRUW repository at the Waste Isolation Pilot Plant (WIPP) site in the state of New Mexico.

THE SITUATION

The WIPP site was selected in 1974 as a potential site for the development of a facility for disposal of defense-related TRUW generated since the 1970 decision by the Atomic Energy Commission (a predecessor agency to the DOE) to begin above ground interim storage of transuranic waste rather than shallow burial. In 1992, the WIPP site was withdrawn from public use by the US Congress for completion of the development of the nation's TRUW repository.

The DOE established the Carlsbad Area Office (CAO) in December 1993 to:

1. Integrate the characterization and preparation/packaging of existing and future TRUW at ten main TRUW generator/storage sites through the National TRU Program (NTP); and to
2. Open and operate the nation's first deep geologic disposal system (repository) (Fig. 1) for TRUW at the WIPP site.

Fig. 1

THE PROBLEM

It is estimated that 144,000 cubic meters of TRUW exists in the U.S. Approximately 63,000 cubic meters is already packaged. It is contained in a variety of metal drums and wooden and metal boxes. Since 1970, the waste has been placed in retrievable storage. They are stored in earth-covered mounds, concrete culverts, and other types of facilities. Over 70% of the drums are over 10 years old and 20-30% of the drums stored in mounds contain corrosion pinholes or are beginning to deteriorate. This waste is dispersed across the country with five major sites containing 96% of the waste. Remediation and decommissioning of facilities are expected to package another 81,000 cubic meters of TRUW. Over 30 million people live within a 50 mile radius of TRUW. Taxpayers are supporting a budget that spends approximately $400 million a year to maintain this waste in temporary storage.

THE SOLUTION

As early as 1957, a National Academy of Sciences (NAS) report to the Atomic Energy Commission recommended the burial of transuranic waste in geological formations. A 1992 report from the NAS notes that most countries have concluded that "the best means of long-term disposal...is deep geological emplacement...."

The WIPP site was identified in 1974 as a potentially suitable site for a TRUW repository, and an extensive site characterization program was initiated. The 6.4 by 6.4 Kilometer (Km) WIPP Site is situated 42 Km Southeast of Carlsbad, New Mexico in an arid and sparsely populated desert area. Based on the results from the site characterization program (including laboratory testing, model developments, and analyses), the construction of an underground test facility at an approximate depth of 650 meters below the surface in the center of the candidate host rock, the Salado Formation, commenced in 1982. The Salado Formation is about 250-million-years-old, regionally extensive, 600-meter-thick, stable, sedimentary evaporitic sequence of rocks dominated by rock salt (mainly halite). The construction and testing of the underground facility as well as all facilities and equipment required to commence the receipt, handling, transporting, and emplacement of TRUW were essentially completed in 1988.

THE APPROACH

Since the inception of the CAO, a preliminary waste inventory has been compiled, the experimental program has been streamlined, three permit/certification applications have been submitted to the regulators, the schedule for opening the WIPP repository has been advanced almost three years to April 1998, and the estimated cost to open the WIPP repository has been reduced by more than 300,000,000 dollars.

The NTP has periodically inventoried and updated available information on TRUW at some 20 generator/storage sites and projected future TRUW generation. The most current estimated stockpile of TRUW is about 59,000 m3 of Contact Handled-TRUW and 4,000 m3 of Remote Handled-TRUW.

During 1994 and 1995, the CAO evaluated the experimental programs in terms of contributions to regulatory compliance by means of the System Prioritization Method (SPM) to establish the most promising combination(s) of activities to meet the final disposal standards for TRUW, i.e., Code of Federal Regulations, Title 40, Part 191 (40 CFR 191). Based on the SPM results, the Manager of CAO decided in April 1995 to reduce the experimental programs from 116 activities to eight activity sets. The SPM initiative also pioneered early and iterative regulator and stakeholder involvement in the process of regulatory compliance.

After a thorough evaluation of the WIPP experimental programs, the TRUW inventory (both existing and projected), and the engineered barriers, the Carlsbad Area Office announced a new TRUW management strategy on April 5, 1994. This strategy accelerates the opening of the WIPP repository by three years, i.e., from 2001 to 1998, and the key milestones are shown on Fig. 2, the WIPP Disposal Decision Plan. The main objectives of this strategy are to:

- resolve regulatory compliance and technical issues;
- characterize the waste;
- address transportation and safety issues; and
- involve stakeholders in the regulatory compliance process.

Fig. 2

In 1995, the WIPP entered the permitting phase. The following three applications to receive and dispose of TRUW and mixed-TRUW at the WIPP site were submitted to the cognizant regulator:

1. The Draft Compliance Certification Application (DCCA), describing how TRUW will be safely received and disposed at the WIPP site, was submitted to the Environmental Protection Agency (EPA) on March 31, 1995;

2. The draft No-Migration Variance Petition (NMVP), demonstrating that hazardous constituents will not migrate from the WIPP repository at concentrations harmful to human health and the environment for 10,000 years, was submitted to the EPA on May 31, 1995; and

3. The Resource Conservation and Recovery Act (RCRA) Part B Permit, describing the site, its facilities, and how the hazardous waste received at the WIPP site will be safely handled on a daily basis, was submitted to the New Mexico Environment Department on May 31,1995, one month ahead of the Disposal Decision Plan schedule.

Regulator comments on these applications were received by the CAO during 1995. Additional regulator, oversight groups, and stakeholder comments are expected in 1996. These comments will be considered and responded to prior to the submittal of the Final Compliance Certification Application to the EPA in October 1996.

TRANSPORTATION/SAFETY ISSUES

Since the beginning of this country's nuclear program, there have been more than 2,500 shipments of spent fuel and many more shipments of low-level waste. The safety record to date is very good. Transportation of nuclear waste is of particular concern to states and Native American tribes along the main transportation routes to WIPP. DOE, state and tribal governments, and several national and regional transportation organizations are actively preparing for potential shipments of transuranic waste to the facility. These preparations include the development of policy and procedures for preventing accidents, responding to emergencies and bad road and weather conditions, conducting inspections, and providing equipment.

Packaging provides the primary barrier to the release of radioactive contents during shipment. The Transuranic Package Transporter (TRUPACT II) containers have been especially designed to ship waste to the WIPP. This system meets or exceeds every Department of Transportation and Nuclear Regulatory Commission regulation and is the safest method ever devised for transporting any hazardous material in the US.

IMPEDIMENTS

The CAO works with 16 federal agencies, 23 states, and 39 Indian Tribes. The CAO is committed to increased interaction with all stakeholders regarding major decisions. The CAO is involving stakeholders throughout its decision making process in a visible and accessible way, from start to finish. That is, stakeholders are advised early of involvement opportunities, comments will be encouraged, and stakeholders will be provided with responses and program updates. Our aim is to surface and resolve concerns before they become more difficult and costly to resolve. Trying to satisfy all these customers is about as easy as "herding cats" or actually "herding turtles" in the often-bureaucratic arena in which we work. Besides trying to align all these disparate political agendas, we have to overcome all the misconceptions held by stakeholders. For example, some people along the transportation corridor fear that they will suffer harmful radiation effects during the shipments, a fear fed by the often over-stated, emotional attacks by those special interests opposed to nuclear activities and supported by non-factual statements about the hazards or risks from low-level radiation exposure.

In the regulatory arena, which will be covered in greater detail tomorrow at Session 19, we are required to predict the performance of the repository for 10,000 years into the future. That is two times the age of the pyramids. Given this extraordinarily difficult task, we are applying

the most sophisticated state-of-the-art performance assessment analyses process currently in use; we will try and predict the natural resources needed thousands of years into the future, predict natural forces and changes in the earth, and then defend the soundness of the WIPP repository to the EPA, New Mexico Environmental Department, and other stakeholders. Compared with these requirements, performing on schedule, with a shrinking budget, is easy.

IMPACT

The opening of the WIPP repository will drastically reduce risks to human health and the environment by facilitating the removal of existing TRUW from surface-based and near-surface-based storage facilities and contaminated sites, to a TRUW repository located in a virtually uninhabited desert area at a depth of approximately 650 meters below the surface in the center of a stable and virtually impermeable rock salt formation. Several of the TRUW storage sites are located in the vicinity of sizable population centers. If there is any risks to the public and/or the environment from the alpha radiation emitted by TRUW, it is orders of magnitude greater where the waste is than where we are trying to put it - 650 meters deep in a 250 million year old salt formation.

Moreover, the WIPP will be a world-class first-of-a-kind facility for safe disposal of long-lived TRUW. It will be the first permanent repository licensed under a rigorous, ultra-conservative regulatory program requiring a risk-based performance assessment to demonstrate compliance with the environmental standards for 10,000 years. Its continued safe operation in compliance with several hazardous waste regulations and one of the strictest environmental radiation protection standards in the world will enhance public confidence in the safety of deep geological repositories both in the USA and abroad.

As part of our compliance program efforts, the CAO has requested a review of the WIPP performance assessment effort by the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) and the International Atomic Energy Agency (IAEA). These two international agencies have had long and effective involvement in the nuclear arena throughout the world. The eminent engineers and scientist on the OECD/NEA and IAEA team who are renowned; world wide budget matter experts, will prepare an independent review of the WIPP's program to demonstrate compliance that will serve as a model for other assessment and licensing efforts. Our approach to engaging the regulator early-on in technical discussions in the licensing process and to including the public in the early stages of the effort will serve as a model that can be replicated in other nations licensing efforts. As we have in the past, we are prepared to share our lessons-learned with the international community through our involvement with the IAEA and/or OECD/NEA or any individual nation-to-nation basis. Many of our foreign colleagues, especially those who attend this meeting, have already toured the WIPP and endured extensive technical and regulatory briefings about our approaches to and the status of the scientific, operational and regulatory aspects of the CAO program for operating the WIPP.

CONCLUSION

Whether or not you live near a nuclear TRUW site, the success of the WIPP will have significant effects on public health and on the economy and environment in which you and future generations will live. Any taxpayer, electricity consumer, environmentalist, public health advocate, or

individual who is interested in solving this national dilemma of waste management, should become a champion for the WIPP.
For those who haven't visited the WIPP yet, I want to end this presentation by issuing you a personal invitation for a VIP-tour of and detailed briefing about WIPP. It will meet your highest, world-class expectations for a nuclear waste repository.

Session 02 -- HEALTH & SAFETY: NEW INITIATIVES
Co-chairs: Carol Peabody, USDOE
Connie Callan, UNM
2-1
EXPERIMENTAL BOILING WATER REACTOR D&D
ES&H LESSONS LEARNED
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ABSTRACT
The decontamination and decommissioning of the Experimental Boiling Water Reactor (EBWR) was well planned and professionally accomplished by an experienced contractor. In the process, two particular radiological incidents occurred that were outside of the planning envelope. The first involved trace Pu-241 that had decayed to Am-241 and that was vaporized during size reduction. The second involved microspheres of activated steel that were produced in condensation of plasma-arc cutting vapors. While neither resulted in significant exposure, the americium uptake was clearly an "unreviewed safety question," and both deserve attention as lessons learned.
BACKGROUND
The Experimental Boiling Water Reactor (EBWR) was built in 1957 as an experimental facility to demonstrate the practicality (stability, control, etc.) of a boiling water reactor. As with most experimental facilities built in that period, a wide variety of experimental configurations and fuels were used. It is reported that there were no fuel failures during the operation of EBWR. The final configuration included plutonium fueling for the central nine assemblies.
In 1967, the reactor was shut down, defueled, and placed in a "stable interim condition." Little was done at the facility until 1993. In the 1980 time period, the containment structure was used for interim storage of contaminated glove boxes. In 1979, NUS did a survey and preliminary planning for decontamination and decommissioning of the facility. Significantly, NUS surveys concluded that there would be no alpha (fuel) hazard in the D&D of this facility. Further preliminary surveys and planning were done in 1989-1990, confirming (or more literally, repeating) the assessment that there was no alpha hazard. The only identified radiological hazard was from activation products, dominantly cobalt-60. When waste management funding was obtained for D&D of the facility, there was some question as to DOE's requirement for safety analyses for D&D projects and planning proceeded based on best judgement as to the required analyses. A contract was placed with ALARON, who has worked as subcontractor on many nuclear D&D projects but never before as the lead contractor. ALARON was advised that there was no identified or anticipated alpha concern, but the contract terms called for ALARON to do a characterization survey. That survey also confirmed that there was no alpha contamination. Health Physics (HP) coverage was assigned to ALARON

by the contract with oversight by ANL Health Physics. Entrance and exit bioassays for all subcontract radiation workers were also required, with ANL providing the bioassay support (ANL employees are covered by a periodic assay program based on their exposure potential). Respiratory protection was required (and used) by all workers until it was confirmed that there was no significant airborne cobalt-60 contamination.

DOE later ruled that all D&D projects were to be considered "nuclear" projects and subject to the full DOE Order 5480 set of requirements. Based on this, a contract was placed with NES, an experienced group in the preparation of SAR's, to prepare an SAR in accordance with DOE Order 5480.23. The EBWR SAR states that there is no alpha or fission product hazard, and analyzed, as potential accidents, a crane failure dropping heavy contaminated objects and a tornado causing release for contamination. Since the source term is limited, neither of these accidents is found to be a significant public hazard.

THE AMERICIUM INCIDENT

The first phase of the project was the removal of support systems. When the workers who did that work completed their assignments and left the site, they submitted the normal urinalysis samples. When these were analyzed (in August 1994), clearly measurable levels of tritium were noted. As there was no expected tritium (or identified reason to expect tritium), a more thorough analysis was performed (samples were taken in late August and urinalysis samples were analyzed by September 2). The second phase of the D&D was already underway with the size reduction of core internals, vessel and vessel-pit lead removal, and chipping of concrete. The workers involved in these operations were also part of this bioassay review. Among this latter group, measurable alpha uptake was noted in seven urinalyses and confirmed by fecal analysis to be americium-241. Estimates are that the most highly exposed individual could have obtained a lifetime Cumulative Estimated Dose Equivalent (CEDE) of no more than 300 mrem (Table I). Bioassay results are now complete, and seven individuals show alpha levels in their urinalysis, twelve detectable in fecal analysis. Each of these also shows detectable, but not radiologically significant, uptake of tritium, cesium-137, and cobalt-60.

Upon confirmation of biological uptake via analysis of fecal samples (September 9), all work on the project was suspended pending a complete analysis of the situation.

Various ANL "old-timers" suggested a variety of potential sources for the americium, but no specific recollection of a concentrated Am-241 source was identified. The potential sources included fueled control rod followers and intentional failed-fuel experiments. Both fueled followers and failed-fuel experiments were documented in EBWR test reports and were performed well before the plutonium fueled operations. These earlier experiments involved the use of U-235 and U/Th fuel, respectively, and were documented as having been removed from the reactor. Because of the nature of the fuel, it was unlikely that these were sources of the Am-241, although the failed-fuel experiment would have resulted in fission product release to the reactor system. There was no evidence that fueled followers had been used in control rods during the Plutonium Recycle Program. A careful review and analysis of the radioactivity patterns measured from the control rods demonstrated that no fueled followers were involved in the D&D cutting operation.

Table I

Since Am-241 is a daughter of Pu-241, the committee concentrated on the Plutonium Recycle Program activities that were conducted as part of the last experimental program before the facility was shutdown and placed in dry lay-up. A committee member reviewed monthly Reactor Development Progress Reports generated during this time period. These reports highlight the various reactor programs in operation at the time. This review identified references to an experiment for the measurement of epithermal capture-to-fission ratios of Pu-239 and Pu-241 during power operation in EBWR. This experiment involved the placement of Pu-239 and Pu-241 foils in the EBWR reactor core. Further reviews of the Reactor Physics Division annual reports found a notation that one Pu-241 foil was "lost in EBWR" (ANL-7310, page 42). This was confirmed in the experiment final report (ANL-7795). The existence of this "lost" foil was not identified in pre-job characterization.

Although the Pu-241/Am-241 foil has not been found, the air sample and bioassay data appear to be consistent with the estimated Am-241 foil activity level. A "reverse" calculation of the source term from the maximum measured internal dose resulted in an Am-241 source of the same order magnitude of activity as the lost foil.

The sample measurements indicate that the Am-241 activity is located in the fuel pool, and that particular work associated with the fuel pool, such as underwater plasma arc cutting and pool water filter replacement, causes dispersion of airborne Am-241. Since Am-241 was measured in air samples over many different days, it appears that the Am-241 is a diffuse source. If an intact foil had been contacted with the plasma arc, it is likely there would have been a very large release on a particular day, which is not indicated in the air sample data. The foils were produced by vapor deposition onto a Zr carrier, which was then encased but not sealed. It is likely then that the americium, as generated from decay of the Pu-241, would be released as a colloid, depositing out as crud on the reactor internals.

Fig. 1.

Plasma-arc cutting of the internals vaporizes the materials in the cutting path. The underwater plasma arc uses an inert carrier gas, which transports vapors to the surface of the pool as airborne contamination. The original foil contained only about 200 micrograms of plutonium, and only small fractions of this could have been released, but the quantities are not inconsistent with the observed uptakes. Excluding vaporization and a reasonable direct inhalation pathway, the quantities are not significant.

The one parameter that does not fit this hypothesis is the plutonium isotopic mix. If the Pu-241 foil were the sole source, there would be essentially no Pu-239 or Pu-240, whereas the Pu-239 and Pu-241 are about equally present.

A second candidate source is tramp uranium. In the early days of nuclear power, some tramp uranium on the exterior of the fuel pins was considered normal. The system contamination from fuel failures was expected to be large, and small amounts of tramp uranium would be readily accommodated. It is unlikely that the core contained sufficient uranium to account for all of the americium, but it is considered likely that this (or failed-fuel) is responsible for the Cs-137 encountered, and probably accounts for a significant part of the plutonium found in the system. The irradiation products from several tens of grams of uranium, combined with the foil described above, provides a credible source for the observed

radionuclides. This plutonium would be almost totally Pu-239, suggesting broadly comparable contributions from the lost foil and from tramp uranium. The dose, however, is almost totally from Am-241; that is, from the foil.

The possible effect of tramp uranium was not considered in pre-job evaluations.

The assumption that there were no transuranic source terms for the EBWR D&D project resulted in the implementation of an inappropriate radiation monitoring and air sampling program and inappropriate work practices. These practices included discontinuation of respirator use upon confirmation of no significant gamma (Co-60) airborne activity. Thus, for the majority of plasma arc cutting work respiratory protection was not used, nor was there a local capture exhaust around the fuel pool. At the levels identified, radiation detection was not set up with a low enough minimum detection levels to allow detection of alpha radiation that would be of concern. Further, the alpha contamination free release criteria of 20 dpm/100 sq. cm., even though permitted by the DOE Radiological Control Manual, was not sensitive enough to flag that a problem would be created by unprotected plasma arc cutting.

Smears of the EBWR shell area by ESH-HP following the project shutdown did find alpha contamination but at levels far below free release criteria. This fact is probably attributable to the low levels of alpha contamination that were generated along with the regular general housekeeping that was required to remove beta/gamma contamination as well as that required to clean up dust that infiltrated the shell from outside.

Tritium Sources

There are a variety of methods by which tritium could have been generated during EBWR operation including activation of naturally occurring deuterium, as a product of fission and as a product of the use of boron for reactivity control. The latter reaction occurs for high energy neutrons (> 1 Mev). In commercial reactors, tritium production is approximately two orders of magnitude higher in PWRs than in BWRs because of the use of soluble boron for reactivity control in PWRs, based on published liquid effluent data. In contrast to current BWRs, boric acid was used to reactivity control in EBWR. There were also a variety of other components that contained boron, including control rods and the thermal shield. There is no evident conclusion as to what the specific source is or why the tritium (tritiated water) would be found primarily in the fuel pool. Although the tritium levels are a minor dose concern, the production of tritium during operation would have occurred, and residual amounts should have been expected.

Lessons Learned: Americium Incident

1. Reactor internals, piping, and components must be assumed to be generally contaminated with very small amounts of uranium, plutonium fission products, and tritium, in addition to cobalt-60 and secondary activation products.
2. The source of these contaminants includes:
 a) Tramp uranium and incidental fuel failures that undoubtably occur during operation of all but the most modern reactors
 b) Tritium from boron control rods and soluble boron when used.
3. The levels of contamination may be below the threshold for identification by many traditional survey techniques and still constitute a hazard to D&D workers.

4. Plasma-arc cutting has become a common tool for D&D. Cutting contaminated materials will necessarily produce airborne contamination. This project did not adequately assess plasma-arc cutting as generating a radiological inhalation hazard.
5. A generally knowledgeable and cautious overall approach to worker health and safety, plus some degree of redundancy in levels of protection, is required to provide an adequate margin of safety.

Conclusions - Americium Incident

1. In old light water cooled reactors (LWRs), the reactor internals, piping and components are likely to be generally contaminated with small amounts of americium, plutonium, fission products and tritium, in addition to the cobalt-60 and secondary activation products that had been analyzed in the SAR as potential radiological hazards.
 a) The levels may be below the threshold for identification by many traditional survey techniques.
 b) The contamination may well qualify as "fixed," but high temperature techniques such as plasma arc cutting can serve to release "fixed" contamination.
2. The sources of all contaminants are unlikely to be conclusively identifiable.
 a) The source of the fission products can be reasonably associated with tramp uranium and incidental fuel failures that undoubtably occurred during EBWR operation. These sources are likely to be present in all old reactors.
 b) No specific source of the tritium has been identified. Since this does not appear to be a significant radiological hazard for this project, no further effort was made to identify the origin. The presence of tritium is likely to be a property of any old (or new) reactor.
3. Prior evaluations and assessments must not be accepted without adequate questioning. Evidence of inadequate documentation to support the working conclusions should be taken as a clear basis for suspicion. In the EBWR case, there were repeated assurances that the only radiological hazard was from activation products (dominated by cobalt-60). Several surveys had been used to confirm this premise. The contract was written, and the contractor was led to believe that protection against cobalt-60 was all that was required.
4. Hazards associated with specific D&D techniques (e.g., plasma arc cutting) must be explicitly included in safety analyses and in work planning.
5. Literal compliance with all applicable DOE orders and an exhaustive safety analysis performed in accordance with best current practices are no protection against surprises of this sort. The potential radiological hazards associated with "other contaminants" such as americium, plutonium, tritium, and fission products are not likely to be identified during the even very careful project planning. Therefore, both surveillance planning and surveillance execution must plan for surprises.
6. The bioassay program should be designed to confirm that no significant exposure occurred during the project as evidenced by not only a pre-project and post-project sampling regime but by appropriate periodic sampling. The role of the bioassay program in identifying an otherwise unanticipated hazard should be considered solely for confirmation, not detection.
7. Given that a major concern in D&D must be the protection of workers when they encounter unanticipated sources of radionuclides, and given

that excessive protective gear is highly counterproductive, it is clear that an extensive retrospective sampling and assay program covering all credible contaminants and all potentially exposed individuals is essential. Confirmation of negative findings should be built into planning as scheduled hold points commensurate with a credible risk exposure.

8. A generally knowledgeable and cautious overall approach to worker health and safety, plus some degree of redundancy in levels of protection, can provide the margin of safety.

THE MICROSPHERE INCIDENT

In an unrelated incident, Health Physics surveys discovered contamination in a small pile of floor sweepings. The ANL Health Physics technician on duty was conducting "direct surveys" and discovered contamination with an approximate activity level of 107,000 dpm Beta-Gamma in a small pile of floor sweepings that were within the controlled area of the shell but not an area designated as a contaminated area. At that time, additional direct surveys in the same general area found an ambiguous but high reading of fixed contamination on a small spot under a plate. Direct surveys were continued due to the fact that routinely conducted smear surveys of the area had failed to turn up this contamination.

This contamination has been determined to be small pieces of slag produced by plasma arc cutting operations. This material is in the form of very small hollow spheres ranging in size from a quarter inch in diameter to pin-head size. These spheres are not detectable by smears in that the activity of any one or few spheres is below detection limits. The primary radioisotope in the spheres is Co-60 with some Sb-125, indicating activated stainless steel.

Material transfers during the D&D operation are accomplished by establishing a work area; the floor is covered with plastic sheeting and the area is posted. After packaging is completed, the area is wet mapped and smeared for loose contamination. The plastic is then removed and discarded. It is likely that the small spheres were rolled off the plastic by the mopping. Since they are not detectable by smears, some can remain on the floor undetected. The background radiation in this area complicates finding the spheres by routine direct surveys.

In this incident, the pile of floor sweepings had a large enough collection of material in a low background radiation area to allow detection.

The entire area that had been used as a work area was subsequently vacuumed with a HEPA filtered vacuum and sources of background radiation moved to allow for direct survey of all floor surfaces where possible. Vacuuming and using duct tape have been very effective in picking up these slag spheres.

Conclusions and Lessons Learned

In a high background area, there can be significant contamination that is not detected by smear techniques, and other techniques must be used for sampling, detection, and analysis.

Plasma arc cutting is an accepted size reduction technique for many applications, including work on contaminated and activated materials. Detailed procedures must be put in place and implemented for control of contamination resulting from such cutting. It has not been previously recognized that the plasma vapors can result in condensation microspheres that roll freely (including rolling off plastic matting as it is being removed). The microspheres are not effectively collected on swipes, and

microspheres from activated steels may be low enough in activity to be undetectable by normal radiological surveys when dispersed.
Searches for this type of loose contamination in high background areas can be carried out using vacuuming through HEPA filters with subsequent measurement of the filters in a low background area.
This is another in a series of lessons learned from this particular D&D project. While the radiological consequences were minimal and corrective actions readily identifiable, the implications for other D&D work is significant due to the broad application of plasma arc cutting.

FINAL COMMENTS

In both of these instances, plasma arc cutting has produced contamination pathways that were not identified and that were beyond the reasonable range of empirical identification. Given that D&D should be conducted efficiently, modern tools such as plasma arc cutting must be used. As a consequence, such surprises must be expected. To ensure that the risks involved are minimal, defense in depth is essential; rigid formality and procedures are of limited value in that D&D will always have surprises.

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WHO IS LIABLE FOR WORKER HEALTH & SAFETY? LESSONS LEARNED FROM THE MILWAUKEE TUNNEL EXPLOSION: THE ENGINEER'S PERSPECTIVE

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ABSTRACT

Determining responsibility for worker health and safety on a construction site is an increasingly difficult task. A methane explosion in Milwaukee in November 1988 killed three employees of the construction contractor. The engineer on the project, CH2M HILL, subsequently became embroiled in five separate legal proceedings arising out of its role on the project, providing services to the owner during construction. Despite express contract language which placed sole responsibility for construction site safety on the contractor, CH2M HILL spent hundreds of thousands of dollars in legal defense costs, including dismissal of OSHA citations by the Administrative Law Judge.
The paper describes the background of the project, and the accident, the resulting legal proceedings, and the lessons CH2M HILL learned from its experiences. Of critical importance, the author emphasizes how different the perspectives are of those involved; the owner, the contractor, the engineer, and the enforcement agencies.

INTRODUCTION

On the early morning of November 10, 1988, an explosion ripped through an underground tunnel being constructed as part of the city of Milwaukee's Water Pollution Abatement Program (the Program). Three employees of the tunneling contractor, S.A. Healy, were killed instantly, and work on the tunnel was postponed for the next several months.
Immediately following the explosion, there were a number of investigations to determine the cause of the explosion and to assess blame for the deaths of the three workers. CH2M HILL, the Program Manager on behalf of the Milwaukee Metropolitan Sewerage District (MMSD), became embroiled in five separate legal proceedings, spending hundreds of thousands of dollars to defend its activitiesdespite express contract language which placed sole responsibility for the health and safety of construction workers on S.A.Healy and its personnel.

The purpose of this paper is to explore lessons learned from the explosion, outlining how health and safety responsibilities were allocated on the Program, and discussing CH2M HILL's approach to health and safety issues, as impacted by the lessons learned.

THE PROGRAM

In the early 1970's, the City of Chicago sued the City of Milwaukee alleging that Milwaukee's sewage was polluting Lake Michigan. As a result of that lawsuit, the City of Milwaukee undertook a massive wastewater improvement program. The Program took 18 years, used 20 million hours of construction work, cost $2.29 billion, and included substantial upgrades at two wastewater treatment plants, the development of an in-line storage interceptor system, and deep tunnels to handle combined overflows from sewage and rain water. Utilizing 324 construction contracts, the MMSD and its Program Manager, CH2M HILL, oversaw total construction costs of $1.6 billion dollars. The Program was completed on time and on budget in the fall of 1995.

A major portion of the Program involved underground construction, including approximately $900 million in construction contracts. The result was over 20 miles of tunnels 300 feet below the ground surface, ranging in diameter from 17 feet to 32 feet; 24 drop shafts and approach channels; and over 62 miles of near surface collector tunnels, ranging from 5 feet to 12 feet in diameter, at a depth of 30 feet to 100 feet below the ground surface.

CH2M HILL managed a consortium of five engineering firms, responsible for planning and design of the entire sewage system upgrade, services during construction, cost and schedule control, and claims mitigation. At the peak of the Program, CH2M HILL managed a work force of 650 people from 88 engineering and technical firms. During this same period, the construction work force peaked at 1,200 people under 75 active construction contracts. CH2M HILL's contractual responsibilities were carefully spelled out in its contract with the MMSD. Figure 1 shows that the contract relationship between CH2M HILL and the MMSD, as well as the contractual relationship between the MMSD and its contractors, was a traditional one.

Fig. 1.

Because of the magnitude of construction and the hazards of underground construction in particular, safety responsibilities were carefully spelled out in both CH2M HILL's contract and in each of the contracts between the MMSD and the contractor. The contracts made clear that CH2M HILL's responsibilities included properly training its own staff, review of and adherence to the contractors' safety programs, and reporting safety concerns to the MMSD and the various contractors. By express disclaimer, however, CH2M HILL was precluded from taking direct action against a contractor and had no authority to stop the contractor's work. The contractors' contracts, by contrast, expressly placed responsibility for safety on the contractor. Each contractor was given sole responsibility for the means and methods of construction, including health and safety. In addition, the contractor was advised that periodic visits by the engineer were for the MMSD's benefit, but that such visits would not diminish in any way the contractor's responsibility for health and safety.

THE HEALY CONTRACT

In 1988, S. A. Healy was hired to construct a two-drive, near surface tunnel, two miles long, for a lump sum of $12 million. Consistent with

other construction contracts let by the MMSD, Healy was given sole responsibility for the means and methods of construction, and for the health and safety of its workers and any personnel on the construction site.

Earlier in 1988, another contractor had encountered methane during underground construction approximately one mile from the site where Healy was to begin its work. As a result of CH2M HILL's subsequent investigation, a contract modification was issued to Healy dated April 5, 1988. In anticipation of potential methane in Healy's tunnel, the modification required Healy to ventilate the tunnel 24 hours a day, install particular fans for the main ventilation system, and engage in continuous monitoring for methane. Under applicable government standards for underground construction, Healy's monitoring equipment needed to provide a warning if the methane levels reached 10 percent of the Lower Explosive Limit (LEL), and shut down Healy's tunnel boring machine (TBM) if methane reached 20 percent LEL. The TBM had to meet specified requirements, and Healy had to use explosion proof equipment and supplies in various places in the tunnel. Finally, Healy was instructed to develop an extensive evacuation and recovery plan to protect its employees in the event of encountering methane.

Healy and two other underground contractors were paid for the costs of implementing the April 5, 1988 modification. CH2M HILL's responsibility was to ensure that the contractors implemented the modification and that the contractors were paid for the required changes.

THE ACCIDENT

In early November 1988, Healy encountered methane during construction. All of the monitoring equipment worked exactly as planned, and Healy was able to successfully evacuate and reenter the tunnel.

Shortly after 7:00 a.m. on November 10, the methane sensors sounded again and the tunnel boring machine shut down. According to subsequent investigations, the tunnel was successfully evacuated. For reasons which are still not clear, however, three of Healy's employees reentered the tunnel in less than the one hour required by the contract. The three employees were Healy's top three people with safety responsibility, including Healy's safety director. Shortly after reentry, an explosion occurred killing all three employees. CH2M HILL was not on site at the time of the explosion, nor had CH2M HILL been notified when the methane sensors sounded.

THE LEGAL PROCEEDINGS

The explosion triggered five legal proceedings involving Healy and CH2M HILL. First, the local District Attorney began a criminal investigation under state law. After reviewing the contractual terms, interviewing Healy and CH2M HILL personnel, and investigating the facts of the explosion, the District Attorney's office decided to prosecute Healy, but not CH2M HILL. Healy was found liable for two counts of criminal negligence, each carrying a $10,000 fine. Healy was cited for failure to follow its own evacuation and reentry plan, and its failure to deenergize the tunnel in accordance with the health and safety plan.

Second, the U.S. Attorney's office began its investigation for violations of federal criminal law. Again, after investigation and interviews, the U.S. Attorney's office declined to prosecute CH2M HILL, but successfully prosecuted Healy for its misconduct. Healy was found liable for three counts of violating Federal safety regulations, each carrying a $250,000 fine. Healy was cited for failure to use explosion proof equipment,

failure to shut off the power after methane was detected, and failure to properly train its workers.

Third, the estates of the three deceased Healy employees filed wrongful death actions against Healy and CH2M HILL. Because Healy's contract required it to indemnify and defend CH2M HILL for Healy's own negligence, CH2M HILL successfully tendered the defense of the cases to Healy's general liability insurance carrier. The three cases were settled without trial for over $750,000.

Fourth, the Occupational Safety and Health Administration (OSHA) issued citations to both Healy and CH2M HILL for willful misconduct. Healy was issued 67 citations, each carrying a potential $10,000 fine. At trial, Healy was found liable for 49 violations, resulting in a penalty of $318,500 ($6,500 per violation).

CH2M HILL was issued 47 citations, each carrying a potential $10,000 fine. Trial was held before an Administrative Law Judge (ALJ) in May-June 1992. All citations against CH2M HILL were dismissed in a decision dated August 25, 1993. The ALJ's 23 page decision reviewed CH2M HILL's contract with the MMSD, how CH2M HILL carried out its contractual responsibilities, and the facts of the methane explosion. The ALJ concluded that CH2M HILL had not engaged in "construction work" as defined under the Occupational Safety and Health Act, and that CH2M HILL did not have health and safety responsibilities at the construction site. Therefore, the ALJ concluded that the construction standards under which CH2M HILL had been cited were inapplicable and that the citations should be vacated.

Fifth, Healy sued the MMSD and CH2M HILL to recover the costs of reentering and completing the tunnel. Despite the prior criminal liabilities, Healy alleged that the explosion and its aftermath caused OSHA to reclassify the tunnel and treat it as a greater hazard, forcing Healy to incur additional costs. The case was settled before trial.

The costs to CH2M HILL of these five legal proceedings cannot be accurately estimated. In addition to the attorney's fees and expert witness costs (in the hundreds of thousands of dollars), CH2M HILL spent substantial amounts of internal time defending these claims, faced extensive adverse publicity both in Milwaukee and around the country, and suffered lost opportunity costs.

Legal Issues (From the Engineer's perspective)Two legal issues are fundamental to an understanding of the lessons CH2M HILL has learned from the OSHA case. The key to determining liability on a construction site revolves around the issue of who "controls" the construction site. Resolving this issue depends on applicable legal precedents, the governing contractual language, and the facts of each case.

The first issue is whether the firm is "engaged in construction work" as defined under 29 C.F.R., Part 1926. In general, the legal precedents have established that a firm is not engaged in construction unless: (a) there is physical labor that is actual construction work (e.g., erection, modification, or repair of a structure) or physical labor that is an integral and necessary part of construction work; or (b) an employer substantially supervises the physical work of construction. For a detailed discussion of applicable legal precedents in this area, see two companion cases: Simpson, Gumpertz & Heger, Inc., 15 BNA OSHC 1851, 1992 CCH OSHD 29,828 (No. 89-1300, 1992) and Kulka Construction Management Co., 15 BNA OSHC 1870, 1992 CCH OSHD 28,829 (No. 88-1167, 1992).

The second issue, of particular importance to CH2MHILL's defense of the OSHA citations, is whether a design professional's rendering advice to a contractor constitutes construction work. The seminal case in this area is Skidmore, Owings & Merrill, 5 BNA OSHC 1762, 1977-1978 CCH OSHD 22, 101 (No. 2165, 1977). The key issue is the authority to control the contractor's work, including the authority to stop work, the authority to stop payments, and the authority to reject non-conforming work. That authority may be express authority (authorized specifically under the contract), or apparent authority (when employees of the firm take actions which demonstrate control over the contractor).

Those who work in the construction arena must understand that there is potential exposure to OSHA regulation in four distinct circumstances:

Your firm is in control of a construction site, and your employees are exposed to a hazard.

Your firm is in control and another firm's employees are exposed to a hazard.

Another firm is in control of the site, but your employees are exposed to hazards as a result of their duties.

Other employees or visitors to the site are exposed to a hazard due to your firm's failure to comply with safety standards.

Although contract language is critical (and was pivotal to CH2M HILL's defense of the OSHA citations), contract language is not necessarily determinative. Employees on a site may act contrary to the express language of a contract (e.g., ordering the contractor to stop work), or employees may be exposed to a hazard when they do not follow a contractor's health and safety plan. Even with contractual protection (e.g., an indemnification clause running from the contractor to the firm), there may be protection from civil exposure, but there is no protection from criminal conduct or OSHA regulation.

LESSONS LEARNED

The following are 10 lessons CH2MHILL learned from the various legal proceedings, primarily defense of the OSHA citations. The lessons come from the engineer's perspective on a construction site, but the guidance on health and safety issues is applicable to any firm working in construction.

Lesson 1: The Language In All Contracts Is Critical

Before undertaking construction work , all applicable contracts need to be carefully reviewed. This includes not only your firm's contracts, but other contracts which may directly or indirectly define how your work is to be performed. As noted above, the key is who has control over the construction site. The roles and responsibilities of all who visit or work on the site must be carefully defined, and other project documents should be reviewed to avoid inconsistencies.

For example, in the Milwaukee OSHA case, there was consistency between the contract CH2M HILL had with the MMSD and the contract between S.A. Healy and the MMSD. In both cases, the contract language emphasized that CH2M HILL's periodic visits to the construction site did not supplant the contractor's responsibility for health and safety. On the other hand, various internal CH2M HILL documents used terms like "construction manager" to define CH2M HILL's role during construction, and this language suggested (at least to OSHA) that CH2M HILL's responsibilities included control of the contractor in some aspects.

Lesson 2: Conduct On The Site Must Be Consistent With The Contract Documents

No matter how carefully drafted the contract documents are (Lesson One), each employee must act consistent with the contractual responsibilities. As noted above, an engineer's employee may not have express authority granted in the contract, but if the engineer's employees order the contractor to stop work, the contractor may act in accordance with that directive, giving the engineer apparent authority over the contractor's work.

For example, an engineer's employee may notice that the contractor is not acting in accordance with its own health and safety plan. The natural tendency would be to point out the problem and perhaps even stop the work until the situation is remedied. Therein lies a potential "Catch-22." On the one hand, there is potential OSHA exposure if the contractor is not stopped, and a hazard exposes your employees to potential danger. On the other hand, interfering with a contractor's work without the authority to do so can incur other legal liabilities for acting outside of the contract.

Normally, the engineer's contract addresses this situation, directing the engineer to bring any deficiencies to the owner's attention so that the owner can stop the contractor's work if necessary. Although this appears to be a somewhat cumbersome procedure, it clearly follows the defined roles in the contract among the owner, the contractor, and the engineer.

Lesson 3: Contemporaneous Documentation Is Imperative

Whenever there is a potential safety hazard, the conduct of all parties should be carefully documented at the time of the incidents. The key to success is making the documents objective (i.e., stating only the facts in a clear manner, including who was there, what happened, what was done, etc.) rather than subjective (i.e., attempting to analyze what happened or who was to blame). The best guideline to follow in drafting any document on a construction site is to assume that a document will be made public, either on the front page of the local newspaper or marked as an exhibit during trial.

In the Milwaukee case, CH2M HILL faced both of these circumstances. One of the documents marked by CH2M HILL was reproduced on the front page of the local Milwaukee paper. On its face, the document appeared to show that CH2M HILL had directed the contractor on which safety equipment should be placed in the tunnel. In addition, numerous documents drafted by CH2M HILL both before and after the explosion were introduced as exhibits in the trial. Fortunately, these documents had been carefully drafted and, in many cases, reviewed by attorneys to insure that they did not expose CH2M HILL to legal liability.

A further caution is important here. One should not assume that a document will remain private, even if marked "Confidential." The only documents which are generally safe from exposure are those which are subject to the attorney-client privilege, i.e., those documents which seek or provide legal advice.

Lesson 4: Control Must Be Given To The Party With The Requisite Education, Training , And Experience

Because the legal precedents establish that the person in "control" of the site has health and safety responsibilities, the definition of control is critical in the contract documents (see Lesson 1) and in the conduct of all parties on the construction site (see Lesson 2).

In the traditional construction relationship, the engineer has historically not had the requisite education, training, or experience to make the necessary health and safety determinations on a construction

site. That role has been given exclusively to the construction contractor. On the Milwaukee tunneling program, S.A.Healy was selected as the tunneling contractor because of its prior experience in this arena, and the contract language was consistent with that experience. CH2M HILL, by contrast, had limited experience in health and safety issues on construction sites, except in an observer capacity on behalf of an owner.

Lesson 5: Even If Not In Control, All Those On Site Need Minimum Health And Safety Training

Because of the extended nature of the Milwaukee Water Pollution Abatement Program, CH2M HILL employees with roles during construction were given a general health and safety training course. In the training, it was emphasized that the contractor's contract controlled conduct on construction sites and that CH2M HILL's role was to provide services on behalf of our client, the MMSD. Employees were advised to be aware of the contractor's health and safety plan on each construction site and to follow the contractor's guidance on health and safety issues. In addition, employees were advised that, except in extreme circumstances (e.g., a case of imminent danger with immediate exposure to serious injury or death), CH2M HILL employees were not to interfere with the contractor's work. If a hazard was noticed, the appropriate procedure was to notify the MMSD in writing, with a copy to the involved contractor.

In the circumstances surrounding the methane explosion, OSHA argued that CH2M HILL's prior conduct (in particular, the contract modification in April 1988) gave CH2M HILL control over the site. This argument was tenuous, however, because CH2M HILL was not present on the day of the explosion, nor did CH2M HILL have any involvement in the decision to reenter the tunnel after the methane sensors sounded.

Lesson 6: Everyone Has A Different Perspective Of Roles And Responsibilities On A Site

As noted above, responsibilities of all parties must be clearly spelled out in the contract documents (Lesson 1) and adhered to by all persons on a site (Lesson 2). The perspectives of who had what responsibilities, however, will change if something serious goes wrong. First, the owner relies heavily on the engineer as an educated, experienced "professional," as the "eyes and ears" of the owner on site, someone who "ensures" that the contractor will adhere to the plans and specifications. Following the methane explosion, the MMSD did its own investigation to determine whether CH2M HILL could or should have done anything to prevent the explosion. One of their critical concerns was whether CH2M HILL had properly inspected to ensure that Healy had installed explosion proof equipment and then monitored Healy's use of the equipment. The ALJ's decision, however, clearly recognized that CH2M HILL's responsibilities did not go so far. Rather, CH2M HILL simply had to ensure that the constructed project generally conformed to the plans and specifications, not to oversee all of Healy's activities.

Second, the contractor's perspective is that the engineer should stay clear of the means and methods of construction (including safety). When a serious claim arises, however, it is not uncommon for a contractor's attorney to name the engineer as a defendant, alleging that the engineer had responsibility to prevent the accident from happening. Fortunately for CH2M HILL, Healy's attorneys did not take such a tack. Healy's personnel testified that CH2M HILL had carried out its responsibilities in an appropriate manner, and that the responsibility for health and safety was solely Healy's.

Finally, OSHA perspective is the most troublesome. Despite the clear language in both CH2M HILL's and Healy's contract, OSHA consistently argued throughout the legal proceedings that CH2M HILL had an express role in health and safety issues long before the explosion. OSHA relied on various documents CH2M HILL had produced in generating the April 1988 contract modification to argue that CH2MHILL was in fact "engaged in construction work." OSHA alleged that the activities in obtaining the methane sensing equipment, for example, required CH2M HILL to "substantially supervise" Healy's work. Again, however, the ALJ rejected OSHA's arguments.

Lesson 7: Construction Projects Are Inherently Risky, And The Legal System May Not Mitigate Those Risks

Owners on public works projects (especially government agencies) are often required to select the contractor with the lowest bid who may or may not be the most qualified or have the past experience. In such circumstances, there are inherent risks to all parties engaged in the construction project, including the engineer. In fact, the owner may expect that the engineer take a more visible role in overseeing the contractor's activities, precisely because of concerns about liability. In addition, contractors, stuck with low bids, have incentives to cut costs whenever possible, and unscrupulous contractors may attempt to cut costs in the safety arena in order to maximize profits. Again, the owner's perspective may be that the engineer should oversee these activities to ensure that the contractor does not take such steps.

Finally, the worker's compensation laws are currently designed to give injured employees certain remedies without proof of fault, yet the same laws allow employees to seek additional relief from others (e.g., the owner and the engineer). In the Milwaukee case, this is precisely what allowed the estate of the deceased employees to sue CH2M HILL. After recovering the statutory amount allowed by the worker's compensation laws from Healy, the attorneys then sued CH2M HILL in order to recover additional sums.

Especially in circumstances of severe injury or death, the legal system is not inclined to leave injured employees without a full and "fair" remedy. If a judge or jury determines that the injured employee did not receive "fair" compensation under the worker's compensation law, there is a greater tendency to assess blame to other parties.

Lesson 8: There Are No Clear GuidelinesHire Competent Experts And, If Necessary, Lawyers

Design professionals and others involved with health and safety issues are increasingly being asked to walk a fine line. Contractors are often adamant that the design professional should not interfere with the means and methods of construction, including safety; owners, on the other hand, want to ensure that the construction project goes smoothly, and that the design professional take a more active role in overseeing the contractor's activities. When the responsibility for "control" is blurred in this manner, there are no clear guidelines as to what the courts or OSHA will do. Accordingly, seeking competent legal advice is often imperative.

In the months following the methane explosion, there were public statements by OSHA officials to the effect that design professionals should have an increasing responsibility for construction site safety. From CH2M HILL's perspective, this meant that the OSHA citations became something of a test case for the OSHA administration. OSHA argued

throughout the legal proceedings that general legal precedents involving design professionals and contractors were simply not applicable; instead they argued that prior OSHA cases demonstrated that CH2M HILL's responsibilities included oversight of Healy.

Subsequent to the methane explosion, the Occupational Safety & Health Act itself was amended to increase substantially the fines and penalties. In such circumstances, the need for competent legal and expert help is even more important.

Lesson 9: The Greater The Injury, The Higher The Profile And The Greater The Exposure

Accidents involving serious injury or death are high profile, media intensive, legally risky events. Following the methane explosion, there was extensive coverage in local newspapers, radio, and television, including editorials about CH2M HILL's role in the events leading to the explosion.

The best advice is to get competent help early in the process to mitigate such problems. This may require internal management independent of the project, independent technical consultants with specialized knowledge (e.g., tunneling regulations, standard of care for design professionals in construction, OSHA regulations), or lawyers who are professionally knowledgeable.

Lesson 10: There Are Ambiguities In The Osh Act, The Osha Regulations, And The Applicable Case Law Which May Result In Greater Exposure

Those engaged in work which is subject to the OSH Act need to recognize the ambiguities in the act itself, the applicable OSHA precedents, and the OSHA regulations (part 1926). Throughout the legal proceedings, CH2M HILL was consistently faced with these inconsistencies and ambiguities, despite what we believed were clear contractual responsibilities among CH2M HILL, S.A. Healy, and the MMSD.

CONCLUSION

Firms involved in construction must recognize that the construction industry in general is changing. More and more owners in the marketplace are seeking firms who can provide design-build experience, engineer-procure-construct (EPC) experience, or what is loosely described as "one stop shopping." The traditional consulting work which has long been the main stay of firms like CH2M HILL is diminishing, replaced by these other forms of contracting.

Along with these changes, there will be significant changes in the area of health and safety as well. The lines of demarcation between a traditional design professional's responsibilities and a contractor's responsibilities with regard to health and safety are becoming more and more blurred. Those who continue to pursue construction projects will necessarily be required to have a higher degree of expertise in health and safety issues to continue to meet their clients' needs.

CH2M HILL has learned a great deal about health and safety issues in the painful aftermath of the Milwaukee methane explosion. We have learned from our outside lawyers and expert witnesses; we have learned from the court proceedings on the OSHA citations; we have learned from our clients what their expectations are with regard to our role on construction sites; and we have learned from our project managers and project personnel how difficult it is to separate roles and responsibilities with regard to site health and safety during the course of construction.

In order to continue to meet our clients' needs, we will continually engage in improvement of our health and safety programs and continue to

educate our employees about the importance of understanding and adhering to our contractual responsibilities.

2-4

BENEFITS OF A CUSTOMIZED HEAT STRESS PROGRAM ON JOHNSTON ATOLL

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ABSTRACT

Base reconstruction in the aftermath of Hurricane John presented a unique opportunity to monitor employee heat strain on Johnston Atoll (JA). Employees worked 12-16 hour days performing heavy construction work during the September 1994 restoration, including work in protective clothing and respirators, and in confined spaces. From a maintenance contractor's perspective, hurricane restoration activities presented a worst-case scenario for employee heat exposure. Published data on work-rest regimens were highly restrictive and not feasible. Safety and Health (S&H) staff collected environmental temperature indices (wet-globe) to determine environmental conditions, and biological samples (oral temperature, urine specific gravity, heart rate) to determine employee heat strain. A customized work-rest regimen was then formulated for JA's workforce. Education and awareness training was an integral part of the heat stress program, with the goal that individuals developed awareness of how they are felt (heart rate, sweat rate, dizziness, cramping, thirst), and of their need to drink plenty of water and take regular breaks. Training was informal, most information was exchanged while S&H staff visited the worksite to collect samples. The customized heat stress program provided the following benefits: sensible work-rest regimens were made in accordance with the heat tolerance of the workforce; heat-related illness did not occur; work was accomplished in a reasonable time frame; employees perceived they were cared for.

INTRODUCTION

Heat stress is a potential health hazard for DOE contractor employees in many different working conditions and circumstances. Base restoration in the aftermath of the August 1994 Hurricane John, presented a unique opportunity to monitor employee heat strain on Johnston Atoll (JA), a small Pacific island group located 800 miles southwest of Hawaii, operated by DOE/DNA/DOD. From a maintenance contractor's perspective, hurricane clean-up and restoration work, which occurred from late-August through mid-October 1994, presented a worst-case scenario for heat exposure in general maintenance and construction.

JA's climate is generally hot and humid, with steady tradewinds. August and September traditionally are months with the highest air temperatures of the year, and are times when tradewinds die down. During JA's restoration, Safety and Health (S&H) personnel monitored environmental conditions using a wet-globe thermometer (botsball), and collected biological measurements on employees (oral temperature, heart rate, urine specific gravity) to determine heat strain. These data were used to formulate a customized work/rest regimen that had the objective of preventing heat-related illness, and, accomplishing site restoration in a timely manner.

WORK ACTIVITIES AND CONDITIONS

Work activities performed during the restoration included:

roof repair and resurfacing (hot tar);
utility repair (power, water, sewer, communications);
facility demolition, renovation, and repair;
asbestos removal and repair;
debris pick-up and disposal.

Work conditions are described as follows:

12-16 hours workdays, six to seven days per week;
working outdoors in full sun performing manual labor;
working in full sun on asphalt roofs spreading hot tar;
working in confined spaces;
working indoors and outdoors wearing protective clothing and respirators;
working indoors, without operative ventilation systems.

ENVIRONMENTAL CONDITIONS

Environmental temperature indices such as the wet-bulb globe temperature (WBGT) have traditionally been used to determine employee work/rest regimens, even though these indices have not always correlated well with human responses to heat (1,2). Other environmental indices include the wet-globe (botsball). The botsball index combines the physical parameters of air temperature, humidity, wind, and radiant energy from heated surfaces.

To avoid heat stress, the American Industrial Hygiene Association (AIHA), the American Conference of Governmental Industrial Hygienists (ACGIH), and the U.S. Army recommend work/rest regimens that are based on WBGT indices. WBGT indices easily convert to wet-globe (botsball) indices using the following conversion formula:

(1) WBGT = 0.0118B - 0.560B + 54.9F Where B = botsball temperature index;
F = degrees Fahrenheit.

TABLES I - III represent the AIHA, ACGIH, and the U.S. Army's work/rest regimens, converted to wet-globe temperature (botsball) indices, in degrees Fahrenheit (F).

TABLE I
TABLE II
TABLE III

Wet-globe (botsball) temperature indices on JA during the restoration ranged between 78 and 90 degrees Fahrenheit (F). Botsball temperature indices varied throughout the day. Average index values during the restoration were:

inside buildings 81.7 F;
inside confined spaces 82.3 F;
outdoors 84.1 F;

Based on information in Tables I-III, botsball indices in the low to mid 80's (F) call for implementation of work/rest regimens, rather than allowing continuous work with short hourly breaks. Given the average botsball readings during the restoration, the recommended work/rest regimens were not feasible for two reasons:

1. site restoration and critical military operations would not be accomplished in an acceptable timeframe;
2. the established work/rest regimens were not well accepted by the JA workforce, who maintained they were used to the conditions.

HEAT STRESS SURVEY AND SAMPLING STRATEGY

Based on the AIHA recommendations for continuous work, S&H staff initiated the survey under a regimen of a 10 minute break after every

hour of work. Breaks were taken in the shade, and cold water and electrolyte drinks were available at all times in the work area. S&H staff conducted environmental and biological monitoring to determine employee heat tolerance, and to establish a feasible work/rest regimen for the JA workforce, one that would meet both the mission's timeframe, and with employee acceptance.

Individuals were monitored for internal (oral) temperature, external body temperature (patch), heart rate, and urine specific gravity (USG). Environmental temperature indices (botsball) were also recorded. Samples were collected on 17 different days, in a variety of locations, on different employees, at different times throughout the day (see attached data).

Employees were counseled on the effects of heat stress and made aware of the body's warning signs (sweating, prickly heat, headache, muscle cramps, etc). They were allowed to take additional break time when they felt the need to rest, cool off, and replenish fluids.

DESCRIPTION OF MEASUREMENTS

BOTSBALL THERMOMETER A direct reading wet globe thermometer that combines air temperature, humidity, wind, and thermal radiation into a single index value (degrees F) that reflects environmental conditions.

ORAL THERMOMETER A direct reading internal body temperature indicator (degrees F). In general, an oral temperature > 100 F may indicate heat strain.

TEMPERATURE PATCHA direct reading external body temperature indicator that is supposed to correlate with core body temperature. Temperature patches are affixed to the forehead and a colorimetric change indicates core temperature (F).

URINE SPECIFIC GRAVITY A direct reading checkstrip is immersed in a fresh urine sample; a colorimetric change indicates specific gravity. In general, USG > 1.027 grams per cubic centimeter indicates dehydration, a symptom of heat strain.

HEART RATE A direct reading of the number of heartbeats per minute.

At rest, most men have a heart rate of 60-70/women have a heart rate of 68-78. A working heart rate greater than 100 for men/110 for women, may indicate heat strain.

HEAT STRESS SURVEY FINDINGS

During the survey period, most employee core body temperatures remained normal (98.6 F +/- 1). When a high oral temperature or heart rate was measured, the individual was removed from the work area and allowed quiet time to recover. If/when an employee developed a heat rash, they were advised to shower/change clothes several times per day. Employees found that wet-wiping of their head/upper body was also an effective cooling measure.

Temperature patch readings did not provide useful data in this survey. There was not good correlation between the patch temperature and the oral temperature, so their use was discontinued.

When high USG's were encountered, individuals were counseled on taking water or electrolyte drinks, even if they did not feel thirsty, and

encouraged to get proper rest. They were advised that they could better rehydrate if they continued drinking water throughout the day and evening, and avoided substances containing alcohol and caffeine.

A common premise in heat stress programs is that people know when they need to take a break. During the survey, it was found that employees were so focused in their work that they did not take enough breaks, or enough fluids. Over the course of sampling, employees were shown when they were dehydrating, even though they said they felt fine. This surprised some employees and opened the door to discussion, a valuable part of awareness training.

The data showed there was an initial period when employees just returning to JA show higher oral temperatures, USG's, and heart rates, than acclimatized workers. This is consistent with findings described in the literature (1,2). This means that newcomers/returning workers need acclimatization time, about 10 days. During the acclimatization period, there needs to be a more restrictive work/rest regimen, and employees need to be counseled on heat stress awareness.

The survey indicated that employee awareness and education played a paramount role in a practical and effective heat stress program. The goal of a practical heat stress program is to make employees aware of the hazards, aware of the warning properties, aware of what their bodies are telling them, and aware that rest and water intake provide balance to the stresses they are subject to in the workplace. Information in the heat stress awareness program was provided during informal workplace conversations, which was a good way to reinforce the taking of breaks. Based on the data, the regimen used in this survey was appropriate to the heat tolerance of JA's acclimatized workforce, and was effective in preventing heat-related illness.

DISCUSSION

Sampling instruments used in this survey were inexpensive, portable, and provided instantaneous direct readings. Botsball thermometers cost about $100. Oral thermometers and urine check strips can be obtained at most drug stores for a nominal fee. Heart rate can be measured by taking a pulse and using a wristwatch with a second hand.

In the process of collecting heat stress data, S&H personnel spent time on the worksite talking with employees about their work, their sample results, how they felt, etc. This fostered an atmosphere of caring in the workplace. The measurements provided instantaneous feedback to employees that was meaningful to them, letting them know how their bodies were responding to their environment and what they needed to do to take care of themselves.

Employees were encouraged to be self-regulating with respect to taking breaks/fluids. Some employees felt empowered when allowed to be self-regulating. Malingering complaints were not a problem, instead, JA's employees had to be reminded to take more breaks.

When botsball or WBGT indices reach the low 80's, line operations can be informed of the need to implement a regimen that includes hourly breaks, and to provide cold water/electrolyte drink at the worksite. Tasks can be evaluated qualitatively for energy expenditure demands.

Line operations can then present heat stress awareness information during their regularly scheduled safety meetings, and S&H staff can visit the worksite and talk with employees about the key points of the heat stress awareness program and/or collect biological measurements. Employees performing tasks with significant energy expenditures, or with greater

potential for heat strain, can be individually counseled on heat stress symptoms and preventive measures.

Using the biological monitoring approach and measuring employee heat strain, the workforce tolerance to environmental conditions can be determined, and feasible work-rest regimens can be established. WBGT or other environmental indicators can be used qualitatively as a trigger mechanism for conducting biological monitoring, implementing heat stress training and establishing work/rest schedules.

This data applies to maintenance and construction activities, but use of biological monitoring to establish a feasible working schedule has potential for other applications. For example, S&H staff also used this formula to monitor non-typical operations, such as projects where employees have to wear protective clothing that increases their heat load (tyvek, anti-C's, etc).

CONCLUSIONS

1. The most valuable data to assess employee heat strain was found to be oral temperature, USG, and heart rate.
2. It was found that employees need a few weeks to acclimatize to changes in their physical environment, even if they have only left the environment for a few weeks, such as during vacation time.
3. Employee education and awareness is important in an effective heat stress program; informal workplace conversations are a valuable forum for information exchange.
4. A regimen for continuous work with hourly breaks was implemented and tested with biological monitoring. Biological monitoring showed employees when they needed to rest/replenish fluids, and was successful in preventing heat-related illness. Environmental temperature indices can be used to indicate when a biological monitoring program may be of benefit.
5. Temperature patch readings did not provide useful data in this survey.

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Session 03 -- PANEL -- PRIVATIZATION PERSPECTIVES
Co-chairs: A. Lowell Snow, Consultant
J. Ed Day, ADTECHS
Manuscripts not available for publication

Session 04 -- THE PLUTONIUM ISSUE: STORE, BURN OR VITRIFY
Co-chair: W.A. (Bill) Seddon, AECL Technologies Inc.

4-1
THE PU-ISSUE: GOING MOX OR ALTERNATIVE SOLUTIONS?
MATERIALS SCIENCE ASPECTS

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ABSTRACT
At present, an excess of plutonium exists worldwide, due to reprocessing of spent commercial LWR fuel and to Pu arising from dismantling of nuclear warheads, while the construction of fast reactors is delayed. If the reprocessing capacity available after 2000 is fully used the amount of Pu from civilian spent fuel could reach 20 t Pu/yr in 2005. The increase of the inventory from dismantled warheads presently being discussed is 10 t Pu/yr from 1996 to 2005 thus making available 100 tons of fissile Pu. The decision of how to store or how to recycle this Pu is mainly a matter of politics. In the present paper, the materials science aspects of three options are discussed: going MOX, alternative fuels, or vitrification. Work in the materials research of these three fields has been performed at the European Institute for Transuranium Elements for more than two decades.
Going MOX: present and near-future MOX fabrication capacities existing in Europe and Japan are sufficient to accommodate between 15 t Pu/yr (lower limit) and near to 25 t Pu/yr (upper limit). MOX fuel fabrication and Pu recycling are becoming a mature industrial activity. The extensive irradiation experience with MOX fuel is positive. To make (conventional, hence UO2-based) MOX fuel more attractive and to achieve a high Pu disposition rate, the fuel cycle costs should be reduced and high core loadings should be achieved. This asks for high burnup in a once-through cycle with automated or dust-free fuel fabrication, e.g. with the sol-gel process.
Alternative fuels: An inert matrix completely free of U-238 (e.g. spinel MgAl2O4, CeO2 or others) containing e.g. 5 % Pu allows burn-out of Pu to a large extent. These inert matrices can also be used for transmutation of the minor actinides (Np, Am, Cm), provided they fulfill a number of criteria: their thermal conductivity and their mechanical properties should be similar or better than those of UO2, they must be compatible with clad and coolant, fabrication procedures must exist and, importantly, they must show an acceptable behavior against radiation. Based on measurements of these properties, suitable matrices can be selected.
Vitrification: A large series of actinide-containing borosilicate waste glasses was produced in the European Institute for Transuranium Elements in the past 20 years. Solubilities of actinides, relevant mechanical, thermal and chemical properties were measured and the effects of radiation damage due to a-decay of the actinides was followed, mainly in curium-doped glasses thus simulating the behavior of HLW glasses for 105 yrs storage, or of glasses containing 7 wt% Pu for up to 2000 yrs storage. This large data base is discussed, as is the possibility of incorporating Pu in suitable ceramics instead of glasses.
INTRODUCTION
The presently existing excess of plutonium can a priori be dealt with by adopting one - or a combination of - three options: going MOX, burn Pu in alternative fuels, e.g. in an inert matrix, or vitrify and bury Pu. The decision on any of these options will have to be made by politics, but should be based on engineering, neutron physics and materials science aspects as well. It is this latter aspect the present paper is concerned with.
Current reprocessing commitments represent arisings of Pu from civilian spent fuel (reactor plutonium, R-Pu in the following) of roughly 15 tons per year up to 2000 and on average of 12.5 tons per year from 2000

onwards. However, if full use is made of the reprocessing capacities existing after the year 2000, this amount might increase to 20 t Pu/yr in 2005. This R-Pu contains typically only about 65% Pu-239. For instance, when reprocessing PWR UO2-fuel with a burnup of 33 GWd/tonne, the isotopic composition is ~ 1.3 % Pu-238, 62.8% Pu-239, 23.5% Pu-240, 8.3% Pu-241 and 4.1% Pu-242 or, at increased burnup in a pressurized water reactor, the composition is 2.6% Pu-238, 54% Pu-239, 24% Pu-240, 12% Pu-241 and 7.4% Pu-242. To the above amount of R-Pu, the dismantling of warheads might add ex-weapons Pu (W-Pu in the following) at a rate of 10 t Pu/yr from 1996 to 2005. W-Pu contains typically 93 % Pu-239, 6 % Pu-240 and 1 % Pu-241. Its critical mass is therefore smaller than that of R-Pu, the spontaneous neutron emission rate and the heat production are significantly smaller and it therefore deserves particular attention to guarantee proliferation resistance, as well as an expeditious decision on the strategy of its dispositioning. Given the present unavailability of fast reactors and of accelerator-driven reactors, there are essentially two ways to effectively prevent W-Pu from being used for warheads production: I) use it as MOX fuel in civilian power plants to bring it to the level of spent fuel, hence converting it to R-Pu grade while, at the same time, mixing it with fission products and keeping open the options of direct disposal in a repository as spent fuel or intermediate storage of the spent fuel for later reprocessing. Using inert matrices rather than UO2-based MOX fuel can increase the disposition rate. Or: ii) dispose of W-Pu without producing electricity by applying a process which causes the material to be inaccessible for diversion, e.g. by blending the Pu with fission products, to vitrify the mixture, as it is done for high level liquid waste from reprocessing, or to solidify it in a ceramic matrix and to dispose of the product in a safe, deep geological repository. To increase the proliferation resistance for intermediate storage for option I) and to reduce the storage costs, W-Pu could be mixed with R-Pu. The blend has a higher heat production and a higher rate of spontaneous neutron emission than W-Pu due to the introduction of Pu-238 and of Pu-242, and the higher content of Pu-240. In addition, the higher content of Pu-241 decaying into Am-241 leads to an increase in g-activity and thus to an easier detectability in case of illicit traffic. The emphasis of the present paper is placed on materials science aspects of these options rather than on such strategies themselves. Research on fabrication and materials property studies in the areas of MOX-fuel, inert matrices, vitrification and solidification in ceramic matrices has been performed in the European Institute for Transuranium Elements for more than two decades. In the following, we will concentrate in particular on fabrication and R&D for inert matrices and vitrification, realizing that other aspects such as nuclear safety, licensability and economic viability are of large importance as well. Frequently, e.g. in neutronic considerations, the non-fissile, non-fertile matrix is treated as inert to the extent that its materials properties are not considered at all. This is particularly true for its radiation stability, which will therefore be treated in more detail in this paper.

UO2-BASED MOX FUEL

Fabrication and irradiation experience of Pu-recycling as UO2-based mixed oxide (MOX) fuel, i.e. UO2 with 3 - 5% PuO2 exist now for about 30 years. For instance, in Germany, Pu-recycling started in BWRs (Kahl) in 1966 and in PWRs (Obrigheim) in 1972. In Belgium, it started even earlier in 1961. The fabrication experience covers rather different Pu-compositions, from

Pu from reprocessing MAGNOX fuel with a high content of Pu-239 to the Pu mentioned above from reprocessing of LWR fuel. Through the years, the fabrication techniques were optimized to improve the early mechanical blend MOX fuel, in order to reduce hot spot problems due to PuO2-rich inclusions in the fuel and to achieve a better solubility in HNO3 in case reprocessing is to be done. As examples, the ammonium uranyl plutonyl carbonate (AUPUC) process of powder preparation and the optimized co-milling (OCOM) process were developed and tested by Siemens, and the micromized masterblend (MIMAS) process was developed and tested by Belgonuclaire. Satisfactory Pu-solubility (>99%) already in the unirradiated pellets was achieved and a good irradiation performance could be proven (1, 2). The existing and projected MOX fuel fabrication facilities are summarized in Table I. They are all situated in Europe and in Japan. The plants at Hanau are mentioned without numbers because of the known political problems. Facilities originally built to produce MOX for fast reactors can be included in this consideration, since modifications to change from FBR MOX (typically 20 - 25% PuO2) to LWR MOX can be introduced, and dual purpose production lines are also feasible. The table shows that the MOX fuel fabrication capacity may be able to catch up with Pu-arisings when considered globally, though the stockpile will temporarily increase if W-Pu becomes available at the rate of 10 t/yr for 10 years, the amounts depending also on the Pu-content used for the MOX fuel. The equilibrium between production and consumption of Pu would probably be achieved around 2015 rather than 2005 for R-Pu alone.

Table I

To achieve a good homogeneity, i.e. a solid solution of PuO2 and UO2 in the MOX fuel, coprecipitation could be used, or the sintering program could be modified, e.g. by varying the oxygen potential during the sintering process. Increasing the oxygen potential increases interdiffusion rates (3, 4). A subsequent reduction step at lower temperatures can be used to bring the material back to the stoichiometric composition, since oxygen diffusion in UO2 and in MOX is much faster than U and Pu (inter)diffusion, e.g. D0/DM 108 at 1200C, M = U or U+Pu. To make MOX fuel more attractive, fabrication costs have to be kept low, and, for R-Pu, the increasing concentration of the g-emitter Am-241, formed by decay of Pu-241 with a half-life of 14.9 years, has to be taken care of. If Am-241 is not separated from R-Pu stored for some time, a dust-free fabrication process is advantageous. Such a process exists and has been demonstrated: the sol-gel process, or the gel-supported precipitation process (5).

The irradiation performance of MOX fuel is also very encouraging and satisfactory (1, 2, 4). Experience exists from experimental irradiations and from many MOX fuel assemblies loaded in thermal reactors in Japan, Germany, France, Belgium, Switzerland and USA. Some of these went to extended burnups. These assemblies and transient-tested MOX fuel showed a dimensional behavior comparable to that of UO2 fuel. Very few fuel rod defects have occurred, and these were neither specific for MOX fuels nor did they have their origin in the use of MOX fuel. Some features of the performance deserve further study, e.g. the fission gas behavior in Pu-rich MOX inclusions, the behavior of such MOX agglomerates at the pellet surface if they are not fully surrounded by the UO2 matrix etc. Also, though there is no significant difference between MOX and UO2 fuel for fission gas release at comparable power densities and power histories, up to 40 GWd/tM, the MOX fuel has a greater power at higher burnup because

the change in reactivity with burnup is less than for UO2 fuel. The fuel rod design should allow for the space needed to appropriately accommodate the larger amounts of released fission gas at extended burnups.

As of today, about 35 LWRs are licensed for MOX fuel, most of them in western Europe. For another 25 power stations licensing is underway, usually for reloads with 1/3 MOX fuel. Many more (about 100) power stations are potential candidates to operate MOX fuel since there are no fundamental technical problems for licensing. By adequate design and by suitable distribution of the MOX fuel elements in the reactor core, no adverse effects result on reactor control and operation for the 1/3 MOX option. The higher inventory of Pu and actinides does not affect reactor safety. 100 % MOX cores are also possible when adjustments and changes of core design and equipment for reactor control are made (1).

Going MOX is thus an existing option to significantly reduce stockpiles of either W-Pu or R-Pu, and to reach equilibrium between production and dispositioning to the spent fuel standard. To make this option economically attractive the fuel cycle costs should be reduced: automated, or dust-free fabrication (sol-gel), high burnup fuel, high core loadings are the aims to achieve a high Pu-disposition rate. Comparatively like R&D in materials research is necessary given the advanced state of fabricating and irradiating MOX fuel which exists already today.

ALTERNATIVE FUELS, INERT MATRICES

An inert matrix is a material which replaces conventional UO2. It should not form Pu or other actinides, but otherwise have properties similar or superior to those of UO2. Hence, it should have a high melting point, good thermal conductivity and mechanical properties, and it should be compatible with the cladding and the coolant. Also it should have a low neutron capture cross section to achieve a good neutron economy, and, importantly - but often not considered to the appropriate extent - it should be resistant against radiation (fission, neutrons and a-decay of Pu.). To avoid actinide buildup, all the components of the inert matrix should have atomic numbers significantly smaller than 92.

Since no new actinides are formed, a large degree of burnout can be achieved if irradiation is carried out for a long enough time. This possibility and the advantage of inert matrices are exemplified with a novel proposal (6) to destroy up to 95% of the W-Pu in LWRs while at the same time increasing the proliferation resistance of the spent (inert matrix) fuel by forming high levels of Pu-238 in the small quantity of remaining Pu. As shown in Fig. 1 (left and middle parts), a fuel with an inert matrix and with 2.5% each of W-Pu and W-U (i.e. U enriched to 94% U-235), irradiated for approximately 2000 d together with standard fuel of 900 days lifetime in a PWR shows a burnout of 99.4% of Pu-239. Of the originally added 25 kg W-Pu/t (23.5 kg Pu-239), only about 6% of total Pu-remain. Of this small amount, 28% are Pu-238, 57% are Pu-242, and only 6% (or 100 g) are Pu-239. The isotopic composition of Pu - even if separated from the inert matrix - is very proliferation resistant because of the large content of Pu-238, mainly originating from irradiating the added W-U, and of Pu-242, yielding very high heating rates in excess of 150 W/kg Pu and a neutron emission rate which is a factor of 32 higher than that of W-Pu, respectively (6). The W-U is also transferred into proliferation resistant form since the U-235 content of the small remaining U-mass is only 17%, hence below the 20% level considered proliferation-safe. Furthermore, as shown in the right part of Fig. 1, a

similar irradiation of the inert matrix with 10% R-Pu leads also to a large burnout (7). An LWR with 1/5 of the assemblies being made from such fuel - the remaining part being standard UO2 fuel, can be shown to have a balance between Pu-production and Pu burning (7).
Fig. 1
In such kind of calculations, materials properties of the inert matrix itself are usually not considered. Significant R&D is still necessary to select suitable matrices for given irradiation schedules. Also, improved or unconventional clad materials have to be selected and tested to achieve the desired high burnups. Significant work in this area is presently ongoing in the related field of transmutation of Am and Np in nuclear reactors. If these minor actinides are partitioned from the reprocessing HLW, they can be transmuted by fast neutrons from fast reactors or accelerator-driven reactors, or - usually at a smaller rate - in thermal reactors. Inert matrices are considered for this purpose as well, e.g. in the large international project EFTTRA (Experimental Feasibility of Targets for Transmutation) (8), providing experience and a data base of use for Pu-dispositioning as well.
Materials suggested and fabricated as inert matrices include binary oxides (e.g. Al_2O_3, MgO, CeO_2, ZrO_2), ternary oxides (spinel $MgAl_2O_4$, Zircon $ZrSiO_4$ or monazite $CePO_4$), complex oxides (e.g. PuO_2-stabilized ZrO_2-Al_2O_3-MgO or a two-phased mixture of a fluorite phase (PuO_2-ThO_2, PuO_2-ZrO_2) with Al_2O_3 (9), and binary carbides and nitrides (e.g. SiC, Si_3N_4, ZrN etc.).
Basically, two types of matrices exist, a homogeneous case, i.e. a matrix which forms a solid solution with Pu, e.g. CeO_2, zircon ($ZrSiO_4$) or ZrN. If Pu is not soluble, as in Al_2O_3 or MgO, a heterogeneous fuel is formed with PuO_2 precipitates as a second phase. This resembles to some extent the conditions of the early MOX fuel. For a final selection of suitable matrices, their properties (thermal diffusivity, specific heat, thermal conductivity, mechanical properties, hardness, fracture toughness, absence of chemical reactions with clad and coolant) have to be measured, not only on the matrix alone, but also on the fabricated fuel including Pu. This leads to exclude some of the candidates for specific conditions, e.g. MgO for LWR and CeO_2 for FBRs because of reactions with H_2O and Na, respectively, or to concerns as for stabilized ZrO_2 because of its low thermal conductivity. Furthermore, the solubility in HNO_3 is of interest if reprocessing is foreseen, whereas very low leach rates in ground water are asked for if a once-through option is selected. This, however, is not sufficient since information on the above properties are also necessary for the Pu-matrices subjected to radiation damage. Sometimes, data are available for neutron damage, e.g. for MgO, for Al_2O_3 and for spinel $MgAl_2O_4$. Stability against neutron damage alone is not sufficient, since the inert matrices will also be subjected to a-decay damage (5.15 MeV a-particles and 88 keV U-235 daughter recoil atom in the decay of Pu-239) and fission damage (fission products of 70 to 90 MeV). a-decay occurs following fabrication during storage before and after reactor use, essentially at room temperature when damage recovery is small. To give an example: $(Pu\text{-}239)O_2$ shows a decrease in thermal conductivity of 60% after one year of storage. Fission damage occurs at reactor operating temperatures. Some potential matrices can be shown to be structurally unstable or to swell to a large extent. An example is Al_2O_3 which swells by about 30% under fission fragment impact and which gets amorphous after rather short irradiations 10. As another example, spinel ($MgAl_2O_4$) known

to be very stable under neutron irradiation, the reasons for this stability being understood, swells also under fission product impact and undergoes polygonization, i.e. a state with very small grains is formed, if irradiation is performed at ambient temperature. The behavior is, however, more benign at elevated temperatures, e.g. T>500C.
Work in this field is ongoing and will lead to a selection of good matrices.

VITRIFICATION

Work on incorporation of actinides (Th, U, Np, Pu, Am, Cm) into borosilicate waste glasses and property measurements of such glasses have been performed in the European Institute for Transuranium Elements since 1975 (11). This work was done in the frame of R&D studies to vitrify high level waste from reprocessing of spent LWR fuel. In this HLW, the actinide content is small, i.e. the sum of UO_2, NpO_2, PuO_2, AmO_2 and CmO_2 is < 1 wt.% and can safely be assumed to be in solution. For this work, a number of different waste glasses was used, including the German (simulated) product GP 98/12 and its precursor (see Table II).

Table II

To determine the solubility and/or to observe possible segregation of PuO_2 at higher Pu-contents, the precursor product GP was produced with up to 10 wt.% PuO_2 by mixing the proper quantity of the glass frit with or without denitrated waste solution and with Pu (or also Am) (12). Melting was performed in Pt-crucibles. The resultant glass was pulverized and molten again in graphite crucibles at 1150C. Discs were cut from the glass cylinders produced in this way. Alpha-autoradiographs showed no visible segregation for the highest AmO_2 content used, i.e. 4.9 wt.% AmO_2, whereas very obvious segregation (spots very rich in PuO_2) were seen when 10 wt.% PuO_2 were added. Additionally, a base glass without HLW and the glass with very high HLW loadings, i.e. either 20 wt.% HLW, or 20 %HLW with additionally 10 wt.% Gd_2O_3, were produced with 5wt.% UO_2 or 5wt.% PuO_2, respectively. 80 mm long cylinders were kept molten (1175C) in Pt-crucibles for 25 h. Subsequently, discs were cut from the bottom, the middle and the top section of these rods and analyzed by electron probe microanalysis, epma. These analyses showed the absence of sedimentation of actinides. In contrast, the noble metal particles were enriched in the bottom section.
Glasses doped with 5 wt.% AmO_2 or 3 at% UO_2 were still vitreous with no crystalline phase being present. In contrast, the glasses containing 5 wt.% PuO_2 consisted all of two phases, the major vitreous phase and a minor crystalline phase. In detail

- the base glass contained 4.6 wt.% PuO_2 in solution. The second phase was PuO_2
- the glass with 20% HLW contained less PuO_2 in solution, i.e. 4.4 wt.%, the crystalline phase being again PuO_2
- the glass with 20% HLW and 10 % of the neutron poison Gd_2O_3 showed a very small solubility of only 1.2 wt.% PuO_2 and contained segregated crystalline phases of an (average) composition of $(Pu_{0.68}Ce_{0.2}Gd_{0.12})O_2$.

These solubility values are large compared with the actinide contents in HLW solutions, but they may be of concern for the fabrication of Pu-containing glasses, if fission products (in particular rare earths) are added to these glasses.
Thermal and physicochemical properties and leaching of these and other waste glasses have been measured extensively and the behavior against a-decay damage including He-accumulation and release have been measured

(13, 14). Figure 2 shows the relation between the accumulation of a-decay damage (a-decays/m3) and the storage time for

the test glass containing 1.5 wt.% Cm-244 (lower abscissa)
the same glass containing 7 wt.% Pu-239 (upper abscissa)
typical HLW from reprocessing vitrified in the same glass (upper abscissa).

Fig. 2

The damage studies were terminated at damage levels of 15 to 25x1024 a-decay/m3. A damage level of 15x1024 a-decays/m3 will be reached in the glass with 7% Pu-239 in about 1000 years.

Damage due to a-decay is caused by a-particles and by recoil atoms. a-particles loose 99% of their energy of 5.15 MeV (for Pu-239) by ionization, have a range of ~20 mm in glass and produce only ~200 atomic defects, mainly at the end of their range. The recoil atoms (U-235 in the decay of Pu-239) have an energy of ~90 keV. About 90% of this energy is used to produce 1200 to 1500 displaced atoms thus forming a dense collision cascade along their range of ~25 nm. This type of damage can be simulated with ion implantation techniques. This was done extensively at ITU, in addition to the work with the Cm-glasses. The properties measured were: swelling, leaching, hardness, fracture toughness, Youngs modulus, Poisson ratio, damage recovery, recrystallization and He-release. Damage ingrowth could be fitted with equations of the type Dp = A(1-exp{-BD}) where p = property, A = change at damage saturation, B = fraction of glass totally damaged by a-decay and D = a-dose.

The main results can be summarized as follows:

Swelling: volume changes at saturation damage were very small (~0.5%)

Leaching: Figure 3 shows that the leach rate does not increase with damage level, it is even indicated to decrease (leaching conditions: static test, demineralized H2O, 14 d, 150C), thus overcompensating the small difference between the glasses without Cm and the ones with Cm for no or very small damage levels. Figure 4 confirms this trend showing results for 4 elements (Si, B, Ca, Mo) of a total of 30 elements analyzed, all confirming the absence of observable damage effects.

Fig. 3

Fig. 4

Mechanical properties: a rather fast decrease in hardness by about 30% with saturation at ~3x1024 a-decays/m3 was observed. At the same time, crack probabilities were significantly reduced and the fracture toughness KIc increased by up to 100%. Youngs modulus decreased also by ~30%. The net result of damage formation was thus reduced brittleness and better resistance against crack formation, i.e. technologically positive effects.

He-release: glasses with 1.5 wt.% Cm-244 which accumulated He as product of the a-decay for about 9 years showed no swelling when annealed for 10 h at 300C. He-release was measured in a Knudsen-cell: between 300 and 420C, release was compatible with thermally activated diffusion of single He-atoms, above 420C and up to 550C, most gas was released with kinetics compatible with gas bubble diffusion. Some He remained in the glass up to ~ 1000C. Note that glass with 7 wt.% Pu-239 would accumulate such a He-level only after ~ 2500 years.

These results cover many of the questions to be answered when vitrifying actinides. One should, however, keep in mind that vitrification does not change the isotopic composition of Pu, and that some further questions have to be answered. The main areas are: while Pu is relatively insoluble

in water, boron is known to show high leaching rates and to become depleted relatively deep in the glasses. B, as a neutron absorber, helps to prevent criticality problems, and should not become depleted. Then, if g-emitters are added to the glass to increase the proliferation resistance, the still open question of the effect of ionizing radiation creating (oxygen) bubbles and the possible consequences on the behavior of the glass should be answered.

Ceramic matrices for direct storage: a large number of ceramic matrices, including SYNROC, which accumulated a-damage due to decay of incorporated Cm-244, have been studied in ITU. Typically, significant swelling (by up to 9%), with the related risk of cracking was found. Also, metamictization (loss of crystallinity and formation of amorphous phases) was often observed at high damage levels (corresponding to 4000 to 10000 years of storage for a 7% Pu-ceramic) with a parallel increase in leach rates. Careful study of damage effects is therefore necessary if additional new ceramics are suggested. Some of the inert matrices discussed above could also be used for solidification and disposal of Pu, e.g. CeO_2, $CePO_4$ and, in particular, zircon ($ZrSiO_4$) because of their good resistance against corrosion in water. For zircon, a good knowledge exists on a-damage effects due to investigations of both old, naturally occurring zircon minerals with a certain content of U and of synthetic zircon with ~ 10 wt.% Pu substituting for Zr (15). Additional work is also needed to define the occupied lattice sites in complex ceramics, the need having been shown for Pu in SYNROC before (16).

SUMMARY

For expeditious dispositioning of Pu, going MOX is a mature and technically proven possibility. Comparatively little R&D in materials science is necessary given the advanced state of fabricating and irradiation MOX fuel in thermal reactors.

The use of inert matrices promises a higher destruction level of Pu. Suitable candidate materials exist and are being tested. More materials research, however, is needed before a final selection can be made, e.g. in the field of radiation damage.

Some of the inert matrices could be used for solidification of Pu, as an alternative to vitrification. A large data base for both glasses and ceramics exists at ITU, and in the literature, to suggest materials and loading levels with Pu. For a fully optimized, tailor-made material some more research is necessary.

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USING THE BRUCE CANDU REACTORS FOR WEAPONS PLUTONIUM DISPOSITION

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ABSTRACT

The overriding objective of U.S. and Russian disarmament initiatives is to dispose of nuclear weapons in such a way that they cannot be reused or reconfigured in the country of origin, or sold to others, for military purposes.

Significant progress has been made in achieving this objective for high enriched uranium from nuclear weapons. Unfortunately, none of the three options now under serious consideration for plutonium disposition, (deep burial, immobilization, or MOX fuel conversion), achieves this objective if implemented in the country of origin. For the MOX option, both the U.S. and Russia, as nuclear weapons states, have the technical and physical resources to extract residual plutonium in the spent fuel via PUREX reprocessing, should future Governments so decide. For the alternative of immobilized plutonium in solid wastes and deeply buried plutonium, the two countries also have both physical and technical resources for later extraction and use of the plutonium.

One approach towards eliminating this potential future threat of reconversion in the country of origin is to implement the MOX Fuel conversion option in a non-weapons state, one with a sufficiently stable political environment to give the world assurance that the plutonium will not be reclaimed for use in nuclear weapons. The Canadian option for utilization of excess weapons plutonium as MOX fuel in CANDU reactors is currently the only option on the table which accomplishes this objective. Atomic Energy of Canada and Ontario Hydro have studied the consumption of up to 100 tonnes of weapons plutonium resulting from projected disarmament programs in Russia and the U.S. by utilizing it as MOX fuel in the Bruce A Reactors, four 825 MWe (gross) CANDU reactors operating on the shores of Lake Huron in the Canadian Province of Ontario. No changes to the reactor designs are required, the fuel is predicted to operate within the existing licensed performance envelope, and the project can begin within four to five years.

Fabrication of the relatively simple CANDU MOX fuel bundles can be accomplished by modifying existing nuclear materials processing facilities in Russia and the United States. Engineering studies of typical U.S. facilities such as the FMEF at Hanford, Washington, and the BNFP in Barnwell, South Carolina, indicate that such facility conversion can be completed, including licensing by the NRC, within a period of 4 to 5 years, and at relatively modest costs. Further studies of other U.S. facilities are underway; engineering studies of existing Russian facilities which could be used for fabricating CANDU MOX fuel have just begun.

Subject to agreement amongst the Governments of Canada, Russian and the United States, the entire program, including storage of plutonium oxide,

fabrication of MOX fuel, transportation, irradiation, and disposal of the spent fuel, would be subjected to independent third party inspection.

CANDU MOX IN BRUCE - THE BASIC TECHNOLOGY

Figure 1 presents the material flow for our reference case, which utilizes the same standard 37 pin fuel bundle design as is currently operating in most CANDU reactors. As indicated in this figure, the burnup of 9,700 MWD/T is slightly higher than the 8,300 MWD/T achieved with natural uranium, thus leading to a slight reduction in spent fuel produced.

Fig. 1

As shown in Fig. 2, the plutonium concentration is about 1.2 % in the outer 18 fuel pins, and about 2% in the next row of 12 fuel pins, with the central 7 pins containing only depleted uranium oxide and dysprosium oxide burnable poison. Preliminary analysis indicates that the thermal performance of this MOX fuel bundle during normal and accident situations will be well within the current AECB licensing envelope at Bruce.

Fig. 2

A multiphase program is planned, including a first phase involving production of up to 50 CANDU MOX bundles for testing at the ZED and NRU facilities at Chalk River. Results will provide confirmation of nuclear, mechanical, and thermal design of the MOX fuel and core design, and would be used to support safety analyses and licensing by the AECB. A second "pilot scale" phase, consuming up to 1 tonne of weapons plutonium, would produce larger quantities of CANDU MOX bundles for irradiation in Bruce. And finally, a program of full MOX production, utilizing up to 2.1 tonnes of U.S. plutonium per year, in two Bruce reactors would begin by 2002. A similar program in the Russian Federation is now being evaluated by a joint Canadian - Russian technical team, with results anticipated by summer, 1996.

An advanced design CANDU fuel bundle, called CANFLEX, is now under development. The design uses 43 fuel pins instead of the 37 pins in the reference design. Commercial validation of the CANFLEX design using natural uranium is expected by around 2000, and if successful, could then be adapted to MOX application via additional fuel qualification efforts. When applied to MOX fuel, CANFLEX would allow higher plutonium loadings and would achieve a burnup of about 15,500 MWD/T, thus permitting an increase in plutonium consumption rate utilizing the same MOX fuel production rate as in the reference case. Figure 3 presents one deployment option which will be considered for this CANFLEX case.

Fig. 3

MOX FUEL SUPPLY IN THE U.S; OPTIONS BEING EVALUATED

Atomic Energy of Canada and Ontario Hydro have been participating in DOE's Material Disposition program, which includes a Programmatic Environmental Impact Statement (PEIS), separate technical and economic evaluations, and a Record of Decision currently scheduled in early Fall, 1996. Because of the unique flexibility of the CANDU design to adapt itself to many different fuel cycles, preliminary conclusions indicate that MOX fuel can be incorporated in the design with no changes to the reactor hardware, and within the current licensing envelope. Because Ontario Hydro's Bruce A Station contains four 825 MWe CANDU reactors, it is capable of utilizing all of the U.S. and Russian plutonium at a single station, thus simplifying the logistics, including transportation, as compared with other options which require multiple sites. For these reasons, we anticipate that DOE's draft Environmental Impact Statement,

to be issued shortly, will select the CANDU option as an option to be evaluated in the final stage of DOE's decision making process.

Atomic Energy of Canada and Ontario Hydro recognized early on that the criteria for final selection will focus on economics, since only those options which meet the safety, technical viability, safeguards and security will have survived the prior screening process. Economics are driven primarily by the incremental cost of MOX fuel.

We concluded that a major cost savings could be realized if an existing U.S. facility could be converted for MOX fuel fabrication purposes. In addition to the obvious cost savings as compared to a greenfield facility, use of an existing facility allows an earlier start to the program, and in some cases has the advantage of an existing base of local support, as compared to potential difficulties of siting a greenfield facility. The facility mission for MOX fuel from weapons plutonium would require as little as 15 years followed by a short Decontamination and Decommissioning period. Therefore, long lifetime structures and components, such as used in European MOX facilities, would not be required. The entire program could thus be treated separately and independently from other, often divisive, issues such as the ultimate disposition of plutonium in civilian spent fuel.

Figure 4 shows the Bruce Reactor site, along with four existing U.S. facilities which are being examined for possible conversion to CANDU MOX facilities: the Fuels Materials Examination Facility (FMEF) at Hanford, Washington, the Barnwell Nuclear Fuel Plant at Barnwell, South Carolina, the P Reactor Fuel Building at Savannah River, South Carolina, and the DAF at the Nevada Test Site.

Fig. 4

Another deployment option being considered is to design the MOX fabrication facilities in the U.S. so they can later be expanded to produce LWR MOX fuel, thus further accelerating the full disposition of U.S. weapons plutonium. The LWR phase would begin after NRC licensing approval of a U.S. utility application for MOX fuel in a commercial LWR.

MOX FUEL SUPPLY IN RUSSIA; OPTIONS BEING EVALUATED

Under the sponsorship of the Canadian Government, Atomic Energy of Canada, Ontario Hydro and Minatom have begun a feasibility study aimed at evaluating the production of CANDU MOX fuel in Russia from excess weapons plutonium, for consumption in the Bruce Reactors. Several exchange visits have already taken place, and the study is due to be completed this summer. The study is similar to the one sponsored by the DOE in 1994. The Russian - Canadian feasibility study has three main elements: 1) MOX Fuel Supply, 2) Safeguards, Security and Transportation; and 3) Economic Evaluations. The economic evaluations, which will permit a comparison of the MOX option with other options being considered in Russia, are obviously critical to a final decision on disposition of Russian plutonium. A key element in this evaluation will be the cost of MOX fuel fabrication, including the cost of facility conversion, and the cost of transportation.

There are three pilot installations in Russia to produce MOX fuel: two at the MAYAK association in Osersk, and one at RIAR in Dmitrovgrad. The current capacity of the "Paket" installation at MAYAK allows production of 10 to 12 breeder sized assemblies per year for use in BN-600/350 reactors. Each assembly contains about 300 KG of MOX fuel containing 20% plutonium. A modified "Packet" installation at MAYAK has a capacity of about 40 breeder fuel assemblies per year (1 tonne of MOX fuel per year).

And the "Granat" installation at RIAR in Dmitrovgrad is about 1 tonne of MOX fuel per year.

In addition to these pilot scale facilities, an industrial scale MOX facility was erected at the "Complex 300" plant at MAYAK, but construction was stopped at about the 50% completion stage, due to lack of near term demand. This facility was to have a capacity of 900 breeder fuel assemblies per year, or about 60 tonnes of MOX fuel per year.

CONCLUSIONS - FIVE REASONS WHY THE U.S. AND RUSSIA SHOULD SELECT THE CANDU OPTION.

In summary, the authors believe it is in the best interests of both the US and Russian Governments to select the Bruce A CANDU reactors for disposition of their excess weapons plutonium for the following reasons:

It is the only option which affords both countries the opportunity to demilitarize their excess plutonium weapons in a third, neutral, non-weapons state.

It affords the international community assurance that this excess weapons plutonium will be under full IAEA safeguards from the moment the plutonium pit is converted to oxide feed material until the fuel is in safe geologic storage.

It can be initiated in a short period of time, less than five years, and can be completed within 15 to 25 years thereafter.

Costs for the disposition program are moderate, as compared to alternatives for plutonium disposition, and especially as compared to the prior costs of military readiness.

It requires only one site for MOX utilization, thus enhancing the safeguards and security and simplifying the transportation logistics, as compared with using multiple sites for plutonium disposition.

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DISPOSITION OF EXCESS PLUTONIUM USING "OFF-SPEC" MOX PELLETS AS A SINTERED CERAMIC WASTE FORM

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ABSTRACT

We describe a potential strategy for the disposition of excess weapons plutonium in a way that minimizes 1) technological risks, 2) implementation costs and completion schedules, and 3) requirements for constructing and operating new or duplicative Pu disposition facilities.

This is accomplished by an optimized combination of 1) using existing nuclear power reactors to "burn" relatively pure excess Pu inventories as mixed oxide (MOX) fuel and 2) using the same MOX fuel fabrication facilities to fabricate contaminated or impure excess Pu inventories into an "off-spec" MOX solid ceramic waste form for geologic disposition. The key to the combination approach is the use of a sintered ceramic waste form (SCWF) consisting of a UO2 encapsulating matrix for the excess PuO2 inventories. The SCWF will have a high probability of being acceptable for geologic disposition because it is similar to an unirradiated MOX fuel matrix without fission products. Because it can be fabricated using the same basic technologies, processing equipment, and facilities used to fabricate the MOX fuel for reactors, there is no need to construct and operate specialized and duplicative facilities using other immobilization technologies such as vitrification.

Diversion protection for the SCWF to meet the "spent fuel standard" introduced by the National Academy of Sciences (1,2) can be achieved in at least three ways. 1) One can utilize the radiation field from defense high-level nuclear waste by first packaging the SCWF pellets in 2- to 4-L cans that are subsequently encapsulated in radioactive glass in the Defense Waste Processing Facility (DWPF) glass canisters (a "can-in-canister" approach). 2) One can add 137Cs (recovered from defense wastes at Hanford and currently stored as CsCl in capsules) to an encapsulating matrix such as cement for the SCWF pellets in a small hot-cell facility and thus fabricate large monolithic forms. 3) The SCWF can be fabricated into reactor fuel-like pellets and placed in tubes similar to fuel assemblies, which can then be mixed in sealed repository containers with irradiated spent nuclear fuel for geologic disposition.

INTRODUCTION

With the end of the cold war, major planning efforts were begun for the management of the excess fissile materials that would result from the reduction of U.S. and Russian nuclear stockpiles.This was motivated by the fear that such a large inventory of weapon-quality fissile material posed a worldwide risk of nuclear terrorism or nuclear proliferation. The United States and Russia are seeking cost-effective methods for the management and ultimate disposition of these materials, with special emphasis on plutonium, that achieve each country's nonproliferation goals and that can be implemented in a timely manner. The current Russian position strongly favors burning the excess Pu in nuclear reactors. The United States is evaluating options that include, but are not limited to, burning excess Pu in existing reactors or immobilization with radionuclides in glass or ceramic matrices; in either case the final product is buried in a geologic repository. Deep-borehole (2-4km) burial of Pu in an immobilized form is also being evaluated. All these options have advantages and drawbacks. This paper outlines a potential strategy for disposition of excess Pu in two streams: burning suitably pure Pu (as MOX) in existing reactors, and disposing of contaminated or impure Pu as a sintered ceramic waste form (SCWF) containing 2-10 wt% PuO2 in a UO2 matrix, similar in form and concentration to either unirradiated or spent MOX fuel. This alternative approach may minimize capital outlays, repository acceptability issues, and implementation duration schedules.

DISPOSITION CRITERIA

Disposition of excess Pu must satisfy three key objectives to achieve overall safety and nonproliferation goals:

1. Rendering the Pu relatively inaccessible for nuclear weapons use

This objective was one of the principal outcomes of the National Academy of Sciences study, (1) which recommended placing the excess Pu in a form that is roughly as inaccessible for weapons use as are the much larger worldwide stockpiles of Pu contained in spent commercial nuclear fuel (SNF). To achieve this degree of inaccessibility requires measures that would make it difficult for a terrorist group, a non-weapons state, or even the United States or Russia themselves to use (or reuse) the Pu in any kind of nuclear explosive device. The National Academy study observed that greater protection than this for excess Pu would be pointless and expensive, and that less protection would be dangerous. This objective is generally interpreted as meeting the "spent fuel standard."

Access to SNF for weapons use is made difficult by the existence of a high radiation-field barrier, by its having Pu isotopic mixtures different than that of weapons-grade Pu, by dilution of the Pu in a ceramic or glass matrix that requires significant chemical and physical processing for recovery, and by containment of the Pu in large, easily accounted and controlled unit item sealed forms. These factors are being considered by the DOE in developing a spent fuel standard to be used in judging the acceptability of any proposed final disposition form before the implementation of any disposition technology approach. (3,4)

2. Conducting disposition operations so that their health and environmental impacts are acceptable

Any processing activity or final disposition material forms that could adversely affect the environment after geologic disposal must meet myriad environmental, health, and safety regulations. Pu processing operations, geologic repository operations, and final disposition forms that could affect worker or public radiation doses or exposures to hazardous chemicals must meet or exceed the regulative requirements.

3. Achieving timely, cost-effective disposition

If a given strategy is to achieve timely, cost-effective disposition of excess Pu, all necessary operations must be technologically viable so that R&D schedule times can be minimized. That viability depends on the existence (or ready construction) of appropriate facilities, equipment, and technologies. Pu disposition options requiring high front-end costs (e.g., for construction of new facilities or processes and for the supporting R&D) will not be regarded as desirable by the current balanced budget-minded Congress. Life-cycle costs (which include operational expenses and decommissioning), which are usually discounted, have less immediate, but still non-negligible, impacts and must also be minimized.

OVERVIEW OF DISPOSITION STRATEGIES

Many disposition strategies were considered in the NAS report, (1) including space launch and sea dilution. Three of these general Pu disposition strategies are under evaluation by the DOE Office of Fissile Materials Disposition (3,4): 1) nuclear burning in a reactor, 2) immobilization with radionuclides in solid matrices to achieve the spent fuel standard, and 3) burial in deep (2-4km) boreholes in ancient rock. The Russians favor nuclear burning as a way of using the energy content of the fissile Pu. Many in the United States, on the other hand, consider the Pu as "waste" to be disposed of by the most expedient and cost-effective method. The draft Preliminary Environmental Impact Statement (3) assumed the use of a single strategy for all excess U.S. Pu inventoriesthat is, for both relatively pure Pu inventories recovered from weapon components and other, less pure Pu in storage or contained in

processing residues or scraps at various U.S. DOE sites. The relative advantages and disadvantages of the three strategies are discussed below.

1. Nuclear Reactor Burning. This approach achieves disposition by burning MOX fuel containing the Pu in a nuclear reactor and then discarding the spent MOX fuel, without reprocessing, in a geologic repository. The major advantages of this approach are its technical maturity (particularly in Europe), its partial favoring by the Russians (who however favor recovery and recycle of the Pu in the spent fuel), and the achievement of the spent fuel standard by the irradiated product form. Both the once-through and recycle approaches reduce the overall weapons Pu inventory and change the plutonium isotopic mixture. The primary disadvantages include possible opposition by opponents of nuclear power, the need to purify impure Pu before MOX fabrication, and reliance on a final geologic repository that is not yet operational. The aqueous technologies used in purification of Pu are expensive and generate significant quantities of secondary low-level and transuranic (TRU) wastes.

2. Immobilization with Radionuclides. This approach achieves the spent fuel standard by incorporating the Pu in a stable, solid matrix containing radionuclides (high-level radioactive defense wastes or 137Cs) whose radiation field, in conjunction with the chemical dilution, large sizes, and safeguards and security, acts as a deterrent to theft or reuse. (5) The immobilized form is eventually discarded in a geological repository in a manner similar to SNF. Immobilization may be a faster, simpler, and cheaper way to achieve the spent fuel standard than reactor burning. The technology is less mature than reactor burning, however, and shares the same concerns over the lack of a geologic repository.

3. Deep Borehole Burial. This approach depends on the depth of the geological isolation, rather than on a radiation "spike" and on physical and chemical characteristics, as in the first two approaches, to achieve the nonproliferation objectives. Its chief advantages are simplicity and perceived permanence, neither of which is offered by the reactor or immobilization approaches until a geologic repository is operated, filled, and sealed. Significant licensing and siting issues must be resolved before implementation, however. (3,4)

DEVELOPMENT OF SINTERED CERAMIC WASTE FORM STRATEGY

None of the three technology approaches just described can alone provide an optimum disposition strategy. We therefore propose a hybrid strategy, not yet evaluated by DOE, that combines the advantages of two of the technologies to achieve cost-effective, timely disposition.

Because both the United States and Russia possess reactors capable of burning excess Pu, we take reactor burning as a main constituent of our hybrid strategy. And because U.S. Pu policy discourages the development of breeder reactor technology, we assume the use of existing thermal reactors in the United States (or possibly Canada). It remains to be ascertained that there is sufficient reactor capacity which can be made available and that this approach is politically and publicly acceptable. Reactor burning alone has the disadvantages already described which includes the need to purify much of the feedstock excess Pu to remove impurities incompatible with reactor fuel specifications. The monetary value of Pu as a reactor fuel in the U. S. is negative, given the current cost of low enriched uranium fuel, so the United States has little incentive to recover impure Pu. We suggest instead that a better approach would be to discard any excess Pu that would require significant purification by fabricating an acceptable geologic repository solid form.

This would reduce the Pu inventories requiring reactor burning and would shorten the completion schedule (by reducing time-consuming reactor operations); it would probably also reduce the number of reactors needed. For this approach to be attractive, however, there must be a low-cost existing technology for discard of the impure Pu not going to the reactor burning option that meets existing environmental laws, anticipated repository requirements, and nonproliferation objectives.

DOE is considering two discard approaches (3,4): deep borehole disposition and immobilization with radionuclides followed by geologic disposition. Deep-borehole disposition would require significant effort to modify existing laws, complete the licensing processes, and characterize a potential site. This preparatory effort would be the same for any borehole disposition strategy, essentially independent of the quantity of Pu to be discarded. New facilities would be required for the borehole operations, increasing up-front costs and possibly introducing implementation time delays. It is unlikely that this approach would be cost- or time-effective for the smaller quantities of impure Pu not destined for burning as MOX in a reactor as part of the proposed hybrid disposition strategy. Thus, deep-borehole disposition does not appear particularly advantageous as part of this strategy.

The immobilization approaches under consideration involve use of a variety of matrix forms for Pu entrainment, including glasses, ceramics, or possibly a glass-bonded zeolite. (3,4) Any waste form intended for repository emplacement must qualify for long-term geological disposal; qualification is a lengthy process and would add to the disposition cost and time for any new waste forms.

Glasses are being developed for disposition of high-level waste from the Defense Waste Processing Facility (DWPF). Since glasses proposed for DWPF high-level waste immobilization were not developed for Pu containment, there are technical problems involving Pu solubility and qualification problems regarding repository acceptability of existing glass waste forms containing high concentrations of fissile materials for Pu disposition. Ceramic or other waste forms under consideration have similar and probably greater concerns.

For a multitude of technical reasons, existing glass immobilization facilities (such as DWPF) cannot be used directly for Pu disposition in glass. In the case of DWPF, for example, criticality considerations would require new melter designs and extensive feed preparation facility modifications and new supporting technology development. Ignoring any issues of delaying the current DWPF mission schedule for HLW vitrification, this would require major front-end investments in research and (eventually) in facility construction and modifications that could exceed the cost and time penalties of simply purifying the impure Pu with current Pu processing technologies for reactor burning.

The acceptability of the Pu disposition waste form within the repository must also be considered. Two waste forms have received significant attention in the United States: DWPF glass and spent commercial nuclear light water reactor (LWR) fuel. DWPF glass, while acceptable for containment of defense high-level waste at the Savannah River Site, is not likely to be directly suitable for Pu containment. This would require the development of alternative glass compositions to serve as the Pu containment systems. Spent LWR fuel is of particular significance in this context, because U.S. policy contemplates direct disposal of such fuel in a geological repository, without reprocessing. In this case, the primary

barriers to Pu release to the geologic environment are the containers, any other engineered barriers used to emplace the spent fuel in the repository, the spent fuel assembly structural materials, and the ceramic UO_2 matrix form encapsulating the PuO_2 and fission products in the spent fuel.

To achieve waste form acceptability for the repository, to avoid new facility construction or new process development, and to control costs and schedule, we suggest a hybrid strategy that produces only one type of material form. The proposed waste form is a UO_2/PuO_2 matrix containing 2-10 wt% PuO_2. Such a matrix is prepared and sintered similar to MOX reactor fuel fabrication and has a chemistry and morphology similar to MOX reactor fuel. Figure 1 shows the proposed strategy, here specifying a light water reactor (LWR) to burn the relatively pure Pu in MOX fuel and using the MOX fabrication technology to fabricate and discard the impure Pu as a sintered ceramic waste form (SCWF). The primary advantages of such a waste form include likely repository acceptability with minimal additional qualification testing for the SCWF and compatibility of SCWF processing and facility requirements with those routinely used in Europe for MOX fuel fabrication. If a permanent discard solution other than a geologic repository for the spent LWR nuclear fuel must be found, that solution would also apply to the discard SCWF waste form proposed here. Disposition of 10 to 20t of impure Pu in any of the SCWF forms suggested would result in a MOX fuel-like component of repository waste that is less than 1% of the projected U.S. SNF inventory (about 80,000t).

Fig. 1

The first step in fabricating such a compatible waste form for the impure Pu would be to make a compressed PuO_2/UO_2 mixture and pellet similar to the starting form for MOX fuel, as indicated in Fig. 1, but without the dimensional tolerances and feed Pu purity requirements applicable to MOX destined for a LWR. Impurities introduced with the impure Pu feed would in most cases probably be contained in the waste form and would likely not decrease repository acceptability any more than the fission products normally contained in spent LWR fuel. Such a compressed pellet form would then be sintered in a reducing atmosphere to increase grain size and intergranular bonding to form a sintered matrix, which would constitute the final Pu containment waste form. This sintered ceramic waste form (SCWF) pellet would then be encapsulated in one of two types of container, as suggested in Fig. 1: 1) relatively small (2 to 4L) cans, which would then be sealed and installed in the glass can-in-canister configuration for addition of a radiation barrier such as DWPF glass containing defense HLW, or 2) relatively large DWPF type canisters (0.6m o.d. - 3m long), to which would be added an encapsulating grout matrix containing ^{137}Cs. A third, hypothetical approach could be to fabricate canisters of SCWF pellets in geometries like fuel assemblies and mix these canisters with irradiated SNF in the multipurpose canister (MPC)-like repository container (6) for final emplacement. Radiation fields calculated for mixing five fuel-like SCWF assemblies with 12 SNF assemblies from pressurized water reactors (PWRs) are on the order of 300 rem/hr at 1m from the centerline 30yr after MPC container loading; this is similar to the radiation fields for the PWR spent fuel assemblies alone.(5) Such fuel-like SCWF assemblies would require a level of protection similar to MOX assemblies until they were mixed with the spent fuel, at which time the essence of the spent fuel standard would be achieved.

Clearly, such an approach has the advantage that the SCWF waste form would be produced with the same technology used for MOX reactor fuel production. If integration of the private and government sectors can be accomplished, it may be possible to utilize the same facilities for both processes, as indicated in Fig. 1. If such integration of the MOX fuel and SCWF fabrication is not economically or politically feasible, duplication of the MOX fuel fabrication technology in the Pu recovery facility for SCWF fabrication would very likely be faster and cheaper than development of a new immobilization technology and processing capability.

FEASIBILITY OF THE SINTERED CERAMIC WASTE FORM (SCWF) IMPLEMENTATION

The feasibility of the approach shown in Fig. 1 was assessed by assuming that the 50t of projected excess Pu feed material consists of 40t of relatively pure Pu suitable for reactor burning and 10t of impure Pu to be discarded in the SCWF. The number of reactors required to burn 40t of relatively clean Pu would depend on the reactor types selected and on the MOX Pu loadings, but should be within the range of available existing reactors. The 10t of impure Pu would be blended and sintered with depleted UO2 to form the SCWF pelletsessentially, "off-spec" unirradiated MOX pellets.

Figure 2 indicates the number of DWPF cans that would be required to hold the 2- to 4-L SCWF cans in the DWPF can-in-canister option, as a function of average Pu loading in the SCWF matrix. Since 5000 to 6000 DWPF canisters are projected for completion of the defense high-level waste immobilization mission, discard of the impure Pu within these canisters, with only a modest increase in the total of DWPF canisters, using the can-in-canister approach is possible with average Pu loadings of 0.5 wt% or higher in the SCWF. This loading is substantially lower than that in new MOX fuel and is comparable to that in normal spent LWR fuel. Higher Pu loadings (up to 5wt%) would substantially reduce the number of 2- to 4-L cans that would have to be handled in the can-in-canister option and would reduce the disposition cost. Plutonium loadings of a few weight per cent should not have a significant repository impact as compared with SNF assemblies from the standpoint of criticality concerns, particularly during the post-closure period. In fact, the use of depleted UO2 as a matrix material virtually eliminates long-term criticality concerns as the 239Pu decays to 235U. If the 239Pu were contained in a non-uranium matrix, the possibility may occur for 235U geological reconcentration as a result of long-term migration and waste form degradation; this could not occur after the 239Pu decays with the use of a depleted UO2 matrix.

Fig. 2

Figure 2 also indicates the number of containers similar to the proposed multipurpose containers (MPCs) (6) that would be required to discard 10t of impure Pu if the SCWF pellet products are mixed with a 137Cs-loaded grout in a DWPF-like container or fabricated as canisters looking like MOX fuel assemblies and loaded five at a time into an MPC-like unit. No more than a few hundred MPC-like containers would be required for Pu loadings of a few weight per cent. This is consistent with the normal Pu loadings in both spent MOX fuel and normal SNF. Even at loadings as low as 0.5wt%, fewer than 1000 MPC-like containers would be required.

CONCLUSIONS

Using the hybrid strategy of existing thermal reactors to burn clean excess Pu and fabricating a sintered ceramic waste form (SCWF), largely

of depleted UO2, for discard of impure excess Pu has the following advantages:

1. The SCWF mimics the chemical and morphological forms of unirradiated MOX LWR fuel and should have similar repository impacts over geological times. The SCWF should be as acceptable for repository disposal as spent MOX LWR fuel, or even spent commercial low enriched uranium LWR fuel.
2. The proposed SCWF pellet fabrication for impure Pu utilizes facilities and technologies virtually identical to those routinely used in Europe to fabricate mixed oxide (MOX) fuel, thus minimizing new required facilities or new technology development and demonstrations.
3. The proposed use of SCWF pellets to dispose of impure excess Pu eliminates the need to purify it, as would be required in any MOX reactor-based disposition strategy.
4. The SCWF provides a direct avenue for disposition of TRU scraps of impure Pu generated during the recovery of Pu from excess pits and for residues from MOX fuel fabrication. This avoids the need for additional processing facilities or operations for recovery and minimizes disposal of these Pu-containing residues.
5. Radiation barriers and the achievement of the spent fuel standard can be readily implemented by 1) using the can-in-canister approach with DWPF as the source of a glass containing a high-level waste radiation barrier, 2) encapsulating the SCWF pellets in a grout matrix containing 137Cs in a DWPF type can in a small hot-cell facility, or 3) by mixing the SCWF pellets, packaged as a fuel-like assembly (or similar size container), with SNF from normal power reactor operations.
6. Use of a single waste form and fabrication process similar to unirradiated MOX for impure excess Pu and for MOX fabrication of the pure Pu for burning in existing reactors would reduce up-front costs for capital construction, research, development, and demonstration.
7. Because existing reactor capacity in the United States (or possibly Canada) would be utilized, and because no new technology development is required, this strategy would minimize the time required to achieve the spent fuel standard for excess Pu.
8. Adoption of the reactor burning approach may assist in developing effective linkages with reactor disposition approaches for pure excess Pu currently favored by the Russians.
9. MOX fuel fabrication and reactor burning can be time phased with the production of the SCWF fabrication to best utilize facility capacity and to optimize schedules. Multiple-line MOX plants could be considered for simultaneous processing and fabrication of pure and impure excess Pu forms.

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CONTRIBUTIONS OF THE COMMERCIAL FUEL CYCLE INDUSTRY TO THE US WEAPONS PLUTONIUM DISPOSITION PROGRAM

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ABSTRACT

The manufacture and burning of Mixed Oxide (MOX) fuel is a technically mature industry in Europe, having developed as a natural progression of the closed fuel cycle infrastructure in the United Kingdom, France and elsewhere in Europe. This paper discusses the relevance of the European MOX fuel experience to the US excess weapons plutonium program, identifies unresolved technical issues, and presents a step-wise progression towards utilization of MOX fuel in North American reactors, utilizing the capabilities and expertise of the European commercial fuel cycle industry.

BACKGROUND

The end of the Cold War has dramatically altered the course of the Department of Energy (DOE) from one of nuclear materials production to that of nuclear materials disposition. The National Academy of Sciences has conducted detailed studies on the preferred approaches to the disposition of excess weapons plutonium and concluded that concurrent programs on (1) reactor use as Mixed Oxide (MOX) fuel and (2) immobilization in vitrified High-Level Waste (HLW) were worthy of pursuit as the best, most timely approaches to plutonium disposition.

The DOE Office of Fissile Materials Disposition is studying these two options, among others, as the subject of a Programmatic Environmental Impact Statement (PEIS), which will serve as the vehicle for narrowing the options down to one or more finalists, expected to be announced in the fall of 1996.

Meanwhile, DOE has enlisted the help of its laboratories, as well as the private sector, in determining the path forward for deployment of a MOX use program, should the decision be made to proceed in this direction. The DOE program for MOX use assumes that consumption rate can be maximized by using full-core loads. This will necessitate the introduction of burnable poisons in the fuel to control reactivity (in contrast, the commercial MOX fuel experience is based on partial core loadings of MOX fuel without any added burnable poisons).

Lawrence Livermore and Los Alamos Labs are developing non-aqueous chemical processes to convert the plutonium metal "pits" to an oxide form suitable as feed material for a MOX plant; Los Alamos is the "lead lab" for MOX fuel manufacturing facilities and technology; and Oak Ridge is

the "lead lab" for MOX use in North American reactors. These laboratories are reaching out to the commercial fuel cycle industry for advice, experience, and expertise to insure that a MOX program, if chosen, will succeed. Additionally, Oak Ridge and DOE-Headquarters are in continual dialogue with utilities, reactor vendors, and fuel vendors to determine the nature and extent of domestic interest in a MOX use program in North America.

EUROPEAN MOX EXPERIENCE

While it is fair to acknowledge that the US once had a MOX fuel cycle program underway (including six reactors that burned MOX fuel in the U.S.), it is likewise fair to say that the program experienced a premature death in the 1970's when reprocessing, and therefore MOX use, was abandoned in the US as a matter of policy. Meanwhile, the European program continued to evolve to the point that, today, that experience provides a wealth of knowledge directly applicable to the US program - - and, for that matter, the Former Soviet Union program - - to dispose of their excess weapons plutonium inventories.

MOX Fuel Production and Irradiation

Belgonucleaire, Siemens, Cogema, and BNFL have provided MOX fuel as far back as 1963 for irradiation in research reactors and commercial nuclear power plants in Switzerland, Germany, France, Italy, Belgium and Japan. Belgonucleaire and Siemens pioneered the MOX manufacturing business, having produced over 1000 MOX fuel assemblies over the last three decades in their respective facilities sized at 30-35 tons/yr production capacity. Cogema and BNFL will soon eclipse the Belgonucleaire and Siemens production capabilities with their respective, large-scale plants (120MT/yr each) that are presently either under construction or coming on-line to provide MOX fuel to their customers worldwide.

While specific fuel performance data on MOX fuel is generally not available in the open literature, in general, the performance has been excellent, with burnups reaching as high as 30-40 GWD/MT, and good response to reactor transients. No data is available on severe fuel damage behavior, but there is little indication that this would much different than the behavior of urania fuel.

MOX Fuel Manufacturing Technology

The basic technology for manufacture of MOX fuel involves co-milling of the uranium and plutonium oxides to obtain a homogeneous mixture, pellet pressing (with or without binders), pellet sintering, pellet finishing, and loading of the pellets into pins for fuel assembly.

Recent advancements in the MOX manufacturing process have resulted in streamlined production and improved MOX fuel performance. The French and Belgians have developed a process called MIMAS (Micronized Master Blend) in which a "master blend" mixture of 20-30% plutonium is premilled (micronized) and then blended with uranium oxide to form the final mixture (nominally 4-6% Pu). Use of this "master blend" significantly reduces the milling time required in the process. A similar process called "Optimized Co-Milling" (OCOM) has been developed by Siemens. BNFL's advanced process is referred to as the "Short Binderless Route" (SBR), which utilizes a high-energy Attritor Mill to blend the uranium oxide and plutonium oxide feed materials
within a relatively short cycle time, to achieve extremely good homogeneity. Following the mixing step, the powder is agglomerated in a spheroidiser which allows the mixture to flow freely to the pelleting press.

All the advanced MOX fuel manufacturing processes offer significant improvements over traditional co-milling in terms of fuel homogeneity, fission gas retention, and high burn-up.

THE POTENTIAL FOR DISPOSING OF US WEAPONS PLUTONIUM AS MOX

The Technical "Show-Stoppers"

Assuming MOX is chosen as a disposal route for US excess weapons plutonium, there are several technical uncertainties that have been identified by the various technical experts involved in the program. None of these technical uncertainties are true "show stoppers"; nevertheless, some demonstration work will be required to deal with these issues, which are summarized as follows:

1. Acceptability of the feed PuO_2 material for use in a commercial MOX fabrication process. LLNL and LASL, who are developing the pit conversion processes, will need to interface with the commercial MOX fuel fabricators to determine the suitability of their feed material in terms of morphology and other specifications.
2. Presence and behavior of gallium. Gallium is used as an alloying agent in the manufacture of plutonium pits. Studies need to be conducted to determine the behavior of the gallium during the pit conversion process and then during the pellet sintering process. If the gallium escapes during either of these processes (as is expected based on preliminary findings), then this needs to be taken into consideration in terms of the impact on the processing design and performance. If the gallium remains in the MOX pellets, then a determination needs to be made as to its effect on fuel performance.
3. Use of burnable poisons. At core loadings above 50% MOX fuel, the excess reactivity of the core needs to be controlled by the introduction of burnable poisons (gadolinium, erbium, dysprosium, ZrB_2, etc.). Since the US MOX program assumes 100% core loadings to maximize plutonium consumption, burnable poisons will need to be incorporated into the fuel design. No data exists on MOX doped with burnable poisons, though there is some experience with the use of burnable poisons in urania fuel which may be directly applicable. Further demonstration work on the fabrication and irradiation of MOX fuel with burnable poisons is thus a prerequisite for licensing of the fuel.
4. Applicability of the existing database (from European MOX fuel experience) to the US program. There are of course differences in the isotopic mix of weapons-grade plutonium as compared to commercial-grade plutonium. These differences stem from the higher proportion of the heavier isotopes of plutonium present in the commercial-grade material, which results in higher radiation exposure. Dose to operators from commercial-grade plutonium has been controlled in the design of the plants that utilize commercial-grade material (remote handling and additional shielding). Thus, existing experience in handling "dirtier", commercial-grade plutonium represents the "worst case" envelope within which the "cleaner", weapons-grade material will fall. No surprises are expected in dealing with the purer, weapons-grade material. The major difference will be in the criticality design aspects of the MOX fuel manufacturing plant, which are easily accommodated.

Jump-Starting the US Program for MOX Utilization

A driver in the US Excess Weapons Plutonium Disposition Program will be the rate at which progress is made in disposing of this material. Thus, a program that is cluttered with endless R&D and hampered by licensing uncertainties is Dead-On-Arrival. For this program to actually succeed,

early demonstration and rapid deployment are the keys to economic, technical, and programmatic success.

Thus, the program should be designed to begin with near-term steps that can be achieved with technical certainty, while other unknowns are gradually and systematically addressed for full deployment of the program. Such a step-wise program could be jump-started by utilizing existing facilities, expertise, and experience base within the European MOX community. One example of such a step-wise program is given below, though other permutations are certainly possible.

Step 1: Fabricate MOX fuel pellets utilizing pure and Gallium-contaminated plutonium feed material from the US Excess Weapons Plutonium inventory. Pellet fabrication could be undertaken at a number of existing MOX fabrication "demonstration" labs in Europe, such as the BNFL MOX Demonstration Facility at Sellafield.

The importance of using commercial fabrication technologies cannot be overemphasized. Any pellet fabrication tests that do NOT use commercial technology are worthless in the eyes of the NRC for licensing purposes and would have to be repeated. The information gained from these simple pellet fabrication tests would demonstrate (a) the acceptability of the oxide feed material for the pellet fabrication process, and (b) the behavior of the gallium during the pellet manufacturing process. These tests could be performed during 1996, i.e., pre-Record-of-Decision (ROD) on the Programmatic Environmental Impact Statement, thereby advancing the program by an entire year compared to waiting until post-ROD.

Step 2: Manufacture MOX Demonstration Assemblies (MDA's), four assemblies per reactor, for irradiation in candidate North American reactors selected for MOX use. These MDA's can be manufactured at existing demonstration-scale facilities in Europe, thereby utilizing known, commercial manufacturing processes. For BNFL, capacity could be made available for such use at BNF's Sellafield site before the turn of the century. For simplicity, these MDA's would not contain burnable poisons, so as not to complicate the NRC licensing and certification processes. Once approved, core loads of 20-50% MOX could proceed immediately, and meanwhile, test pin irradiations and MDA's for full-core loads of MOX containing burnable poisons could proceed on a separate and slower track. If, for whatever reason, the burnable poisons present unsurmountable certification and licensing problems, then at least the program can proceed using the partial core loads of MOX without poisons. Burn rate can be achieved by utilizing more reactors in the disposition program.

It is assumed here that existing data from European test reactor irradiation trials of commercial MOX will suffice for these early US MDA's and that no separate irradiation trials of MOX derived from this weapons plutonium material will be required.

Step 3: Begin full-scale production of MOX assemblies utilizing any available excess production capacity in Europe while a new plant is designed and constructed on US soil. Utilizing available excess capacity in Europe will bridge the gap between initiation

of a new, US-based MOX plant project and the completion and licensing of that project, which could take as long as 8 years. Utilizing available European production capacity dictates that any US fuel manufactured would be poison-free, so as not to contaminate the otherwise "clean" European facilities. Likewise, utilizing this available excess capacity might mean that the gallium would have to be removed as well, depending on the effect that gallium would have on the operation of these facilities.

Transportation aspects of the European option would have to be taken into consideration. Based on US and European experience in moving both plutonium and MOX fuel, this should not be an unsurmountable obstacle to the use of European facilities in the early stages of the US MOX program.

Step 4: A full-scale US program on MOX utilization would be carried out as dictated by the results of the earlier steps.

ASSISTING THE RUSSIAN PLUTONIUM DISPOSITION PROGRAM

European fuel fabricators are actively involved in providing technical assistance to the Former Soviet Union for the timely disposition of their inventory of excess weapons plutonium. In addition, the European fuel fabricators have been in dialogue their governments to develop multi-lateral mechanisms to assist the Former Soviet Union in implementation of an appropriate plutonium disposition strategy.

Political Will and Political Winds

The single most threatening show stopper to this program, in BNFL's opinion, is the US political will and stamina to get on with the program! DOE's track record for completing any large undertaking over the last 20 years is miserable at best. Yet it is heartening to observe in this instance a keen desire on the part of the DOE labs and the DOE headquarters program managers to go beyond the "Plutonium Disposition" study phase into an action-oriented phase involving the capabilities of the private sector.

Political winds tend to shift every election cycle, providing little assurance of continuity of any long-term commitments to a long-term plutonium disposition program. Thus, the best means of eliminating the political uncertainty is to place the MOX program firmly in the hands of the private sector as much and as soon as possible.

A report prepared by Oak Ridge National Laboratory that surveyed world-wide MOX experience emphasizes the significance of the European knowledge base that will enable the US to leapfrog over an expensive and time-consuming R&D phase and go directly to a demonstration and implementation phase. Hopefully, the ability to leapfrog ahead into a commercially driven program will be just the driving force that is needed to overcome the potentially detrimental effects that politics would otherwise play in this important program.

CONCLUSIONS

A proven, mature MOX industry exists to assist the US in the disposition of excess weapons plutonium. That industry is alive and well in Europe, and willing to provide expertise and capabilities to facilitate early deployment of a US MOX program.

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Session 05 -- DIRECTIONS IN REGULATORY DEVELOPMENT AND RISK ASSESSMENT
Co-chair: D.E. Wood, GaeaTech Services
5-1

RISK-BASED DECISION MAKING IN THE DOE: CHALLENGES AND STATUS

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ABSTRACT
The primary responsibility of the Environmental Management program is to address the most immediate, urgent risks to human health and the environment from the nuclear weapons complex, while managing long-term contamination and safety threats. Environmental Management cannot meet this responsibility unless it integrates the best risk management practices into its decision making processes. In response to direction and guidance from Congress, the National Academy of Sciences, and other organizations, the Environmental Management program has implemented a risk-based approach to decision making, successfully using a qualitative evaluation process to inform decisions in the fiscal year 1996 and 1997 budget cycles. Risk information will be used even more rigorously and more in-depth for the fiscal year 1998 budget development cycle and beyond. The process of using risk information to establish priorities will improve as the Department improves data quality, incorporates peer review, defines the future of its sites, and keeps its stakeholders fully informed and involved.

INTRODUCTION

In the United States, the nuclear arms race resulted in the development of a vast research, production, and testing network that became known as the "nuclear weapons complex." The complex consisted of 2.3 million acres of land and 120 million square feet of buildings and ranged in diversity from a vast tract of land in the deserts of Nevada to warehouses in downtown New York City that once stored uranium.

The Department of Energy created the Office of Environmental Management in 1989 to manage the legacy of fifty years of nuclear weapon production and research at 137 sites in more than thirty states and territories. The Environmental Management program's responsibility is to address the most immediate, urgent risks to human health and the environment, while managing long-term contamination and safety threats. The program also manages wastes currently being produced during nuclear energy research and development, basic science research, and ongoing missions.

The Environmental Management program is currently facing a decreasing budget while still having to deal with competing requirements and risks to workers, public, and environment. The Department recognizes that credible risk assessment and the best risk management tools are needed to meet its primary mission of protecting human health and the environment. The decisions involved in managing these problems include long-term environmental and public safety concerns, national security issues such as nuclear proliferation, and federal budget limitations. The future course of the Environmental Management program will depend on several fundamental technical and policy choices, many of which we have not yet made.

THE CHALLENGES OF USING A RISK-BASED DECISION MAKING PROCESS

Many concerns have been raised regarding risk assessment: our ability to define what the risks are on a site-by-site basis and in a systematic way; the fact that it matters "who" performs the risk assessments; and the many methodology issues about identifying and assessing risks as well as uncertainty, data gaps, and concerns over the quality of information. Knowing these controversies surrounding risk and the use a risk-based approach for environmental management, two years ago the Department asked the National Academy of Sciences to decide whether and how risk and risk-based decisions could be incorporated into the Environmental Management program. The study resulted in the January 1994 report Building Consensus through Risk Assessment and Management of the Department of Energy's Environmental Remediation Program. In the report, the Academy identified the major obstacles, issues and barriers to implementing a risk-based management approach. The report concluded that the use of risk-based approach could help compare outcomes, build consensus, and gain early public involvement to include cultural, socioeconomic, historical, and religious values, if its purposes and limitations are well-defined.

The Academy also noted that to be effective and useful, the procedures and institutions adopted for risk assessment satisfy several objectives:

They must be credible to stakeholders and the general public.

They must operate expeditiously without threatening scientific validity.

They should consider the full range of risks of concern to the stakeholders in the light of social, religious, historical and political values, future land uses, and cultural values and needs.

They should be efficient and cost effective and produce results that contribute to identification of remedies and priorities that are themselves efficient and cost effective.

Based on the findings of the report from the National Academy of Sciences, that risk-based decision-making was both feasible and desirable for the Environmental Management program, the Department of Energy adopted a set of principles for using risk analysis. The principles (developed by an interagency working group) are designed to be a first cut at defining risk analysis, its purposes, and the principles to be followed by the Department of Energy if it is to be done well and credibly. These principles include four major categories:

Risk Assessment. Use the best available information from all sources; all judgements and assumptions should be explicitly stated.

Risk Management. Analyze the distribution of risk and costs/benefit of potential risk management strategies, using the best available tools and techniques.

Risk Communication. State risk management goals, assumptions, uncertainties and comparisons clearly, accurately, and meaningfully; provide public access in a timely manner.

Priority-Setting. Compare risks by grouping them into broad categories of concern (e.g., high, medium, low) and identifying the population at risk; include as broad a range of views as possible, ideally with consensus.

"THE FIRST STEP" TO LINKING RISK WITH THE BUDGET

Since its formation six years ago, the Environmental Management Program has been beset by public and Congressional concerns over priorities and the pace of cleanup versus total program costs. The Department of Energy's Office of Environmental Management's overall budget grew from approximately $2.3 billion in 1990 to approximately $6.5 billion in 1994. Concerned about this rapid budget increase, yet sensitive to the public concerns about the risks posed by the department's sites, the Congress urged the Department to begin to develop a risk-based approach for sequencing or prioritizing its activities. Specifically, the Conference Report of the Energy and Water Development Appropriations Subcommittee for Fiscal Year 1994, said that the Department "...needs to develop a mechanism for establishing priorities among competing cleanup requirements." Furthermore, the Department was directed to "submit by June 30, 1995 a report...evaluating the risk to public health and safety posed by the conditions at weapons complex facilities that are addressed by compliance agreement requirements." The committee emphasized that it did "not intend [for] the Department to perform an exhaustive formal risk assessment, as that term is frequently used, of the thousands of cleanup activities required by compliance agreements. Instead, the Department [was] directed to estimate the risk addressed by cleanup requirements on the basis of the best scientific evidence available."

In response to the Congressional request, the Department initiated a major effort to define its risks on a site-by-site basis systematically. In reevaluating the Environmental Management program in 1994, the Department announced its intent to establish more credible and consistent methods of conducting risk assessments at its sites and facilities.

In the current climate of decreasing budgets, the public is especially concerned about ensuring that funds are being spent wisely and cost effectively. There are demands to achieve the highest level of risk reduction possible for every taxpayer dollar spent. This can only be achieved by fully understanding all the risks that the program must address, understanding the costs associated with addressing those risks, and making decisions based upon that comprehensive understanding.

Information about risks is generally collected and analyzed at a specific facility or site, or for a particular contaminant or hazard. Such information has not been available for decision making, or for establishing priorities. Many risk-related reports are completed based on requirements specific to a regulation or a compliance agreement and do not allow for comparisons of risks, or for an integrated complex-wide analysis of risk. A primary objective of the draft report entitled Risks and the Risk Debate: Searching for Common Ground "The First Step" (from now on called the "Draft Risk Report"), submitted to Congress in June 1995, was to develop a process that provides an integrated approach to evaluating the risks to human health, worker safety, and the environment posed by conditions at the Department's sites and facilities, and links those risks to compliance requirements and the budget.

The Environmental Management program's Office of Integrated Risk Management adopted a qualitative evaluation matrix for the Draft Risk Report. The process was not designed to replace existing site-specific approaches, but to increase the understanding of all Environmental Management activities, particularly as related to risks to public health, worker health, the environment, compliance requirements, and budget allocations across the EM programs. Department of Energy Field program managers with expertise about these activities at their site categorized the activities in six areas. The intent of the qualitative approach was to describe all Environmental Management activities to develop a consistent, Environmental Management-wide framework for capturing and communicating the information from all Environmental Management sites and facilities.

CURRENT STATUS

The Draft Risk Report to Congress provides the first link between budget, compliance requirements, and risk reduction/pollution prevention activities. The process used for the report provides an initial framework to capture the spectrum of risks (public, health, worker, and environment) associated with planned Environmental Management activities and to link these risks to compliance requirements and the budget qualitatively.

The information provided a baseline from which both DOE and its stakeholders can use to engage in dialogue about the risks and costs associated with the various Environmental Management activities at each site. This baseline information was successfully used in the fiscal year 1996 and 1997 budget processes, as one tool in the decision making process that determined how Environmental Management would allocate its funding, establish priorities, and sequence its work.

The Environmental Management Advisory Board, an advisory group charted under the Federal Advisory Committee Act, was asked to review the Draft Risk Report and the qualitative evaluation process used to develop information linking risk, compliance, and budget for all Environmental Management activities. The Board recognized the process used to develop the Draft Risk Report as an important first step in linking both compliance and budget information. They endorsed the use of the process and endorsed the recommendations made to improve the data quality and assure consistent application and interpretation of those data. Their recommendations to the Department were to:

Improve Data Credibility and Quality by clearly identifying and communicating the assumptions, implementing public and peer review, and using corporate training and workshops. In addition, the categorization of activities needs to be consistent across sites, clear, recognizable and meaningful.

Implement a "three-tiered approach" to improve the risk assessment/management process further, in developing guidance, in conducting qualitative evaluations, and in evaluating the results and process.

Improve stakeholder involvement through a variety of creative approaches to engage stakeholders in the qualitative evaluation process meaningfully. Better information and communication tools are needed for this complex subject if the program is to be effective. Future land use and land use assumptions are critical components of this process, and each site needs specific future land use criteria.

Pay close attention to "timing" issues. For example, a low risk activity can result in much higher risk later if wastes are unstable or leaking. In addition, effects can occur due to the performance of an activity, and should be considered in the decision making process.

Fully integrate the entire process with the budget, long-term cost projections, future land use planning, and stakeholder involvement.

As Environmental Management goes forward to use this process and framework in the fiscal year 1998 budget and planning process, it is carrying out these recommendations. The information provided within the process will change as policy decisions such as land use options are determined for each site, as new regulations and compliance agreements are made, as risk information improves, and as more data become available. The Department believes that having this framework consistently implemented and understood will take more than one year, but that the lessons learned each year will be incorporated into the framework. The Department will continue to work on enhancing risk evaluation methods throughout the year. To enhance a consistent approach that captures the spectrum of risks associated with Environmental Management's programs across the weapons complex, Headquarters has been working with the Field Offices to develop detailed guidance for use in developing the fiscal year 1998 Environmental Management budget. Personnel representing all EM sites and programs and external peer review experts from outside the Department of Energy will conduct evaluations of risk data and process quality at the EM-wide level to improve the risk assessment/management process. Peer review will be accomplished through the following "three tiered approach":

The first tier consists of a central group of experts, stakeholders and regulators to develop the guidance for the comparative risk assessment process. (Complete for the Fiscal Year 1998 budget process.)

The second tier consists of risk assessment professionals, environmental experts, former Department of Energy employees, and Department of Energy field office representatives to conduct the comparative risk assessment. This method will ensure cross-site input, thereby reducing bias, promoting consistency and building credibility for the process. This review will take place on a local and national level. The local review processes will be defined and managed by Field Offices and will take place in the November to April time frame. The national review process will be coordinated by Headquarters budget and risk organizations and is taking place in the January to May 1996 time frame.

The third tier will consist of a group to evaluate the process. This group will consist of some of those in the first and second tiers, and other independent experts. The review group will be responsible for reviewing the process and providing guidance on carrying it out. This group will perform its review in the April through July 1996 period.

In addition to peer review, the Department recognizes that stakeholder involvement is important to both the quality of information and the credibility and validity of the decision making process. The Environmental Management program is using a variety of creative approaches to engage stakeholders in the risk and budget issues meaningfully so that stakeholders have access to accurate, understandable, and timely information. The goal of these activities is to allow stakeholders sufficient time to be able to assimilate the information and an opportunity to be heard during the decision making process.

In collecting information to inform its decision making processes, the Department has begun to account for the various attitudes, interests, and community activities that could be affected or disrupted by performing or not performing an activity. The Department will collect information regarding the potentially negative social, cultural, and economic impacts of activities and the environmental situations they address (e.g., temporary or permanent cessation of important community activities, disruption of traditions or ceremonies practiced by specific populations or groups). Such information will be considered with the other information used to inform priority setting.

Improved stakeholder involvement is dependent on the ability of the Environmental Management program to communicate risk information effectively and to identify meaningful and timely opportunities for public participation. It is vital that EM be clear about the nature of opportunities for public input on environmental decision-making, and be clear about how the public input will be used. For example, the Department is helping stakeholders to participate in the 1998 budget process by preparing a guide entitled Public Participation in the Fiscal Year 1998 Office of Environmental Management Budget. This document identifies the decisions to be made, the key documents being provided to decision-makers, and the opportunities for stakeholder involvement.

Assumptions regarding the future use of land at the Department's sites directly affect estimates of risks to the public, to workers, and to the environment. To conduct meaningful risk assessments and select appropriate remedial methods, sites must first define future land uses with significant public input. To date, several sites have already formulated stakeholder-preferred land uses as a result of the Secretary's initiative to develop site land use plans; these land use assumptions will serve as a basic input into site risk assessments.

Additionally, Environmental Management is supporting several cooperative agreements with independent institutions to work together helping the Department further define and implement approaches to credible risk assessment and risk management practices. These include: development of comprehensive, innovative methods to assess and communicate ecological and human health risks; research programs to foster a greater understanding of the ecological and human health effects associated with remediation activities at specific sites; and development of information for ecological risk analysis. Specific activities will include: working with Field elements and their local stakeholders to improve risk communication between these groups; providing peer review and analysis of Field generated risk information; involving public participation in risk evaluation; and developing tools for communicating risks to nontechnical audiences.

CONCLUSIONS

As the Office of Environmental Management attempts to accomplish its primary mission, protecting human health and the environment, while facing a decreasing budget and competing regulatory and stakeholder requirements, it is developing and using a framework that builds on all of the risk information collected on a site and facility basis and allows for a consistent approach to evaluating risks throughout the nuclear weapons complex.

Clearly identifying and communicating the assumptions used in developing evaluation data will improve the credibility and quality of the data, as will implementing public and peer review and using training and workshops

across the Environmental Management program. Across sites, categorization of activities will be made consistent, clear, recognizable, and meaningful.

The "three-tiered approach" to peer review being implemented will improve risk assessment and risk management processes in: developing guidance, conducting qualitative evaluations, and evaluating the results and process.

Values of stakeholders, including Native Americans and other groups must be considered and solicited actively, meaningfully, and effectively. Sites should clearly explain the basis for their land use assumptions, particularly to identify the degree to which area citizens, tribal governments, regulatory agencies, and other interested parties accept these assumptions. Cleanup decisions must be tailored to accomplish the future land use plans.

Environmental Management successfully used a qualitative evaluation process to inform decisions in the fiscal year 1996 and 1997 budget cycles. Risk information will be used even more rigorously and more in-depth for the fiscal year 1998 budget development cycle and beyond. The process of informing priorities with risk information will improve as the Department improves data quality, incorporates peer review, defines the future of its sites, and keeps its stakeholders fully informed and involved.

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5-2

IMPROVING REGULATION OF USDOE NUCLEAR SAFETY

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Federal Advisory Committee on External Regulation of DOE Nuclear Safety

ABSTRACT

The Federal Advisory Committee on External Regulation of U.S. Department of Energy Nuclear Safety recently issued its Final Report, in which it unanimously recommended full external regulation of facility safety, worker protection, and environmental protection of essentially all DOE nuclear facilities. A great majority of the Committee recommended that the regulators be existing agencies, but the Committee did not choose between the U.S. Nuclear Regulatory Commission and the Defense Facilities Nuclear Safety Board, concluding that either one would have to change significantly before it could carry out the responsibilities of a regulator of facility safety at the DOE nuclear complex. The Committee believes that external regulation will increase the assurance of safety at DOE, help restore the credibility of the Department's safety efforts, and increase the efficiency of its operations. The Committee made a number of other recommendations designed to bring about streamlined, effective, and efficient external regulation of safety at DOE nuclear facilities. The recommendations cover such issues as State roles in regulating DOE, public participation, the jurisdictions of the external regulators, and the use of a variety of ways to regulate. The Committee also recommended that DOE maintain a strong "corporate" safety function, to provide assurance that the Department will conform to regulatory requirements, and that any transition to external regulation be managed carefully, so that commendable efforts the Department has made on behalf of nuclear safety are not disrupted. The Department has formed an internal group to evaluate the Committee's recommendations.

INTRODUCTION

The aim of this paper is to give readers a brief overview of the work of the Advisory Committee on External Regulation of Department of Energy Nuclear Safety. The Committee was established in January 1995 by Secretary of Energy Hazel O'Leary. Its Charter assigned it the task of advising the Department, the Office of Management and Budget (OMB), and the White House Council on Environmental Quality (CEQ) on whether, and if so, how, the Department's nuclear facilities should be externally regulated. On January 19 of this year, the Committee's recommendations were made public. The fundamental recommendations were that essentially all DOE nuclear activities be externally regulated by existing agencies, and that DOE retain a strong internal safety management system.

HOW THE COMMITTEE CAME TO BE

The Committee grows out of a long history of concern with the substantial lack of external regulation of nuclear safety at DOE and its predecessors. From the "Manhattan Project" on, nuclear safety at the Department and, before that, the Energy Research and Development Administration and the Atomic Energy Commission has been largely a matter of self-regulation. In the beginning, it could have been no other way, because almost all the expertise was in the Atomic Energy Commission and its contractors. Various federal statutes have long exempted the Department and its predecessors from much of the external regulation to which other federal nuclear facilities and private nuclear facilities are subject. Only since the late 1970's has the DOE nuclear complex been subject to environmental regulation by the Environmental Protection Agency (EPA) and the States, and even now some environmental statutes exempt certain nuclear materials at DOE from external regulation. Only since 1989, when the Defense Nuclear Facilities Safety Board (DNFSB) was

established by Federal statute, has there been any defense complex-wide degree of external oversight of what the Committee calls "facility safety" at nuclear facilities, namely design, construction, operation, modification, and decommissioning. However, the Defense Board investigates and recommends; it does not write or enforce safety rules. (The Nuclear Regulatory Commission (NRC) has had a regulatory role in facility safety at a limited number of DOE sites.) The Occupational Safety and Health Administration (OSHA) has no regulatory role over protection of workers at DOE nuclear facilities, though DOE and OSHA are now studying the possibility of having OSHA regulate DOE.

DOE has a great many nuclear facilities to self-regulate. It manages some 3500 nuclear facilities at 34 individual sites in 13 states. Some of these sites cover hundreds of thousands of acres. The 3500 facilities include uranium separation facilities, chemical processing and reactor fuel fabrication facilities, nuclear production and research reactors, fuel reprocessing facilities, nuclear weapons assembly and disassembly facilities, tritium recovery facilities, nuclear materials storage vaults, some of the world's largest high energy particle accelerators, a fusion laboratory, and a variety of large and small defense and non-defense research laboratories. DOE also manages very large quantities of radioactive materials, including more than 430 metric tons of fissile and non-fissile radioactive materials and isotopes and more than 2,700 metric tons of spent reactor fuel. There are also contaminated inventories of, among other things, roughly 800 million cubic meters of ground water and 60 million cubic meters of contaminated soil.

Self-regulation of such a large and variegated nuclear complex has long been the subject of criticism by outside persons and organizations. An excellent summary of these criticisms was compiled by the Congressional Research Service of the Library of Congress. Two of the criticisms are fundamental: 1) No organization should be permitted to practice "self-regulation" because this inevitably places safety in conflict with primary mission objectives, and 2) self-regulation is particularly ineffective when conducted under a cloak of secrecy. Since the 1980s, a number of Secretaries of Energy, including John Herrington, James Watkins, and now Hazel O'Leary, have worked hard to improve DOE's internal management of safety at nuclear facilities, but the basic issues associated with self-regulation -- particularly the conflict of interest inherent in self-regulation -- have remained.

In early 1994, the House Subcommittee on Energy and Mineral Resources held hearings on proposed legislation to establish external regulation of new DOE nuclear facilities and explore the options for external regulation of existing facilities. At those hearings, Charles Curtis, then Under Secretary of DOE, proposed that DOE initiate a Federal examination of whether and how the Department's nuclear facilities should be externally regulated. In January 1995, DOE established the Committee on External Regulation to conduct that examination. At Secretary O'Leary's urging, the Committee took a broad view of safety, deciding that it should look at all aspects of safety at DOE nuclear facilities: facility safety, worker protection, and environmental protection. The Committee believed that safety requires a coordinated approach encompassing not only radiological protection, but also protection against all hazards at DOE facilities, especially as the Department's emphasis shifts from weapons production to cleanup and waste management,

and chemical and physical hazards to workers come into greater prominence.

THE COMMITTEE'S COMPOSITION AND WORKING METHODS

The membership of the Committee was extraordinarily diverse and very capable. Represented among the 24 members were Federal and State regulatory agencies, national and local citizens' and environmental groups, the nuclear power industry, Native Americans, and labor unions. Among the members were the head of OSHA, the Lt. Governor of Colorado, the acting Provost of the University of California System, and the Chairman of the Chemehuevi Tribe in California. The two Co-Chairs of the Committee were John Ahearne, a former Chairman of the NRC, and Gerard Scannell, a former head of OSHA.

The Committee's work was supported by a small staff of persons drawn from DOE, EPA, and NRC, and by a DOE Internal Working Group formed of persons from most DOE offices. These support efforts were managed by Thomas Isaacs, the Committee's Executive Director.

The Committee conducted its deliberations completely in the open. It held eight two-day meetings at sites across the country, from Washington, D.C. to the Hanford site in the State of Washington. These meetings generated about 4500 pages of transcript, 400 of which were comments from members of the general public. Before each meeting, a team of Committee staff members met with citizens and organizations near the site to encourage their participation. The meetings were advertised in local newspapers and over the DOE system. The Committee set up an 800 number and a World Wide Web site to distribute information about the Committee's activities. The Committee also asked each DOE program office, operations office, and laboratory to tell the Committee what that office or lab thought the major regulatory problems were, and what the solutions might be.

Even before the Committee's first meeting, the Department's Internal Working Group completed several major reports for the Committee, including a compendium of all DOE nuclear facilities, a history of nuclear safety at the Department, an account of the safety laws applicable to DOE nuclear facilities, a description of DOE's many laudable initiatives to improve health and safety at its nuclear facilities, and an account of how nuclear safety at comparable facilities in other countries is regulated.

The Committee itself formed active subcommittees on five subjects: problems with the current system of regulating DOE, what external regulation can and cannot accomplish, worker safety, national security, and options for regulatory models and the distribution of regulatory responsibility. At the Committee's request, the staff wrote papers on several subjects, including enforcement against federal agencies, regulatory models and options, criteria for judging options, contract reform, the role of the public in regulation, and transition issues.

The reports by the Committee, the subcommittees, the staff, and the Internal Working Group are all available from several sources, including the World Wide Web (as of February 2, there had been about 28,000 accesses to the Committee's Web material). The transcripts of the meetings are also available on the Web.

PROBLEMS AND THEIR CAUSES

The Committee found these symptoms of problems in the way safety at DOE nuclear facilities is now being regulated: frequently poor safety practices, which have resulted in costly cleanups and unnecessary exposures; distrust of DOE, in the public, in Congress, and among other

governmental agencies; frustration throughout DOE's work force with a regulatory morass that often impedes efforts to do work both safely and well; numerous cases of non-compliance with established safety requirements; and failure to address hazards by their relative importance.

In the Committee's view, these symptoms have fundamental causes rooted in DOE's approach over the years to safety, and in its regulatory structure. Although the Department has made substantial progress in recent years, the Committee concluded that most of the root causes of the problems remain. In the Committee's view, these causes are: the built-in conflict of interest in self-regulation; the legacy of secrecy that has historically shielded DOE's activities from outside view, thus leading in many cases to safety practices less sound than those used in the private sector; lack of stability in safety management and policy, stemming from the fact that DOE's leaders are of necessity political appointees whose tenures are short; lack of management accountability and inadequately coordinated regulatory and oversight functions; redundant, confusing safety requirements (the Department is working hard to straighten this out); and a lack of balance in addressing hazards, with chemical and physical hazards to workers sometimes receiving insufficient attention, and hazards covered by enforceable sanctions receiving more attention than risk and cost/benefit considerations can justify.

WHY EXTERNAL REGULATION, AND HOW?

The Committee concluded that "it is sound public policy to ensure through external regulation that DOE is regulated to standards and other requirements comparable to those to which the private sector is regulated." More particularly, external regulation can eliminate the conflict of interest inherent in self-regulation and thus assure that safety will receive consistent and adequate consideration; external regulation can thereby help restore the Department's credibility; it can bring increased insulation from political change, and thus increase the stability of safety policy in the Department; it can increase the efficiency of the Department by eliminating the redundancies in internal self-regulation and allowing the Department to focus on its missions; and it can make clearer just who the regulator is, and thus help induce clearer lines of accountability in the Department.

The Committee considered several possible distributions of regulatory responsibility among external agencies. It spent some time articulating an option in which DOE would be regulated by a new agency with plenary jurisdiction over all aspects of safety at all DOE facilities, and with the power to choose the best from existing safety standards and processes, focusing particularly on risk-informed prioritization of hazards, and, where possible, performance-based standards. Some members of the Committee hoped that such an agency might become a model for regulatory reform in the private sector. In the end, however, a large majority of the members decided that it would be less disruptive of existing external regulation of the Department, and less likely to aggravate the wide-spread perception that DOE is getting special treatment, if DOE were regulated by existing agencies. The Committee recommended the following distribution of regulatory responsibilities:

An existing agency -- either the NRC or a restructured DNFSB -- would regulate facility safety at DOE nuclear facilities. (In the Committee's view, neither the NRC nor the DNFSB was at present the kind of facility safety regulator the Committee thought would be best.)

OSHA would regulate all protection of workers at DOE nuclear facilities, unless regulation of worker risks at a given facility could significantly interfere with maintenance of facility safety (for example, if criticality is possible), in which case the regulator of facility safety would regulate all protection of workers at the facility.

EPA would continue to regulate environmental protection for all DOE nuclear facilities.

States with programs authorized by the Federal regulators would exercise roles in environmental protection, worker protection, and facility safety comparable to the roles the States now exercise in the regulation of private nuclear facilities. States would have the authority to set more stringent standards as long as those standards did not unduly hinder DOE's performance of its missions (the Committee took a cue here from the analogous standard in the Occupational Safety and Health Act).

This framework would apply to defense facilities also, with certain limited exceptions. Following the lead of a subcommittee whose membership was almost as diverse as the full Committee's, the full Committee concluded that it was possible and highly desirable to protect safety and national security at the same time. The essential element needed to preserve national security interests is a legislative provision for an effective and prompt method of invoking Presidential authority in the rare case of a conflict that the Department and an external regulator cannot resolve. The Committee made two exceptions to the rule of external regulation of defense facilities. First, to protect the most sensitive design information, the Committee recommended that DOE continue to regulate "nuclear explosive safety", namely the formalized program designed to prevent the accidental or unauthorized detonation of high explosives in proximity to special nuclear material. Second, at least in the early stages of a fully external regime, safeguards and security should be left to DOE, to avoid diluting the new external regulators' focus.

In the Committee's recommended framework, the agencies do not have quite the same jurisdictions over hazards that they have in the private sector. In private sector regulation, the NRC sets some standards for environmental protection and worker protection. In the framework set out by the Committee, the facility safety regulator would be limited to facility safety, except in a limited number of cases involving worker protection. The Committee took this approach for fundamentally two reasons: to avoid disrupting existing external regulation of environmental protection at DOE, and to help assure that radiation hazards to the environment and workers are considered in the context of all hazards to the environment and workers.

Of course, the framework just described would require legislation, and the Committee's final Report contains a list of the legislative implications of these and other recommendations of the Committee's.

OTHER RECOMMENDATIONS

The Committee made several other recommendations, some which would apply whether increased external regulation of the Department comes about or not. The chief of these other recommendations are discussed briefly below.

"Corporate" Safety Function

The ultimate responsibility for safety lies with the regulated party. Therefore, even under external regulation, DOE must retain sufficient resources to assure that it implements policies and practices which are

consistent with external requirements. This is particularly important in the area of worker protection, because OSHA's resources are quite limited. However, if the practice of large, multi-site corporations is any guide, those resources can be modest. Department resources now being devoted to regulatory functions -- standard-setting, inspection, enforcement -- could be moved to the new external regulators.

DOE the Regulated Party

To help assure that the highest level of management of a nuclear facility is in step with regulatory requirements, the Committee recommended that DOE be a regulated party -- for example, the "licensee", or the "permittee" -- at every site, and that contractors at some sites also be regulated parties.

Lead Agency

The Committee urged the complex-wide adoption of a "lead agency" approach, in which DOE would have to deal with only one regulator for any given matter at a given site. The Committee recommended that this approach be "facilitated" by legislation.

Citizens' Roles

The Committee strongly supported giving citizens effective roles at all stages of the regulatory process, from setting standards, to applying them in approval actions like licensing, to enforcement. Concerned that the trial-type hearings conducted by the NRC in nuclear power plant licensing proceedings consume a disproportionate amount of resources, the Committee recommended that legislation provide for a simpler, less formal, hearing process for use at DOE nuclear facilities. The Committee was particularly concerned that federal enforcement against DOE be as effective as federal enforcement against private entities. To this end the Committee recommended that citizens be able to sue the Department for not complying with external safety rules, and that citizens be able to sue the external regulators for not performing their duties.

One Size Does Not Fit All

The Committee strongly recommended the use of a variety of regulatory models, ranging all the way from "audited self-regulation" of the sort now employed by OSHA in its Voluntary Protection Program, to the full licensing proceeding of the sort employed by the NRC in licensing nuclear power plants. DOE's nuclear facilities and missions are too varied to be subjected to the same regulatory model. The Committee also urged the use of performance-based criteria wherever possible, to provide the regulated party with the flexibility to achieve cost-effective solutions to safety problems.

WHAT WILL THIS COST?

Given its limited resources, and the unpredictable nature of some of the possible costs, the Committee was not able to estimate the costs of putting in place the regulatory framework it recommended. After a careful cataloging of DOE's nuclear facilities, and a comparison of what it costs the NRC to regulate whatever comparable facilities there are in the private sector, the NRC estimated that it would need 1100-1600 more FTE and $160 million to $200 million annually to regulate what the Committee recommends be assigned to the facility safety regulator (given the number of facilities to be externally regulated, and the sizes of DOE's budget and workforce, these estimates do not seem unreasonable). The DNFSB did not give the Committee any estimate. DOE was unable to say how much it spends on self-regulation.

However, the Committee did make some general observations on what the likely costs of implementing its recommendations would be. The amount the Department spends on internal oversight should decrease; there will be transition costs; the cost of complying with external requirements may go up in the short term. Each regulator should be ensured the resources it needs to carry out additional responsibilities; regulators without adequate resources are only paper tigers, and a transition to a badly funded regulatory framework will only weaken the assurance of safety and reduce credibility. Nonetheless, the cost of regulatory oversight is a small part of the total cost of meeting any set of safety requirements, and there is a real potential for savings through greater efficiency and productivity in DOE's performance of its missions, once the more straight-forward, credible, accountable, external regime recommended by the Committee is put in place.

TRANSITION

The Department has underway a number of laudable safety initiatives. The Committee urged that these be continued. The Committee also recommended that, pending transfer of regulatory authority from DOE to external regulators, all internal regulatory functions in the Department be consolidated in the Office of Environment, Safety, and Health (OESH). For example, at the present time, program offices must concur in new safety standards; the Committee believes that OESH should have sufficient authority to establish standards independently of the program offices. The Committee also recommended that the transition to OSHA regulation of the Department's nuclear facilities, which the Department has long sought, be expedited. To the DNFSB, the Committee recommended that the Board make its processes more open, for instance by issuing its Recommendations in draft form for public comment.

The Final Report contains several cautions about the transition to external regulation. It will be difficult; it will take time; safety must not become the victim of inattention while everyone waits for the new regulator to take charge. However, there have been similar transitions in the past, and much can be learned from these.

DOE RESPONSE

Secretary O'Leary has formed a DOE working group to recommend to her by March 20 what response the Department should make to the Committee's recommendations. The group is head by Acting Under Secretary Thomas Grumbly. Members of the group will include representatives from the Department of Justice and the other two agencies to which the Committee's Final Report was directed, namely, OMB and CEQ. The Secretary has also directed the Department's working group, in coordination with the external regulators, to come up with an implementation plan by May 15.

CONCLUSION

When the Committee's Co-Chairs transmitted the Committee's Final Report to DOE, OMB, and CEQ, the Co-Chairs said that they were "pleased that this distinguished and diverse Committee, which represented the broad range of interests involved in regulation of DOE, has been able to reach conclusions on the difficult issues it has had to consider." The Committee believes that the external regulatory framework it recommends will clear the path to greater productivity and efficiency within the Department by removing regulatory redundances and overlaps and by ensuring a regulatory framework that remains stable as Administrations and officials change. The flexibility the Committee urged in the regulatory approach through, for example, relying where possible on

performance standards rather than rigid prescriptions will allow a more cost-effective approach to ensuring safety. External regulation can help improve public confidence in DOE and provide increased assurance that its future record of nuclear safety will be free of the mistakes of the past. The full title of the Committee's Report is Improving the Regulation of Safety at DOE Nuclear Facilities, and it is available from the DOE Environment, Safety and Health Information Center by calling 1-800-473-4375. The Report and related material can be found on the World Wide Web at http://www.em.doe.gov/acd/index.html.

5-3

ENVIRONMENTAL RADIATION STANDARDS FOR YUCCA MOUNTAIN

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ABSTRACT

The Environmental Protection Agency (EPA) has been given the responsibility of setting site-specific health and safety standards for the potential repository at Yucca Mountain, Nevada. The same legislation that gave the Agency that responsibility, mandated a study by the National Academy of Sciences (NAS) to provide input into the bases for the EPA standards. The NAS has completed and presented a report to the Agency; this paper summarizes the report's recommendations and conclusions. Following receipt of the report, the Agency opened a comment period and held public meetings to gather comments; the major issues from the comments are summarized. Based upon the report from NAS and the public comments, EPA has started formulating proposed standards which will be known as 40CFR Part 197. It is planned for the proposal of 40 CFR Part 197 to occur in the Spring of 1996.

HISTORY

In 1985, the Environmental Protection Agency (EPA), under the authority of the Atomic Energy Act, issued 40 CFR Part 191, Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Waste (1). These are generic standards and, as such, apply to all pertinent facilities, including Yucca Mountain (which has since been exempted by legislation, see below), the Waste Isolation Pilot Plant (WIPP), and the Greater Confinement Disposal Facility. With regard to Yucca Mountain, this situation changed seven year later.

In October 1992, two Federal laws were enacted, the Waste Isolation Pilot Plant Land Withdrawal Act (WIPP LWA) (2) and the Energy Policy Act (EnPA) (3). While the WIPP LWA dealt mainly with WIPP, a potential repository, in New Mexico, for transuranic radioactive waste, it also exempted Yucca Mountain from coverage under 40CFR Part 191. At the same time, the EnPA directed EPA "to set generally applicable standards for the Yucca Mountain site...for protection of the public from releases from radioactive materials stored or disposed of in the repository at the Yucca Mountain site." It also directed EPA to contract with the National Academy of Sciences (NAS) to "conduct a study to provide...findings and recommendations on reasonable standards for protection of the public health and safety." In Section 801 of the EnPA, Congress also mandated responses to three inquiries:

"(A) whether a health-based standard based upon doses to individual members of the public from releases to the accessible environment (as that term is defined in the regulations contained in subpart B of part 191 of title 40, Code of Federal Regulations, as in effect on November 18, 1985) will provide a reasonable standard for protection of the health and safety of the general public;

(B) whether it is reasonable to assume that a system for post-closure oversight of the repository can be developed, based upon active institutional controls, that will prevent an unreasonable risk of breaching the repository's engineered or geologic barriers or increasing the exposure of individual members of the public to radiation beyond allowable limits; and

(C) whether it is possible to make scientifically supportable predictions of the probability that the repository's engineered or geologic barriers will be breached as a result of human intrusion over a period of 10,000 years."

THE NATIONAL ACADEMY OF SCIENCES REPORT

In early 1993, the Agency established a contract with the NAS to produce the study required by the EnPA. The Agency received that report, entitled Technical Bases for Yucca Mountain Standards (hereafter called the NAS Report), on August 1, 1995 (4). The NAS divided the results of the study into two categories, recommendations and conclusions. The remainder of this section summarizes the recommendations and conclusions; the interested reader is urged to obtain the entire NAS Report to read the entire rationale.

Recommendations

In the Executive Summary of their report, the NAS recommended:

(a) the use of a standard that sets a limit on the risk to individuals of adverse health effects from releases from the repository;

(b) that compliance with the standard be measured at the time of peak risk, whenever it occurs within the limits imposed by the long-term stability of the geologic environment (which NAS judged to be about one million years);

(c) against a risk-based calculation of the adverse effect of human intrusion into the repository since it is not possible to assess the frequency of human intrusion far into the future; (d) that the consequences of an intrusion be calculated to assess the resilience of the repository to human intrusion;

(e) that in those instances where science cannot provide the guidance necessary to resolve an issue, policy questions should be resolved through a rule making process that allows opportunity for wide-ranging input from all interested parties;

(f) that the critical-group approach for assessing individual risk be used; and,

(g) that EPA require that the estimated risk calculated from a specified, assumed intrusion scenario be no greater than the risk limit adopted for the undisturbed-repository case because a repository that is suitable for safe disposal should be able to continue to provide acceptable waste isolation after the specified intrusion.

CONCLUSIONS

Also in the Executive Summary, the NAS concluded that:

(a) an individual-risk standard would protect public health, given the particular characteristics of the site, provided that policy makers and

the public are prepared to accept that very low radiation doses pose a negligibly small risk;

(b) the physical and geologic processes are sufficiently quantifiable and the related uncertainties sufficiently boundable that performance can be assessed over time frames during which the geologic system is relatively stable or varies in a boundable manner;

(c) it is not possible to predict, on the basis of scientific analyses, the societal factors for an exposure scenario. Specifying exposure scenarios, therefore, requires a policy decision that is appropriately made in an EPA rule making process;

(d) with respect to the second question of Section 801, it is not reasonable to assume that a system for post-closure oversight of the repository can be developed, based upon active institutional controls, that will prevent an unreasonable risk of breaching the repository's engineered barriers or increasing the exposure of individual members of the public to radiation beyond allowable limits;

(e) with respect to the third question in Section 801, it is not possible to make scientifically supportable predictions of the probability that a repository's engineered or geologic barriers will be breached as a result of human intrusion over a period of 10,000 years; and,

(f) there is no scientific basis for incorporating the ALARA (as low as reasonable achievable) principle into the EPA standards or Nuclear Regulatory Commission (NRC) licensing regulations for the repository.

PUBLIC COMMENTS

Shortly after receiving the NAS Report, the Agency opened a comment period and held public meetings to receive comments and to explain the rule making process. The public meetings were held in Amargosa Valley (in Nye County) and Las Vegas, Nevada and Washington, D.C. About 60 comment letters, including about 30 copies of a form letter, were received. There was no transcript of the public meetings; the only formal record of the oral comments that was made is in an EPA summary document (5). The reason for no transcript was that these were not hearings and the feeling that the free interchange of ideas and questions would be encouraged in a less formal atmosphere. The majority of comments and questions were from private citizens. However, there were also comments and questions from Federal agencies, the State of Nevada, Native American groups, several counties, industry groups, environmental organizations, and public interest groups, The major themes of those comments were:

- agreed with the NAS that no system of active institutional controls can preclude human intrusion;
- agreed with the NAS that scientific predictions of human intrusion cannot be made;
- there was a split in opinion on whether a critical-population-dose limit would adequately protect the general population. Those who agreed also pointed out that the final decision must be made through a public-policy making process such as the EPA rule making and that the limit should be set at approximately one chance in a million of contracting a fatal cancer. Those who disagreed had three reasons: 1) use release limits, as was done in 40 CFR Part 191, would avoid dilution; 2) use a limit on maximum-individual and population doses since risk is not an enforceable quantity; and 3) standards should not be lower than in 40 CFR Part 191;

regarding the definition of the critical population group, several considerations were suggested, e.g., base the size and location on the ground-water pathway and assume the high end of potential intake rates; the group could be located in either Nevada or California but at any rate wherever the ground water exits the ground or is closest to the surface; the probability-based critical population group is no understandable; use the subsistence-farmer group;

disagreed with NAS on use of the negligible incremental risk concept, i.e., do not use it in the standards;

there was a split on the NAS suggestion to eliminate subsystem requirements. Some thought they were useful and essential; others thought that there were potentially counterproductive and stifling;

respect the ideology of native people;

how to deal with uncertainty was not sufficiently addressed by NAS;

calculate risk until the peak is reached; one commenter was concerned with being able to show reasonable expectation of compliance over hundreds of thousands of years and suggested that the time frame be much shorter, e.g., 10,000 years;

base the standards on health protection not on the projected containment capability of Yucca Mountain;

the NAS did not give a sufficient justification for not using the ALARA concept;

include doses from the Nevada Test Site and the low-level waste disposal systems in the area;

work closely with the NRC to make standards amenable to licensing; and,

establish an information center closer to Amargosa Valley.

AGENCY ACTIONS AND DIRECTION

The Agency has taken a number of actions to keep the public informed about the Yucca Mountain rule making. It has established a toll-free, 24-hour recording for the rule making which can be reached by calling 1-800-331-9477. There is also EPA's Technology Transfer Network electronic bulletin board which can be reached by calling 919-541-5742 or via TELNET ttnbbs.rtpnc.epa.gov. The Agency is also working to develop a home page on the World Wide Web.

Also, the Agency has taken definitive action on two of the public comments -- the final two listed above. In line with the comment, the Agency and the NRC have each named a technical liaison to facilitate the interchange of technical data and issue resolution. The Agency recognizes that the time limits placed upon it by the EnPA allow only limited, independent, technical analyses and that the NRC has been studying Yucca Mountain for many years. Therefore, the Agency anticipates that NRC will be a significant source of technical data. To date, the liaison approach has proven to be very useful and successful.

As for the final comment, in September, the Agency worked with the staff of the public library in Amargosa Valley to establish an information center. While the intent is to place the same information into it as will be placed into the formal dockets in Washington, DC and Las Vegas, Nevada, the designation of information center relieves the limited library staff from the necessity of keeping a formal, legal file.

The other comments are the heart of the decisions that will need to be made as a result of the rule making. The Agency has taken into account many of the comments as it formulates the proposed standards. However, it is premature to report at this time on the outcome of any particular issue since the Agency has not reached any definitive positions on them.

However, the Agency has two goals. One goal is to set standards that assure both that Yucca Mountain will not open unless it is safe and that it will be able to open if it is safe. The Nation can neither afford to emplace the wastes into a site that is not safe, nor to needlessly abandon a safe site. Another goal is to set standards that are protective of public health and the environment and that are implementable by NRC. Only standards that meet both of these goals will be acceptable to the public. Public acceptance is crucial, without it, nuclear waste disposal programs will fail. The public's perception of the credibility of the Yucca Mountain evaluation could impact all nuclear programs in the U.S.

CURRENT STATUS AND THE NEXT STEPS

The Agency has received the NAS Report, held public meetings, received comments, and is formulating proposed standards, to be known as 40 CFR Part 197. It is anticipated that the proposal will be published in the Federal Register in the Spring of 1996. The next step will be to produce a proposed set of standards and a set of draft background documents. There will then be a public-comment period and public hearings on the proposed standards. This will be followed by final standards, background documents, and a response-to-comments document. These will be completed as expeditiously as prudently possible.

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5-4

EFFECTIVE STRATEGIES FOR STREAMLINING NRC'S WASTE MANAGEMENT PROGRAM UNDER REDUCED BUDGETS

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC), Division of Waste Management (DWM) has been examining its regulatory strategies for ways to

optimize protection of the public health and safety and the environment while minimizing the burden on licensees from the regulatory framework, all in the face of a declining budget. Based on this examination, NRC staff has revised its regulatory strategies in the areas of uranium recovery (UR), high-level waste (HLW), and decommissioning of Site Decommissioning Management Plan (SDMP) sites.

In May 1993, the staff identified four areas in the UR licensing process which could be revised to reduce the regulatory burden on UR licensees while ensuring the same level of protection to the public and the environment. DWM is now implementing a program which has streamlined the licensing process by: 1) using more flexible license conditions that incorporate license criteria rather than prescriptive, detailed conditions; 2) implementing a performance based license condition approach that allows licensees to make changes to their facilities, under certain conditions, without NRC approval; 3) eliminating dual regulation of in-situ leach well fields by relying on State reviews; and 4) relying on reviews by States concerning cultural artifacts.

In 1995, DWM revised its HLW prelicensing program strategy in response to budget reductions and the U.S. Department of Energy's (DOE's) new program approach for streamlining site characterization, site suitability determination, and licensing. DWM's revised HLW program strategy is still based on the Overall Review Strategy (ORS) developed in 1994, but its implementation has been modified by using a "vertical slice" approach to provide more focused and timely feedback to DOE. The vertical slice approach builds on the strategies in the ORS for conducting prelicensing reviews and for developing review capability that are focused on, and limited to, key technical issues most important to licensing. Each technical issue is addressed with one or more sharp cuts (thus the term vertical slice) encompassing an appropriate range of review and review capability development activities.

DWM recently streamlined decommissioning procedures in a management plan as part of the annual update of the SDMP. This management plan is directed at accelerating remediation of SDMP sites and minimizing the regulatory burden imposed on licensees and other responsible parties. The plan calls for revising current procedures to reduce the amount of NRC effort invested in site characterization and confirmatory radiological surveys. Revised procedures will require, under most circumstances, the staff to review site characterization information and decommissioning plans concurrently. NRC is also revising the procedures by placing a greater emphasis on the termination radiological survey conducted by the licensee or responsible party, with a corresponding reduction in the extent of NRC confirmatory surveys.

This paper will describe in detail the revised regulatory strategies being implemented by DWM in the areas of UR, HLW and decommissioning of SDMP sites. Further, the rationale for the program changes and the advantages and disadvantages of these changes will be discussed.

INTRODUCTION

In an effort to optimize protection of the public health and safety and the environment and minimize the burden on licensees in a period of declining budgets, NRC has initiated efforts to streamline the licensing process in high-level waste, uranium recovery, and decommissioning program areas. In each of these areas the staff has evaluated the regulatory approach and identified improvements directed at enhancing public protection, improving public participation in the regulatory

process, facilitating communication with licensees and improving the timeliness of regulatory actions.

URANIUM RECOVERY PROGRAM

In May 1993, NRC committed to explore ways to reduce the regulatory burden of uranium recovery licensees withoutcompromising protection of health and safety and the environment (NRC, 1994). The staff determined that there were four areas in which the regulatory impact might be reduced. First, licensees could decrease the number of amendment requests by using more flexible "criteria based" license conditions rather than extremely specific conditions, such as conditions that specify particular individuals or organizational structures. Second, licensees could implement a performance based license condition that would allow changes to their facilities, under certain conditions, without NRC approval. Third, NRC staff can rely on State reviews, thereby eliminating dual regulation of in-situ leach facility well fields provided State oversight is comparable to current NRC requirements. Fourth, NRC can rely on State review of Cultural Artifacts to avoid dual regulation. Each of these areas is described in more detail below.

More Flexible License Conditions

The staff believes it is possible to reduce the regulatory burden on licensees by implementing more flexible license conditions. In its review of several license amendments, the staff found that many amendments resulted from extremely specific license conditions. For example, some license conditions specified licensee facilities, organizations and in some cases individuals, with the effect that licensees could not change these condition specifications without submitting a license amendment. The staff has determined that a number of license conditions could be changed to more flexible "criteria based conditions." Such conditions could identify criteria for specific staff positions (such as radiation safety officer) rather than identify specific individuals. Similarly, organizational criteria could be included in the license condition rather than specific organizational structures. However, such criteria based conditions must be specific to individual licenses, and do not constitute generic changes that NRC can make industry-wide, by rulemaking.

Even if criteria based conditions are implemented, there are some license condition changes that could not be made without prior NRC approval. Examples include a change in facility ownership, and a change in control of the license. However, in general the staff believes that criteria based license conditions that would allow licensees to make minor changes without prior NRC approval, are acceptable.

Performance-Based License Conditions

A second area where the staff found regulatory reduction could be achieved was in the preparation of a performance-based license condition. The staff developed a performance-based license condition which is consistent with the Commission's regulations and licenses for other facilities. The performance-based license condition was modeled on the provisions of 10 CFR 50.59, which allow 10 CFR Part 50 licensees to make changes to, or conduct tests and experiments at, their facilities without prior NRC approval unless the change, test, or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question. The performance-based license condition is structured such that uranium recovery licensees are required to submit applications for all license amendments, unless it can be demonstrated that the provisions specified in the performance-based license condition

have been satisfied. In addition, the performance-based license condition requires that a summary of all changes made under that condition be provided to NRC in an annual report. Therefore, the performance-based license condition provides the same degree of flexibility contained in the regulations and licenses for other nuclear facilities, and is consistent with established NRC policy.

If licensees decide to incorporate the performance-based license condition into their licenses, they would have the burden of ensuring proper implementation of the condition. Summaries required by the condition coupled with information gained from inspections would allow the staff to determine if a licensee had properly implemented the condition. Improper implementation would result in a violation of the license and the licensee would be subject to possible NRC enforcement action.

NRC's Regulatory Role Over In-Situ Leach Facility Well Fields

The staff also believes a reduction in the regulatory burden of uranium recovery licensees can be realized by eliminating dual regulation of groundwater in in-situ leach well fields. Well fields at in-situ leach operations are also regulated by the U.S Environmental Protection Agency (EPA) and State agencies in non-Agreement States. Based on NRC's evaluations, it is acceptable for NRC to rely on the State's groundwater regulatory program provided the program is comparable to NRC's. This approach would allow the staff to ensure that necessary oversight is being achieved while eliminating dual regulation.

The staff is currently meeting with the States that have NRC licensed in-situ leach facilities to determine whether the States would be willing to provide such oversight and whether the States' programs are comparable to the NRC's.

Archeological Surveys

The fourth and final way the staff proposes to streamline the licensing process is by relying on State reviews of licensee archeological surveys thereby eliminating dual regulation. NRCis required by the National Historic Preservation Act of 1966 and the National Environmental Policy Act of 1969 to evaluate the effects of a licensee's proposed action on cultural resources. However, NRC need not conduct these reviews independently, provided the staff can use reviews conducted previously by the State Historic Preservation Officers.

HIGH-LEVEL WASTE PROGRAM

NRC's revised HLW program strategy for the prelicensing program is based on the ORS (NRC, 1994). The ORS was prepared in 1994 to provide the principal policy guidance to the staff for conducting its prelicensing program and license application review. Using the graded approach established in the ORS, the staff will conduct more detailed reviews, using independent assessment methods and results of applied technical investigations for those technical areas most important to repository performance. Therefore, these key technical issues (KTIs) help focus and link together the staff's work, in all parts of the program, on what is most important to licensing.

The new program strategy, termed "vertical slice" approach, builds on the strategies in ORS for conducting prelicensing reviews and for developing review capability for the 10 KTIs most important to repository performance and licensing (NRC, 1995). To implement the vertical slice approach, the staff will develop a review capability and apply it to reviewing DOE's program in selected areas, on an audit basis. This audit

approach is efficient in times of reduced budgets, yet effective because of its emphasis on resolution of KTIs important to repository performance. Further, additional efficiency is gained by establishing NRC staff work schedules based on providing DOE feedback consistent with DOE's major milestone schedules.

To date, the staff has identified the following 10 KTIs;

1. Unsaturated and Saturated Flow Under Isothermal Conditions
2. Igneous Activity
3. Radionuclide Transport
4. Structural Deformation and Seismicity
5. Thermal Effects on Flow
6. Repository Design and Thermal-Mechanical Effects
7. Evolution of the Near-Field Environment
8. Container Life and Source Term
9. Total System Performance Assessment and Technical Integration
10. Support Revision of EPA Standard and NRC Rule.

In addition, technical needs have been identified for each KTI. Technical needs are the work activities the staff feel must becompleted to address the KTIs. Based on the results of these work activities, the staff will comment on the sufficiency of DOE's site characterization program and evaluate the effectiveness of DOE's overall program for preparing an acceptable license application. Consistent with NRC and DOE meeting agreements, issue resolution will be achieved during prelicensing when NRC agrees with the DOE, at the staff level, on the acceptability and sufficiency of the information for eventual licensing.

Numerous advantages are associated with the vertical slice approach. First, the HLW program is focused by giving priority to those staff activities related to the most important licensing issues, and providing timely feedback to DOE. Second, integration of the NRC program is improved by coordinating the necessary activities and technical disciplines needed to prepare for and conduct reviews of each issue. This audit approach will also efficiently evaluate a wide range of DOE activities and how well they are integrated for use in licensing. Finally, in a time of declining resources, this approach is an efficient use of limited resources.

As with any program approach, the vertical slice approach has some inherent weaknesses. By focusing all the staff's prelicensing reviews on KTIs, many parts of DOE's program and associated sections of DOE documents will not be reviewed by the staff before submittal of the license application. Therefore, the license application may be incomplete, resulting in delays for the staff's acceptance review, for docketing the license application and for the compliance review. Another weakness is that by focusing on KTI's, the staff may overlook other potential licensing vulnerabilities in its reviews.

Despite its weaknesses, the staff believes the vertical slice approach is an effective and efficient way to streamline NRC's HLW repository program in response to declining budgets and changes to DOE's program.

DECOMMISSIONING PROGRAM

NRC staff is currently streamlining the decommissioning regulatory program, while ensuring effective oversight of decommissioning projects listed in the SDMP and other significant decommissioning actions at nuclear materials facilities (NRC, 1995). Regulatory streamlining is focused on revising existing procedures to reduce NRC effort devoted to site characterization reviews and confirmatory radiological surveys.

Site characterization

Under existing policies and procedures, NRC staff typically reviews site characterization plans and site characterization reports to ensure that licensees have established the extent and type of radiological contamination present before developing decommissioning plans. The staff plans to change the existing procedures for most sites by delaying the review of site characterization information until after the decommissioning plan is developed and submitted for review. This approach promotes a more coordinated and focused review of site characterization information because reviewers will be compelled to emphasize issues that affect the selection and implementation of a decommissioning approach. In most cases, reviewing the characterization data and the decommissioning plan concurrently will allow NRC staff to initiate the decommissioning plan review earlier and allow for parallel resolution of characterization and decommissioning issues which, in many cases, are interdependent.

Confirmatory Surveys

Currently, at the conclusion of remedial actions, a licensee or site owner conducts a termination radiological survey to demonstrate that residual radioactivity levels have been sufficiently reduced in accordance with NRC criteria. NRC then conducts a confirmatory survey to verify, on an audit basis, the results of the licensee's termination survey. These confirmatory surveys are discretionary for NRC; since regulations do not require confirmatory surveys. However, in recent years confirmatory surveys have become routine for SDMP sites.

In the future, the staff intends to reduce the number of NRC confirmatory surveys by placing greater emphasis on the quality of licensee's termination radiological surveys. Some level of confirmatory surveys will continue to be performed at each site. The extent of NRC's confirmatory surveys will be based on past licensee performance, on the result of NRC inspections of the licensee's survey while in progress, and on the results of the licensee's quality assurance/quality control efforts as reported in the termination survey report and as observed during inspections.

The staff is working with the EPA, DOE and Department of Defense to develop comprehensive guidance on performing termination surveys. This guidance should provide users with a method of performing and documenting termination surveys that will result in higher quality, more defensible termination survey data. The staff believes that higher quality licensee termination surveys will reduce the need for NRC confirmatory surveys.

CONCLUSIONS

In a period of constrained resources in both the public and private sectors, NRC staff is making every effort to optimize protection of the public and the environment while streamlining regulatory programs and minimizing unnecessary burdens on licensees. The staff is acting to make significant flexibility available to uranium recovery licensees if the licensees choose to file amendments that make their licenses more criteria-based. These changes are specific to individual licenses, and do not constitute generic changes that NRC could make industry-wide. In addition, the staff has developed a generic performance-based license condition that can be added to existing licenses, should licensees file an application for amendment to include this condition. The staff is pursuing the potential of eliminating dual regulation of in-situ leach well fields and cultural resources. Eliminating dual regulation can be

achieved in States where the staff finds that State oversight of the facilities is at least equivalent to that of NRC.

In HLW, a vertical slice approach has been developed to streamline NRC's repository program in response to declining budgets and changes to DOE's program. The vertical slice approach focuses on resolving KTIs with DOE, prioritizing staff activities based on issue resolution, improving the integration of all activities, and simplifying and orienting both NRC and DOE products toward what is needed for an acceptable license application. Prelicensing activities will focus on resolving KTIs with DOE at the staff level. The staff will then evaluate the overall effectiveness of DOE's program for preparation of an acceptable license application.

NRC staff efforts to streamline the regulatory process associated with the decommissioning of SDMP sites is focused on revising existing procedures for reviewing site characterization and confirmatory radiological surveys. These revisions will reduce NRC's level of effort applied to SDMP site regulation, while ensuring public and environmental protection.

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Co-chair: Mike Mauzy, Roy F. Weston

6-1

MINERAL PACKAGING OF LOW-LEVEL OR LOW AND MEDIUM-LEVEL ACTIVITY WASTES

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ABSTRACT

At The C.E.A., ceramic type materials are subject of active R&D programs to improve the conditioning of radioactive wastes. The conditioning of low or medium-level activity wastes is based on the tailored ceramics concept. $NaNO_3$ rich evaporator concentrates would be conditioned in nepheline $NaAlSiO_4$; $BaSO_4$ rich precipitation sludges, in the perovskite structure $BaTiO_3$. The conditioning of selectively extracted long-life radionuclides is envisaged in phosphate matrices; minor actinides and iodine in apatite structures, plutonium in monazite one.

INTRODUCTION

Conditioning of radioactive wastes in ceramic type materials formed subject of extensive R&D programs between '75-'85 (1). Nevertheless,

these efforts were brought to a halt afterwards. In France, the act of 30 December 1991 has since instigated new interest in this type of conditioning (2). Indeed, its guideline 3 encourages long-life and high activity radioactive wastes to be scrutinised by long-life surface storage and conditioning procedures. The C.E.A. (Commissariat l'Energie Atomique, French Atomic Energy Commission), D.E.S.D. (Dpartement d'Entreposage et de Stockage des Dchets), is developing a R&D program within this framework. These efforts are carried out in compliance with recent C.N.E. guidelines (Commission Nationale d'Evaluation) (3). This program aims to propose mineral matrices with performances higher than those of traditional ones (bitumen, cement, polymers), for the conditioning or medium-level activity wastes (B wastes as per the French classification), on the one hand. On the other hand, it aims at evaluating the feasibility and performances of crystalline matrices designed to condition selectively extracted long-life radionuclides. We shall discuss of "ceramisation" in the first case, and of "insertion" in the second one. Low-level activity wastes (A wastes) are equally concerned by ceramisation.

RESULTS

Ceramisation

Ceramisation primarily concerns B wastes (low thermal power, a activity >3.7 GBq.t-1) destined for geological disposal. These B wastes contain few radioelements; their mass being essentially made up of non radioactive additives. High activity wastes (C wastes) are presently satisfactorily conditioned in glass, however ceramisation could be a promising procedure for the conditioning of forthcoming sodium rich wastes.

Ceramisation is based on the "tailored ceramics" concept, developed in the 1980's by Harker et al. (4). The idea is to add chemical reagents to the waste in order to form a "customized" crystalline phase which, after consolidation, resembles a natural mineral of proven stability over geological periods ("natural analogue" concept). The final product is thus a matrix of high chemical stability, with improved long-term behavior. Furthermore, a judicious choice of the mineralogical structure to be formed allows optimal use of the waste's chemical constituents, minimization of the quantity of additives, and thus leads to a ceramic having a high waste incorporation rate, and a significant reduction in the conditioned waste's volume. As B wastes should make up the essential of deeply disposed wastes in the future, this second objective meets both economic and safety concerns as well as the concern to rationalize and optimize the management of geological disposal. Finally, as all ceramisation procedures included a high temperature waste calcination step, the products considered downstream from this operation will be chemically inert products; hence the increased safety and security of the conditioning procedure.

The advantages of the ceramisation procedure previously mentioned are in fact conclusive advantages, when choosing a B waste conditioning procedure. Nevertheless, they also merit to be taken into account when choosing a conditioning procedure for A wastes; wastes which will be disposed for 300 years in a surface facility. We shall present two examples where ceramisation can be applied, one concerning A wastes: the evaporation concentrates from a STEL (Station de Traitement d'Effluents Liquides, liquid waste processing facility); and the other, B wastes: precipitation sludges derived from decontamination processing of liquid

effluents from a reprocessing plant. Emphasis is placed on the original technical results obtained.

Ceramisation of evaporator concentrates originating from STEL

The purpose of the work undertaken is to substitute the conditioning of STEL evaporator concentrates by bitumen encapsulation with mineral packaging. In reality, this entails the design of a new industrial facility ensuring a highly stable conditioning process which allows, in addition to manufacturing safety, a reduction in the volume of packages intended for surface disposal. This concentrate takes the form of the very concentrated solution containing approximately 300 g.l-1 of mineral salts; its activity level is in the order of 0.15 GBq.l-1, the principal radionuclides being 137Cs and 60Co.

The average composition of the dry extract of this concentrate is:

NaNO3 81%
Na3PO4, 12H2O 6%
K3PO4, 2H2O 6%
NaCl 2%
Na2SO4 2%

In reality, the observed anion concentrations fluctuate considerably and batches rich in sulphate and phosphate are sometimes encountered. On the contrary, the cationic load of the concentrate, essentially made up of alkalines, is relatively constant; alkaline-earths being present only as minor compounds. Therefore, we have chosen the nepheline structure NaAlSiO4. This is a natural mineral which was extensively studied in the framework of the conditioning of sodium rich radioactive wastes, in particular for certain Synroc formulations (1). This compound has a resistance to water leaching that is perfectly compatible with the 300 year surface disposal period, as well as a capacity to incorporate a significant amount of sodium (21% Na2O). Furthermore, its structure allows sodium to be substituted with caesium.

In order to ceramise the nepheline waste, the essential problem that needs to be solved is the quantitative elimination of initially present anions (5). Obviously, this operation must be carried out without loosing radioelements. The direct calcination of the waste is not adapted to this as it leads to pronounced volatilization of the sodium, in oxide form, and especially of the radioactive caesium. One solution consists in treating the waste with oxalic acid H2C2O4, in the presence of Al2O3 and SiO2 additives. Drying of the transformed waste results in the elimination of nitrates, in HNO3 form, leading to a dry residue that is essentially made up of alkaline oxalates. Heat treatment of this residue leads to the formation of carbonates, near 400C, then to the reaction of these carbonates with silica and alumina, up to approximately 850C, producing alkaline aluminosilicate. The tests performed on simulated samples, according to the preceding composition, and doped with non radioactive cobalt and caesium, show that all the sodium initially present in the waste is found in the final compound. The quantity of the residual nitrates is in the order of 1% of the initial quantity. Lastly, all of the caesium and cobalt is confined in the final material.

The consolidation step is carried out after the powder is crushed and pressed to 100 MPa. Sintering takes place towards 1,100-1,200C (6). The study by X-ray diffraction of the ceramic obtained, using a typical-mean composition concentrate, only allows the nepheline phase to be detected. However, tests conducted on simulated wastes, highly loaded in sulphate and phosphate, show that the pre-treatment does not allow the

quantitative elimination of these anions. The presence of sulphate in particular, leads to secondary phases during sintering, especially nosean Na8Al6Si6O24SO4. Studies are underway to evaluate the confinement characteristics of this particular phase.

Various batches of real radioactive concentrates, sampled from STEL storage tanks, were ceramized according to the reference protocol (5). These tests showed that the quasi-whole of radioelements present in the waste were confined in the final ceramic: 100% yield for 60Co, 137Cs and 90Sr (6). Furthermore, the resistance to water leaching of the samples was tested according to the protocol in use for the surface-disposed wastes (static test, 23C, variable duration). These results were compared to those for a waste conditioned in bitumen. The leaching rate of 137Cs at 90 days is between 0.6 and 2.2 10-3 g.m-2.d-1; it is approximately 100 times lower than that observed for conditioning in bitumen. For the other elements or radioelements, the improvement is even more spectacular: a factor 150 for 90Sr, 3,000 for 60Co and 80,000 for non-radioactive sodium (6).

Finally, the mean factor of observed volume reduction, for ceramisation with regards to bituminization, is in the order of 4.5. Optimization or ceramic densification should allow reduction factors to be obtained in the order of 5.3 (6).

Ceramisation of reprocessing precipitation sludges

During the reprocessing of irradiated fuel, high activity solutions (which contain fission products) are generated, together with low and medium activity effluents. In France, at La Hague, these effluents are decontaminated by a coprecipitation procedure and radioactive sludge is obtained (B waste). 137Cs, 90Sr, 106Ru and 125Sb are the main radionuclides present in the waste which also contains significant quantities of a emitters. This sludge was still very recently conditioned in bitumen. Ceramisation of this waste was examined as an alternative route for its conditioning, and work has been conducted using inactive simulated sludge (7).

The main constituents of this sludge are:

BaSO4	68%
NaNO3	12%
Fe(CN)6Ni2	8%
CoS	9%

Simple heat treatment of this sludge, in air at 900C, leads to decomposition of all products into oxides, except BaSO4 which decomposes to BaO at approximately 1,500C. Moreover, BaSO4 does not have water leaching resisting characteristics.

Therefore, the main problem that needs to be solved to ceramise the waste concerns elimination of sulphates which must be obtained at a low temperature, compatible with the confinement of radionuclides in the final material. This problem, and that concerning the choice of the final conditioning, were solved according to the "tailored ceramic" concept by adding TiO2 to the waste. Indeed, adding this compound to the waste during its calcination in air led, from BaSO4, to the formation of BaTiO3 of perovskite type structure. This structure is that of one of the Synroc's essential constituents: CaTiO3; it is capable of incorporating the main radionuclides present in the waste and corresponds to the "natural analogue" approach discussed previously (1). This calcination reaction can lead to the total conversion of BaSO4 into BaTiO3 (yield near 100%). Its precise mechanism, which implies the formation of an

intermediate reaction product $BaTi_4O_9$, was clarified (8). The temperature of this reaction (near 1,200C), nevertheless remains too high; it is capable of leading to:

- volatilization of radioactive species or sodium oxide,
- pronounced pre-sintering of the precursor powder.

So, the reaction of $BaSO_4$ with TiO_2 was studied in a reducing agent environment (Ar/H_2 5%). In these conditions, the formation of $BaTiO_3$ is obtained near 850C; however, this reaction is hindered by the reduction of $BaSO_4$ to BaS. The formation yield of $BaTiO_3$ is near 80%. As BaSis extremely sensitive to leaching by water, its presence must be avoided. A solution consists in working in a humid atmosphere. In these conditions, BaS can be quantitatively transformed into BaO, which reacts strongly with TiO_2 giving $BaTiO_3$. The corresponding yield is near 100%. Identical results are obtained from the simulated sludge for which the main secondary phase obtained is $Na_2Ti_3O_7$. The powder obtained can be pressed and sintered directly without crushing.

The sintering of this powder can by conducted advantageously in a neutral atmosphere (Ar), at low temperature (1,100C), resulting in a relatively dense ceramic (open porosity 4%). Sintering in air must be avoided as it leads to a less dense material (open porosity 10%), at a higher sintering temperature, and with the appearance of new crystalline phases.

The ceramic's resistance to water leaching, resulting from the transformation of pure $BaSO_4$, was tested according to the standard MCC1 protocol (static test, deionized water, 28 days, 90C). The leaching rate of barium is 0.2 g.m-2.d-1 for a pellet of 10% open porosity, although only 0.05 g.m-2.d-1 for a pellet of 1% open porosity obtained from a higher quality powder. These performances can be compared with those of nuclear glass (9) and are, in fact, much better than those obtained with a conditioning matrix of bitumen type.

The preliminary results presented above allow us to define and propose a process for making a ceramic of high confinement capacity from reprocessing plant waste, at a low temperature compatible with the confinement of radioelements in final conditioning. Tests with real radioactive wastes would now be planned.

Insertion

The separation of long-life radioelements, from high activity solutions, is studied within the framework of guideline 1 of the act dated 30 december 1991, that indicates that work must be conducted on the "search for solutions allowing the separation and transmutation of long-life elements present in high activity and long-life wastes" (2). These studies are performed at the C.E.A. within the larger framework of the SPIN program (Sparation-Incinration, separation-incineration) (10); they aim at diminishing the radiological hazards that these radioelements represent, in the long-term. However, this kind of objective can equally be obtained by improving the confinement of these wastes. The insertion studies conducted at the D.E.S.D. reside in this second opinion, which is encouraged by the C.N.E. (3). They aim at developing specific conditioning matrices, having a very high confinement capacity for long-life radionuclides, after separation. The radioelements a priori concerned by these studies, are the minor actinides Am, Np, Cm; and very long-life fission products, mainly 135Cs, 99Tc, 129I, 93Zr, 107Pd (3, 10). Some of these products, such as iodine and neptunium, are already, or could be separated by the PUREX process.

The materials studied at the D.E.S.D. are phosphate (or vanadate) matrices of apatite or monazite type (11). The apatites form a family of compounds having the general formula $Me_{10}(XO_4)_6Y_2$, where Me represents a divalent cation, XO_4 a trivalent anion, and Y a monovalent anion. The most well-known representative of this family of compounds is phosphocalcic fluorapatite $Ca_{10}(PO_4)_6F_2$. The monazite is a lanthanide (Ln) phosphate: $LnPO_4$, where Ln is mainly La and Ce.

In general, these compounds are characterized as having very low solubility in water, retrograde in temperature, high chemical and thermal stability, and a strong capacity for self-healing of irradiation defects. In this respect, certain natural or synthetic apatites show a quite remarkable, even unique, capacity to have their irradiation defects annealed at a temperature as low as 100C (i.e., the "closing" temperature of fission tracks) (for comparison, the "closing" temperature is 250C for sphene, and 300C for zircon) (12, 13).

These minerals, monazites in particular, formed the subject of advanced studies on the conditioning of high activity wastes and proved to be matrices that are particularly well adapted to the conditioning of Rare Earths and actinides (1, 14). Furthermore, extensive substitution possibilities exist within the apatite structure which allow us to envisage wider usage of these materials for the conditioning of long-life radioelements.

Monazites and britholites (silicated apatites resulting from coupled substitution (Ca^{2+},PO_4^{3-}=Ln^{3+},SiO_4^{4-})) exist in the form of natural minerals which can contain significant amounts of Rare Earths and uranium/thorium (1, 13). The age of some of these minerals can be evaluated at several millions of years. The studies conducted at the natural reactor site of Oklo, Gabon showed the existence of silicated apatites that crystallized during nuclear reaction 2 billion years ago (15, 16). Furthermore, it can be proven that these minerals behaved, with respect to certain elements such as plutonium, Rare Earths or certain fission halogens, as stable trapping and confinement structures, over geological time periods and in conditions of particularly intense irradiation. Thus, at present, these apatites make up the most "natural analogue" example of a waste conditioning material. For a synthetic product formulated in an identical manner, such an established fact provides a serious guarantee concerning its long term behavior and forms a substantial argument in the choice of such compounds.

Here, we present the state of study progress on the conditioning of iodine 129 and minor actinides Am and Cm. Furthermore, the problem of conditioning plutonium will be briefly discussed. Plutonium is a recyclable material, and the best use that can be made of it is to recycle it to produce energy. However, its conditioning could be an alternative allowing for more flexible management of stocks, which could be created following disarmament agreements.

Conditioning of minor actinides

The conditioning of minor trivalent actinides, Am and Cm, is presently envisaged in silicated apatites of the same type as those which can be found at Oklo (17). As with lanthanides, trivalent actinides can be incorporated into the apatite network, thanks to the substitution of phosphates with silicates. Experiments are presently being conducted using Nd^{3+} or La^{3+}, classic inactive simulating of trivalent actinides (11, 18).

The introduction of silicates into phosphocalcic fluorapatite structure was accomplished through a solid-solid reaction at high temperature (between 1,100 and 1,400C, temperature being all the higher as the quantity of silicates to be incorporated is high). Synthesis is carried out from a mixture La2O3/CaF2/SiO2/Ca2P2O7/CaCO3, according to the global reaction:

x/2 La2O3 + CaF2 + 3 CaCO3 + x SiO2 + (6-x)/2 Ca2P2O7
Ca10-xLax(SiO4)x(PO4)6-xF2 + 4 CO2

with x = 1,2,3,4,5,6.

Characterizations by X-ray diffraction and infrared spectrometry show that the total substitution of phosphates with silicates was possible and that a solid solution exists (pure apatite phase) between the poles Ca10(PO4)6F2 and Ca4La6(SiO4)6F2. Similar results were obtained from neodymium. Moreover, the synthesis of oxyapatite Ca10-xLax(SiO4)x(PO4)6-xO was equally accomplished. The tests currently being conducted aim at evaluating and quantifying the resistance to water leaching, and the irradiation self-healing capacity of these synthetic materials.

Conditioning of Plutonium

The conditioning of plutonium presents a specific problem as the matrix envisioned must provide guarantees, not only concerning its long term stability, but also concerning non proliferation. Preliminary incorporation tests were conducted with uranium (+IV) which satisfactorily simulates plutonium (+ IV). Monazite was considered as it has interesting characteristics (1):

- difficulty of recovering plutonium by chemical solubilizing treatments,
- possibility to incorporate actinides in large proportions,
- possibility to easily incorporate gadolinium to deal with criticality problems.

Uranium doped monazites were made through a humid route process, followed by calcination in a reducing atmosphere Ar/H2 5%. Charge balance was obtained by substitution (Ce3+,Ce3+=Ca2+,U4+), and it was established that a solid solution exists from CePO4 to U0.5Ca0.5PO4 (12). The resistance to water leaching of these materials was evaluated by a Soxhlet test (dynamic test, deionized water, 58 days, 95C). The uranium release rate was in order of 0.01 g.m-2.d-1, regardless of the sample's uranium content. This dissolution is incongruent.

Systematic investigations of the synthesis via humid route were performed to correlate the precipitation conditions to the quality (morphology and chemical composition) of the powders obtained, as well as to the densification of the final ceramic (19, 20). These tests proved to be very delicate and led us to develop an original dry route process (21). This is based on the use of an intermediate compound, lanthanum metaphosphate La(PO3)3, as a reaction product to synthesize the monazite. The global solid-solid reaction can be expressed:

- for a tetravalent actinide (Ac):

La(PO3)3 + 3x AcO2 + 3x CaO + (1-3x) La2O3
3 La(1-2x)AcxCaxPO4

- for a trivalent actinide (Ac):

La(PO3)3 + x Ac2O3 + (1-x) La2O3
3 La(1-2x/3)Ac2x/3PO4

This synthesis is performed without a by-product, in particular gaseous by-products. According to our first experimental results, it can implemented, after pressing, by reactive sintering and leads to good quality ceramics (95% of the theoretical density). This procedure thus

minimizes the active steps; only 3 steps: mixing/crushing of reagents, pressing and sintering.

Conditioning of Iodine

The fission iodine (essentially 129I) is separated by the PUREX procedure in the reprocessing plant. Only a small fraction of this species is trapped on a solid support, the rest being discharged to sea. Discharge to sea being presently permitted, studies concerning iodine conditioning find their justification in the zero discharge objective aimed by reprocessing plants. This conditioning procedure should be able to be implemented downstream from a iodine trapping procedure, studied by the C.E.A., which leads to a lead iodide precipitate. This work was conducted from natural iodine, considering an apatite matrix (22, 23).

The choice of the apatitic compound is dictated by steric considerations; in particular, the significant size of the I- ion, and by the fact that the iodated waste to be conditioned will very likely be PbI2. The matrix chosen is vanadium-lead iodoapatite: Pb10(VO4)6I2, which can be obtained through solid-solid reaction (700C for 3h) in a closed environment:

$$3\ Pb_3(VO_4)_2 + PbI_2 \quad Pb_{10}(VO_4)_6I_2$$

Unfortunately, this product decomposes in air starting at 450C, according to the opposite of the preceding reaction, and cannot be correctly densified by natural sintering.

The technological solution which was tested and retained is a reactive sintering under pressure (24). The PbI2 pellet is placed in the center of a mould containing Pb3(VO4)2 in excess. Then, the assembly is sintered at 700C under 25 MPa. The product obtained is a dense matrix having good mechanical strength, composed of a core-formed iodoapatite, encased in sintered Pb3(VO4)2. These compound ceramics were made without iodine loss. The tests currently being conducted aim at optimizing the hot-pressing procedure with respect to the mechanical strength of the so-obtained composites, and quantifying the resistance to water leaching of the conditioning (measurement of the iodoapatite solubility in water in particular).

CONCLUSION

The conditioning of radioactive wastes in crystalline matrices is a present-day topic at the C.E.A.. These tests are being conducted in the framework of the act dated 30 December 1991. For B (and A) wastes, the objectives relate to an improvement of conditioning confinement properties, on the one hand, and to a reduction of the conditioned waste's volume, on the other hand. For separated long-life radioelements, the objective is essentially to significantly improve the confinement properties of the final conditioning. In the first case, the approach developed is of the "tailored ceramics" type; in the second case, the choice of the matrices is dictated by the "natural analogue" approach, the accent being placed on the recently obtained results in the study of minerals formed at the Oklo site. Extensive progress has been made in these two fields. Along these same lines, experimental resources, such as the "Btiment Moyenne Activit" (medium activity building) partly devoted to actinide conditioning experiments, are installed at Cadarache, and modelling studies on new conditioning matrices are underway.

ACKNOWLEDGEMENTS

The authors gratefully acknowledge Pr. Lacout, Mrs. Carpna, Miss Audubert, Mr. Alexandre, Mr. Berson, Mr. Boyer, Mr. Courtois, Mr. Fiquet, Mr. Lebeau and Mr. Fromont for their contribution to this R&D program and fruitful discussions.

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EXTRUDER-MELTER PROCESS CONCEPT FOR VITRIFICATION OF HIGH LEVEL AND OTHER NUCLEAR WASTES

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ABSTRACT
The vitrification process is being universally considered the accepted solution for the treatment of highly radioactive material in order to meet regulatory requirements. Also, vitrification of low and intermediate level radioactive material has recently been considered.

A major and most critical unit in the vitrification process so far is the glass melter. The melter operates at high temperatures of around 1150C; it has a large hold-up of material thus requiring long start-up and shut-down times in the range of 3-4 weeks; it releases volatile radioactive gases and the homogeneity of the glass it produces is sometimes questionable. The melter has to be replaced every two-three years on the average which is a disturbance to the process as well as being costly. In this paper an alternative unit to the glass melter, i.e. the extruder-melter is proposed. This unit is a modified plasticating extruder that would produce a complete mixing, melting and homogenization process, and would eliminate most of the inconveniences mentioned above with the glass melter as well as would reduce the cost of investment and operation. Based on the inherent properties of the extruder-melter such operating conditions as the temperature of the melt, the hold-up of material in the unit, the start-up and shut-down times and the volatiles released by the melt could be significantly reduced. Also, the product would be homogeneous due to good mixing of the melt in the extruder-melter. This application would require the development of an extruder-melter that would operate at high temperatures needed for the vitrification process, but would be lower than those used in a glass melter. Three extruder-melters could be operated in tandem so that particulate mixing and evaporation could occur in the first extruder, nitrogen oxide could be released in the second extruder and the glass could be formed in the third extruder. Twin-screw extruders, tapered extruders and other commercially available extruders should be evaluated for the proposed process.

The extruder-melter process could be tested first by using a model compound in the range of temperatures that existing extruders can handle. At a later stage, when an extruder-melter at high temperature operation is developed, experiments could be made with non-radioactive compounds in order to test the applicability of the unit to the real vitrification process.

Model compounds could also be used to determine the applicability of the extruder-melter for the elimination of water and nitrogen oxide from the aqueous slurries and the nitrates before real systems are tested. A reduction of the volatile Cesium oxide is expected due to both lower operating temperatures for the melt and the higher pressure that can be developed by the extruder-melter.

INTRODUCTION

During this last decade, the problem of high level nuclear waste has received much attention and a number of plants are being constructed or considered for construction for processing the high level nuclear waste materials. The immobilization method pursued by the DOE for nuclear waste from Defense Sites is the vitrification process. The radioactive elements present in the nuclear waste are dissolved or dispersed in a practically unleachable borosilicate glass, which in turn is encapsulated in stainless steel canisters and stored in repositories. This basic process is being considered for all US Defense nuclear waste treatment plants, even though the waste compositions are not the same in each plant. Also, other countries are using or considering using this means of treatment of their high-level nuclear wastes. Vitrification plants in the United States include two which have been built, and others which are in the design or construction stage. The first plant is the West Valley Demonstration Plant (WVDP) at the Western New York Nuclear Service Center

at West Valley, New York which has operated for five years at the high temperature needed to produce the glass. The Defense Waste Processing Facility (DWPF) has been constructed at the DOE Savannah River Plant at Aiken, South Carolina. Cold startup was scheduled to begin at the end of 1991 (Weisman et al., 1988). Conceptual designs have been made for the Hanford Waste Vitrification Plant (HWVP) at Hanford, Washington. Operation is not expected until the end of this century (23).

France has been leading the world in full industrial scale utilization of vitrification technology. They completed their second plant which began operation in 1990. The melters are designed to operate 2000 hrs with a replacement time of 48 hours. (Maillet and Sombret 1988 and Jouan et al. 1995). The British built a plant of the French design with two lines and started operation in 1990. (Eldsen and Woodal 1988). Germany, Japan, and Italy followed the same approach for the vitrification process (Tsuboya and Tsumoba 1988 and 13). The concept of incorporating a mechanical stirrer in the glass melter is being developed (3). Thus, the vitrification process is considered today as the solution to the nuclear waste material problem and shall probably be the accepted process for the foreseeable future. In fact the vitrification process may also become the preferred option for treating lower levels of nuclear waste and even other types of hazardous waste (3).

For the vitrification processes considered by the various plants, an extruder-melter could be utilized instead of the glass melter, which has the potential of making significant improvements over the glass melter in the areas of operating temperatures, material hold-up, mixing, start-up/shut-down times and economics.

TECHNOLOGICAL BACKGROUND

Vitrification in the U.S.A.

Several plants for vitrification of high level nuclear waste are being tested, constructed, or in the development stage in the USA as mentioned before. Also, a pilot scale radioactive liquid-fed ceramic melter (RLFCM) was operated by Pacific Northwest Laboratory (PNL). The Defense Waste Processing Facility (DWPF) at the Savannah River Plant was considered as a prototype plant for testing the process and equipment. This plant was used for testing under the Integrated Cold Run Program, before using radioactive material, including the testing of mixing conditions in the melter (Gentilucci 1989). The West Valley Demonstration Project (WVDP) is using a process similar to the DWPF for the vitrification of the commercial nuclear waste stored at the Western New York Nuclear Service Center (WNYNSC) (7,8). Although the waste composition is different the principle for vitrification is the same. However, special tests had to be made for the determination of the proper waste composition in each case and the characterization of the glass produced (Feng et al. 1988). Details on the facility and the demonstration project are described (6).

The Hanford Waste Vitrification Plant (HWVP) took advantage of the knowledge and experience gained from the DWPF and the WVDP. Experience from German and Japanese projects were also incorporated (23). The Idaho Chemical Processing Plant (ICPP) for high-level radioactive waste, located at the Idaho National Engineering Laboratory (INEL) in Idaho Falls, is developing an alternative process to the current glass melter. A waste immobilization plant is being considered for the waste disposal sometime in the year 2011. The new technology that INEL is developing is the Hot Isostatic Press (HIP) where calcinated waste used for feeding an isostatic compactor along with the glass frit (12).

For the vitrification project, the Department of Energy has sponsored the construction and operation of a pilot-scale radioactive liquid-fed ceramic melter operated by Pacific Northwest Laboratory (PNL) at the Radiochemical engineering facility at Hanford (4). This plant served as a pilot plant for the various vitrification plants under construction in the U.S.A. and Germany and has the major components considered by DWPF, WVDP and HWVP, such as a liquid-fed ceramic melter and a canister turntable. Various radioactive compositions of feed were used for the production of the borosilicate waste glass. The operation was terminated in 1987 (10).

Vitrification In Other Countries

The vitrification process for treatment of high level waste has been applied or is being considered in many other countries in the world as well, among them France, England, Germany, Japan and Italy (13,16). These developments are documented elsewhere and are not reviewed here.

Nuclear Wastes, Frit and Waste Glass Compositions

Nuclear wastes vary in their compositions and therefore different glass frit compositions are necessary in order to obtain the desired waste glass compositions. The variations in the frit and nuclear waste compositions affect the waste glass properties such as viscosities, densities, leachability, etc. In the glass making process the temperature is determined by the viscosity necessary for producing the proper mixing to achieve satisfactory homogeneity. In the joule-heated melter the temperatures are high at around 1150C. We believe that the extruder melter can handle materials of much higher viscosities and that lower operational temperatures, in the range of 900C, can be anticipated. Times required for the chemical reactions forming good glass are much smaller than the residence times that are obtained in present melter designs (2). In fact, most of the local homogenization occurs at the temperature of 850C which confirms that the
temperature in the melter of 1150C is needed more for the mixing by convective flow than for the glassification process.

The Glass Melter

The melter operates at a high temperature of about 1,150C and has a large material hold-up. Because of the harsh conditions in the glass melter, this unit must be remotely replaced about every two to three years. Due to the high inventory (hold-up) in the melter and the very high temperatures, the start-up time is very long, normally in the range of 3-4 weeks. The mixing in the melter is obtained by thermal convection (no internal mixers), thus the homogeneity of the product is questionable. The main reason for the high temperatures prevailing in the glass melter is the need for lower viscosities to promote good convection flow. These high temperatures are the cause for a higher rate of evaporation of Cs2O, which strongly contaminates the off-gas system of the glass melter. Recently the idea of introducing a mechanical mixer that would produce a better quality glass, a higher production rate and lower temperatures in the melter of only 1050C was reported by Bickford et al. (3). Some of the problems encountered with large melters were given by Parsons (17).

FUNDAMENTALS OF EXTRUDER-GLASS MELTER

Extruder Characteristics

Extruders are remarkable machines. They are rugged in construction and reliable in performance. Single screw extruders are not positive-displacement pumps but instead have open, tapered channels. Their ability to generate high outlet pressures is due to drag flow, and is mainly

dependent upon screw geometry and the viscosity of the slurry being pumped.

The single screw extruder can be designed for multiple functions, such as melting, mixing, grinding, and high-pressure pumping. It can still maintain near-plug flow and low residence time (low volume in process) while accomplishing the above functions.

A short description of the extruder characteristics and its applicability to the proposed project is in order. The extruder has the capability to pump very viscous materials and at the same time heat the material to the desired temperatures. The higher the viscosity, the more viscous dissipation is obtained through internal heating caused by the mechanical friction. Additional heat is added through the wall of the extruder barrel. An extruder schematic is shown in Fig. 1.

The classical use for the extruder is in polymer extrusion. The University of Arizona has used extruders extensively over the past 30 years not only for plastics but also in a biomass liquefaction plant, where highly concentrated cellulosic slurries were pumped into a reactor operating at 3,000 psi and around 400C (White et al., 1989). The near-plug flow condition in the extruder and its good mixing capabilities have already been determined analytically and experimentally (18,21). The recirculation flow and the high shear between the flights and the barrel which are tightly fitted are the main factors in the good mixing occurring in the extruder and for the homogeneous product obtained. Also, the channel depths in the mixing section are small, usually less than one inch, which also assists in developing high shear rates for good mixing. The start-ups and shut downs are very simple and easy and it is also easy to exchange the small amount of hold-up with inert material wherever necessary.

Fig. 1

Basic Plasticating Extruder Concept

An extruder consists of a helical screw that conveys material through a barrel and towards a die. The depth of the screw channel decreases from the feed section to the mixing section to compress the material. Details on extrusion theory and extruder performance can be found elsewhere. (14,18). Solid materials in the form of powders, pellets, flakes, beads, granular regrind, and highly-concentrated liquids or slurries may be fed to the extruder. The material is subjected to the necessary heating or cooling and develops sufficient pressure to force the viscous material through the die at the desired flow rate.

The helical geometry of the screw results in a "plug of solids" moving down the channel, and the flow pattern is composed of axial and tangential components which can be expressed as a force balance and as a torque balance, respectively. It is the fact that the torque balance must be satisfied that the screw of the extruder is an efficient conveyor of solids. It has also been proven by Wolf and White (21) that near-plug flow exists in solid screw conveying units.

Plasticating Extruder Operating Correlations

The basic equation found in the literature for calculating the net flow rate is as follows:

Eq. 1

Where: Q = net flow rate, Qd = drag flow rate, Qp = pressure flow rate, Q1 = leakage flow rate, N = rotational speed of screw, DP = P2 - P1 = pressure drop, m = viscosity of melt, A = constant, B = constant, C = constant.

The transverse flow is not included in Eq. (1) since it does not contribute to the net flow output. For the very small flight clearances between the barrel and screw that normally exist in extruders or even for higher clearances developed due to wear and for the very high viscosities that we deal with in extrusion, leakage flow over the flights can be neglected in Eq. (1), (22).

The more fundamental equation for the melting section that neglects leakage flow and flight width but takes into account various screw parameters and operational conditions is:

Eq. 2

Where: Vbz = pDNcosq, W = Dcosq, DZ = L/sinq, L = /n, = length of screw, M = number of flights, D = barrel diameter, H = depth of channel, q = helix angle, Fd = drag coefficient, Fp = pressure coefficient.

To obtain the pressure flow term in Eq. (2) it is necessary to know the pressure drop across the metering section and the viscosity of the melt. The viscosity is a function of both shear rate and temperature and it must, therefore, be determined for each experimental condition separately. For the extruder, the shear rate is obtained from:

Eq. 3

From the shear stress versus shear rate plots for several temperatures, the apparent viscosities are calculated and plotted versus shear rate for constant temperature or versus temperature for a series of thermoplastic materials that has been compiled.

The Proposed Vitrification Process

The proposed process and operation for high level nuclear waste vitrification would use an extruder-melter instead of the glass melter. An extruder 4-5 ft. long and with a screw diameter of about 2.5 inches is already considered a commercial unit. The hold-up of such an extruder would be around one liter. One could consider several extruders in tandem or a single long extruder with several sections.

The input to the first stage could be a sludge or a very concentrated solution of the high level radioactive waste and the other components for the glass. This unit would operate at a temperature of above 100C, thus causing the water and other volatile compounds to evaporate. Another stage would operate at temperatures of above 500C and could be used for the decomposition of nitrates, carbonates and other decomposable materials. The separation of the water vapor from the other gases would reduce corrosion problems. Yet another stage would operate at about 900C for the final stage of glassification. The correct temperature range will be determined so as to achieve pumpability and good mixing of the glass in the extruder-melter. The important point is that an extruder can pump slurries and achieve good mixing at much higher viscosities (and hence lower temperatures) than other mixing devices. An extruder canalso develop high pressures as needed for the pumping of the glass into the canisters reducing Cs2O evaporation and reducing the release of other gases that may be formed from the system.

The potential advantages of this extruder-melter vitrification method of particular interest are:

1. The low holdup or high throughput:
 a. Reduces the size of the glass-melter.
 b. The start-up and shut-down times are short due to the low hold-up of the extruder-melter.
 c. The low material hold-up of the extruder-melter would simplify disposal problems in the event of a failure.

2. The operating temperature for the glass formation would be lower since there is mechanical mixing in the extruder-melter and higher viscosities could be permissible, unlike in the Joule-heated glass melter, where high temperatures are needed to enhance mixing by melt convection. These lower temperatures are beneficial from the corrosion and the energy points of view and for reducing the vapor pressure of volatiles.
3. The high pressure in the extruder-melter would reduce the evaporation of the Cs2O and keep it in the melt. Also, a recycle line is envisaged between the outlet of the extruder-melter and the lower pressure inlet of the feed material, thus eliminating the possibility of escape of Cs2O.
4. Less corrosive gases will be produced since H2O, and NO2, will be drawn off separately.
5. Due to the plug flow characteristics of the extruder-melter and the good mixing due to high shear, recirculation flow and leakage flow between flights of the screw, the product will be well mixed and homogeneous.
6. No problems related to idling would exist in this type of unit operation. In fact a starved extruder-melter would be essentially empty of material. Thus, no problem of refractory metal particles or noble metal precipitates which accumulate in the conventional melter would exist in an extruder-melter (1).
7. The costs of investment and maintenance should be significantly lower.
8. The decontamination process will be simpler due to the small hold-up of the unit operation and its plug flow characteristics that would enable simple flash out of the system by non-radioactive material. In fact, with a starved extruder the hold-up is practically zero.

Details on the Model Glass

In order to test the ideas presented in this paper for the vitrification of high-level nuclear waste by the extruder-melter type process, one could use boric acid (H3BO3) as a model glass. This compound has viscosities similar to that of glass compositions but at lower temperatures and therefore can simulate all three stages needed in the real vitrification process namely dehydration (evaporation, drying), decomposition (calcination), melting, and off-gas treatment. This compound decomposes in two steps:

1. H3BO3 ---> HBO2 + H2O (at 190C)(12)
2. 2HBO2 ---> B203 + H2O (240C) (13)

The viscosity of the B203 depends on the temperature as given in the following table (15,19):

Table I

A sludge of H3BO3 as feed in the first step at temperatures above 100C, could be used to simulate the evaporation of water. Then, obtain the second step at temperatures of above 190C that would involve the decomposition of the boric acid to boron oxide and then produce glass with the following viscosities: 108.6 Poise at 350C; 107 Poise at 400C; 106 Poise at 420C. These viscosities are in the range of the viscosities of the borosilicate glass in the range of 550C-700C. In this simulation process we would expel gases in the form of H2O vapor in the first two steps of evaporation and decomposition, similar to expelling H2O and NO2 in the real vitrification process.

Materials of Construction for Extruder-Melters

Extruders to be considered for use in vitrification must be operating at high temperatures. Such extruders would need special metals to accommodate the strength and corrosion resistance needed at these high

temperatures. However, with the advancement in the development of special alloys for gas turbines, it is practical to select metal alloys which could provide the strength, corrosion resistance and thermal expansion limits required. Metals of the molybdenum alloy class could be one appropriate choice for this purpose. Also Inconel 690 is used at very high temperatures and could possibly be a candidate for the construction of the extruder-melter.

CONCLUSIONS

The scope of this paper was to present the concept of applying an extruder-melter for the high level nuclear waste vitrification process which could replace the presently used glass melter. Experimental work suggested in order to prove this concept would first include the vitrification of a model compound suitable to be operated at low temperatures in the 500C range. The low temperature operation would enable the use of presently available plasticating extruders. The test would prove the operability of the extruder melter in the three major steps involved, i.e. a) evaporation, b) decomposition, and melting. The next stage would be the development of an extruder-melter that is able to operate at around 900C. After such an extruder-melter is developed, non-radioactive elements that would simulate the real nuclear waste should be tested in the 900C range. Operability and homogeneity should be established. When this experimental work is successfully completed the proof of concept is achieved. For optimal design of the extruder-melter, theoretical analysis should be made in parallel to the experimental work.

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APPLICATION OF ADVANCED PROCESSING TECHNOLOGIES TO RADIOACTIVE AND HAZARDOUS WASTES

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ABSTRACT

Because of the issues related to disposal cost, long-term safety, environmental regulations, and public perceptions of risk from disposal, new treatment technologies are needed for processing of radioactive, hazardous, and mixed wastes. In this paper, three such technologies are briefly reviewed. These technologies are being applied at the waste processing facility of Scientific Ecology Group Inc., located in Oak Ridge, TN. The Quantum-CEP, a proprietary technology of Molten Metal Technologies Inc., is an innovative technology that will permit the effective processing of higher activity wastes such as the resins from nuclear power plants.

DRIVERS FOR NEW TECHNOLOGIES

New technologies are needed for processing of radioactive and hazardous wastes because of a number of factors. The primary drivers include:

1. Escalating disposal cost;
2. More stringent regulations and the need for environmentally stable waste forms;
3. Difficulties in dealing with mixed waste; and
4. Public perception of risk.

The uncertainty of access to a disposal site is a major concern of the generators of radioactive, hazardous, and mixed wastes. The Barnwell disposal facility that accepted most of the country's commercial low level radioactive waste closed to generators outside their compact in January 1995. Even though, the facility started accepting waste again in mid-1995, the uncertainty of access to a disposal site became very clear to many nuclear utilities. The state compacts that are responsible for developing low level radioactive waste disposal facilities are not making sufficient progress. After 15 years, most programs are stalled by cumbersome siting issues and public opposition to the development of new disposal facilities. No new disposal sites are expected to open any time soon, and it is now evident that disposal facility development is more complex, time-consuming, and controversial than originally anticipated. The disposal of high level radioactive waste and spent nuclear fuel awaits the opening of a federal facility at Yucca Mountain in Nevada, which may be two or more decades away. In the meantime, the spent fuel sits in wet or dry storage and high level waste is stored at Department of Energy's (DOE) tank storage facilities at Hanford and Savannah River sites.

The U.S. nuclear power industry, with 112 operating reactors, generates about half of the low level radioactive waste (LLW) shipped to commercial disposal sites and faces uncertain access to waste disposal sites and escalating waste management costs. The other small producers of LLW - industries, government (except the defense related research and production waste), academic and medical institutions that account for the remaining half of the commercial LLW - face the same storage, disposal and cost uncertainties. The disposal cost for LLW has escalated from approximately $5/ft3 in 1980 to over $300/ft3 today. Even the federal government (primarily DOE and DoD), which is by far the largest generator of radioactive and mixed wastes, faces new challenges in the areas of waste minimization, volume reduction, and waste stabilization. In a cost analysis report (1), Argonne National Laboratory has estimated that disposal costs for low level waste could range as high as $2,000/ft3 (all figures in 1994 dollars) depending on the disposal method. For hazardous waste the costs are specific to the contaminant. For example, for PCB's the costs can range from $950 to over $1,700/ton of incinerated PCB's in

a landfill operation. The mixed waste disposal cost can range to above $600/ ft3 .

For nuclear utilities, the disposal cost has become the primary driver for waste minimization and application of volume reduction technologies. During the last ten years, the volume of commercial LLW disposed of has decreased by two-thirds. In 1984, the total volume of waste disposed of was approximately 2,600,000 ft3 ; by 1993 it had declined to 790,000 ft3. However, the total activity of the waste disposed has tended to increase during that same period. The higher activity may be attributable to the increased age of several nuclear power plants and the need to replace equipment and components that have high activity levels.

The human health and safety criteria and the environmental protection standards have become more stringent. For radiation protection, the DOE uses a whole body dose of 100 mrem/y for the protection of an individual member of the public. Nuclear Regulatory Commission's 10 CFR 61 standard for the land disposal of low level radioactive waste uses a whole body dose criterion of 25 mrem/y. The Environmental Protection Agency's 10 CFR 191 specifies a criterion of 25 mrem/y whole body dose but it also specifies a groundwater protection criterion (from disposal activities) of 4 mrem/y. The MCL's specified for various hazardous contaminants are also stringent.

For mixed wastes, the regulatory issues have not been fully resolved and no facility in the country accepts mixed waste except the Envirocare site in Utah which currently accepts only certain types of mixed wastes where radioactivity is in the form of NORM or NARM materials.

The public apprehension about the risks of radioactive and hazardous wastes has meant longer and contentious siting processes for disposal facilities.

Above factors have led to a realization of the fact that treatment of wastes is a necessity for producers of radioactive and hazardous wastes. Above factors have also meant that innovative technologies for treatment of wastes are needed to save disposal costs and to produce environmentally stable waste forms that reduce the risks to human health and safety.

MAJOR ISSUES FOR THE UTILITIES

In a limited recent study, we looked at the waste treatment and management experiences at some representative utilities. These facilities included Duke Power Company's McGuire Plant, Boston Edison's Pilgrim Station, and Northern States Power's Monticello and Prairie Island Plants. The practices at these sites are generally representative of the nuclear utility industry. Most utilities do not find it cost-effective to invest in or operate major waste treatment systems for volume reduction or conditioning. Generally, utilities put more emphasis on waste source reduction through worker training, clean housekeeping, regular maintenance of active systems, and effective waste segregation. They may use limited waste processing on-site but for most part they use the services of contractors such as the Scientific Ecology Group Inc., (SEG) to volume reduce or condition their waste. The SEG has become the largest commercial waste treatment company in the country.

It has become clear that new methods are needed to process radiologically contaminated materials like resins, sludges, cartridge filters and EDTA solutions from nuclear power plants, and biological waste from hospitals and research establishments. Processing technologies such as dewatering and solidification do not volume reduce the resins and filters and do not

produce a waste form that can be safely stored for long period of time. Similarly, while incineration can reduce waste volumes significantly, the facilities employing incinerators were designed to handle activity <200 mR/h. The medium activity resins (1-10 R/h) can not be handled in such facilities. Resins are the second largest waste stream generated at nuclear power plants after the Dry Active Waste (DAW).

TECHNOLOGY DESCRIPTIONS

At the SEG facilities located in Oak Ridge,TN, a number of volume reduction technologies are being employed. For instance, the SEG ultracompactor has a force of ten million pounds and delivers the highest volume reduction factors possible for compactable waste even though the spring back characteristics of the waste will affect the ultimate volume reduction. In the past three years, SEG has identified three new technologies that will augment the existing technologies at its Central Volume Reduction Facility (CVRF) in Oak Ridge, TN. These new technologies are: Vitrification, Steam Reforming, and Quantum Catalytic Extraction Processing (Quantum- CEP). Most of the low level organic waste will continue to be processed via incineration. One of the three new technologies will be selectively applied to medium activity organic wastes or to the wastes that can not be incinerated. The targeted uses for these technologies are described below.

Vitrification: low level incinerator ash, glass, soil.
Steam Reforming: moderate activity waste, cartridge filters, sewage/sump sludges, EDTA/Decon solutions, hazardous waste, biological waste, paint/solvents, charcoal filters.
Quantum C.E.P. high activity waste, bead resins, powdered resins

A description of the new technologies and their application at the SEG facilities (2), along with a description of the incineration technology (because it is still one of the widely applicable technologies), follows.

Incineration

Incineration is the most economical and efficient technology for processing most of the low level organic materials like paper, plastic, wood, and rubber. It is also the most developed technology for a variety of uses and the designs on the market include: agitated hearth, controlled air, excess air cyclone, fluidized bed, rotary kiln and slagging pyrolysis, among others. Incineration provides a very high volume reduction, generally of the order of 100 : 1.

The first commercial incinerator for low level radioactive waste in the United States began operation in November 1989 at SEG facilities in Oak Ridge, TN. The system operates at temperatures up to 2,200oF which assures complete combustion of volatile materials. The incineration capacity of the system is 1,600 lbs/h. The emission control equipment consists of carefully chosen technologies consisting of a heat recovery boiler for off gas temperature control, a baghouse filter and a dual HEPA bank for particulate control, a wet scrubber for acid gas removal, and an exhaust reheat system for stack plume suppression. A second incinerator has been added at the CVRF in 1995. The combined incineration capacity at the CVRF is now approximately 14 million lbs per year.

Vitrification

As a subsidiary of Westinghouse, the SEG resources include the extensive experience of Westinghouse Nuclear Technology Group over the past 10 years. Vitrification experience includes design, fabrication, and operation of several different vitrification systems owned by SEG and for

other clients in U.S.A. and Japan. Vitrification systems are expensive to operate but they are well suited for processing of some organic wastes that are difficult to process. Contaminated soils and incinerator ash are prime candidate waste forms for vitrification. Since some generators are uncertain as to their ability to dispose of the incinerator ash, vitrification provides them the means to stabilize their waste into a final form that will be acceptable to all future disposal site operators. A 150 lb/h electric resistance heated vitrification system was installed at the CVRF in late 1994. This system is operational and is capable of vitrifying all the incinerator ash that results from the two large incinerators at the CVRF. Thus, the capacity for vitrification of 14 million lbs of DAW per year is in place at the facility. Sequential application of technologies - incineration followed by vitrification, makes it cost-effective for the processing of low level radioactive waste.

Steam Reforming

Steam Reforming is often called "Steam Detoxification" as it detoxifies or renders pathological or hazardous materials nonhazardous. Steam Reforming is a new thermal process where organic materials are decomposed. The process was originally intended for pathological wastes and hazardous wastes. A Steam Reforming system is generally a small closed loop system which produces a final waste form similar to incinerator ash. However, the processing methodology used is very different and the system is not classified by EPA as an incinerator. Unlike incinerators, Steam Reforming systems do not employ combustion in an oxygen atmosphere. Instead, the system employs steam reforming chemistry (at 600-2500oF), which volatilizes and reforms organics into CO, H2, CO2, H2O, and CH4.

Waste is charged into the system in a variety of ways, depending on the waste form. The aromatic organic wastes can be loaded into the system by loading the waste drum into the system autoclave, the Drum Feed Evaporator (DFE) unit. Higher throughput efficiency can be obtained when solvent or liquid organics deposited on solid materials are shredded and fed into the Waste Feed Evaporator (WFE). Liquid wastes can be flash atomized into the reaction vessel, which is generally a steel drum, inside the autoclave. Small containers, such as paint cans can simply be opened and placed on special rack inserts in the autoclave.

On startup, syngas and excess steam are injected into the evaporator to vaporize organic constituents which exhaust to a detoxification reactor in the main processing unit. At this location, the waste is mixed with excess steam and electrically heated to high temperature to complete the organic molecule destruction. The high temperature reforms the molecules into CO, H2, CO2, and H2O. This syngas then flows through a regenerative closed loop where various parameters of destruction efficiency are also monitored. The loop also contains a regenerative heat exchange train to allow cooling of the gas before it enters an absorber for halogen removal. After it exits the absorber, the gas is reheated and it is either recycled to the WFE or slipstream discharged to the off gas system. The discharge removes excess gas (water vapor and CO) and the radioactive gaseous and particulate activity through the HEPA filters. It is also possible to employ an optional catalytic converter upstream of the HEPA filters to oxidize CO to CO2.

The high temperature exothermic reaction destroys the hazardous and pathological wastes. Organic material is reduced to a small volume of ash

which is retained in the drum used to charge the waste into the system. This drum can later be supercompacted. Typical volume reduction as high as 100 : 1 can be obtained for some organic materials.

The closed loop system releases only water vapor and CO2; hence off gas treatment is generally not required for most type of organic wastes. Radioactive particles are collected in the HEPA filters. The Steam Detoxifier, the main processing unit, incorporates a high temperature reactor, heat exchanger, steam boiler, adsorber bed, HEPA filters, computerized process controller and extensive self monitoring equipment. System size is small, generally requiring less than 1000 ft3 of operating space. The remote operation and shielding of the system can be easily accommodated, thus permitting operation with higher activity waste. Since waste can be charged into the system in a closed drum and removed from the system after processing, still inside the closed drum, airborne contamination risks are minimized.

The SEG has a process which is licensed from the patent holder of the technology, Synthetica Technologies, Inc., of Richmond, CA. An extensive proof-of-concept testing on a variety of organic materials was conducted over the past year. The Synthetica Steam Detoxifier was found to be very effective in volume reducing many types of organic wastes including cartridge filters, sludges, EDTA solutions, paints and solvents, and other aqueous waste. It was also very effective in regenerating charcoal for reuse. Two steam reforming units are currently in place at the CVRF.

Quantum Catalytic Extraction Processing

Catalytic Extraction Processing (CEP) technology employs a bath of molten metal to render waste to its smallest elemental form. The process was developed (and patented) by Molten Metal Technologies Inc., (MMT) of Waltham, MA. Over 250 million dollars have been spent in the testing and refinement of this technology over the past five years to bring it to commercial markets. Even though it was originally intended for processing/destruction of hazardous and toxic wastes, it has emerged as an innovative technology of high potential for the processing of all types of wastes including radioactive and mixed wastes.

In 1993, a bench-scale test system was jointly constructed at CVRF by SEG and MMT. The purpose was to investigate the effectiveness of this new technology in processing radioactive waste, mixed waste, and specifically, the high activity organic resins. The test results led to a conclusion that a variation of the CEP, which incorporated radionuclide partitioning was the best choice for processing high activity organic resins. This new technique for processing radioactive materials is called Quantum-CEP.

Quantum-CEP molten metal bath operates at temperatures greater than 3,000oF. This allows significantly higher energy transfer to the waste material than the other thermal processing techniques such as incineration or vitrification. The process is coupled to a Catalytic Processing Unit (CPU) to destroy organic components of resins and control the partitioning of metallic ions (radionuclides) and chemical elements into three separate phases - gas, ceramic matrix, metal. This process effectively controls the deposition of many radionuclides, such as Cs-137, and chemical elements such as sulphur into preferred locations.

The resins are dried and injected into the CPU's molten metal bath which is contained in a disposable ceramic lined crucible. The metal bath, 1-2 ft3 in size, and operating at 3,500oF, instantly vaporizes organic resin and amalgams all metallic ions captured on resin into the molten metal.

The destruction process permits very high volume reductions of the complex organic materials placed in the molten metal bath as most of the material is converted into CO, H2, CO2, and H2O. For resins, volume reduction of 30:1 may be feasible, which is much higher than that obtained by the vitrification process. Quantum-CEP also produces a solid metal waste form which has superior strength and shielding characteristics as compared to the glass waste form produced by vitrification.

A 15 million dollar joint-venture (SEG and MMT) Quantum-CEP facility, designed to process 80,000 ft3 of resins each year is nearing operational stage in Oak Ridge, TN. This shielded facility will be able to process resins up to 10 R/h. These high activity resins are transported in High Integrity containers (HICs). Because of the high dose rate, all resin transfers and waste processing must be performed via remote handling equipment.

CONCLUSIONS

Three innovative technologies discussed in this paper can provide cost effective waste processing techniques for radioactive, hazardous, and mixed wastes. All three are commercially viable and available for application. For medium activity wastes, such as the resins from nuclear power plants, Quantum-CEP technology can provide a cost-effective processing technique, while also producing an environmentally stable final waste form. It is expected that it will permit processing and disposal of the resins at a cost significantly less than the current Barnwell disposal charges of $500/ft3 for medium activity wastes.

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LUNCHEON SPEAKER

Roy G. Post, WM Symposia, Inc.

L-1

1946-1996 FIFTY YEARS OF "CIVILIAN CONTROL"

Presented at WM'96 Lunch

Roy G. Post
WM Symposia, Inc.

How many of you have been a part of the nuclear community for ten years?, 20? 30? 40? 50? Longer? I started in the West Stands at the University of Chicago 52 years ago this past January. Let's jump back to the birthing of the nuclear age begining with the first controlled nuclear chain reaction in December 1942. This was just 13 months prior to my introduction to the project.

December 1942 - the first controlled chain reaction, power in milliwatts - graphite moderated, no cooling. It became known as CP 1, Critical Pile 1. Security concerns were extreme:

An example: Fermi and the other nuclear scientists knew that only graphite and heavy water were sufficiently efficient as neutron moderators to permit a chain reaction with natural uranium. The best graphite we could buy absorbed too many neutrons. We couldn't tell the vendor what the contaminants were, just that greater purity was required. Even though the scientists did know the offending elements, they couldn't

tell the vendor because it could reveal that there were concerns about neutron absorbers, which in turn could mean we were working with something nuclear.
From the auspicious beginning of that first pile, these milestones, the foundation of the nuclear age was formed in only 30 months.
Development of nuclear reactor theory - including the four factor formula.
The scientists measured critical nuclear parameters - rumor had it that the Germans had made a serious error in measuring the diffusion length of neutrons in graphite. If this were true, it could have led them down the wrong path.
Critical Pile 1 (CP 1) was designed and built. Data were collected in order to
Design and build the Oak Ridge gas cooled graphite moderated reactor. This reactor was operated to make plutonium in gram quantities.
The plutonium was used to develop the chemical process to separate, purify and reduce plutonium.
Metallic plutonium was made so we could measure its chemical and physical properties. When our measurements couldn't be confirmed we would submit that plutonium was such a new element it hadn't settled its properties.
With the data from the operating reactors, Dupont engineers designed and built the Hanford reactor systems. These were chemical and mechanical engineers with no nuclear background or experience. Obviously, no one had any nuclear experience we had to learn as we went. We enjoyed great communication between the engineers and the scientists. My first discussion of my work was to an audience of about thirty which included three Nobel Laurets,
you guys today are a pushover.
The Hanford reactors generated a million kilowatts each with fuel carrying heat loads that would melt them in seconds without water cooling.
Not only the four reactors, but fuel manufacturing plants and fuel processing plantswere built, tested and operated long enough to make, separate, purify and reduce kilograms of plutonium.
Ten to twelve major steps were begun and completed between the end of 1942 and the middle of 1945, only two and a half years from the first chain reaction.
Of course, we were highly focussed on the military job at hand. But we would, in lighter moments, let our imaginations dwell on the future, after the war when we would have the opportunities to develop this wonderful and powerful new nuclear energy. Soon, very soon, the war was over due in large part to the product of our efforts. Now, we thought, we could consider other problems and opportunities.
1946, what a year that was. Those of us working on the project were becoming more and more excited about the opportunities and the work to be done. We had a vision and a mission but we had no means to support our efforts. We needed a unique political action which would provide for the safe commercial development of peaceful benefits from nuclear energy. We could easily see ways to generate electrical energy. We knew how to use isotopes to measure processes. We knew that radiation could be used to enhance the properties of plastics, glasses and gems. We believed that we could pinpoint radiation sources in the body for diagnosis and treatment of a host of human and animal ailments..

1946, we were ready to move on, some of us returned to the Universities to teach or to pursue graduate studies. Many would rejoin the crusade later. Most of us we wanted to continue and explore the opportunities. It was a time of great expectations, unbridled enthusiasm and idealistic hopes. It was a year of great events for me too, Becky and I tied the knot and Ruth Jean was born into our little family.

I would like to share some of the euphoria and attitudes at that time, as I remember them. My viewpoint is that of a twenty three year old chemical engineer with only two years of research experience. After four years of war when almost all of our thoughts were concentrated on "winning the war", the prevailing mood, after the great victory, was to remove the war time controls so we could get on with our lives in peace.

There were three dominant concerns in the newly formed nuclear community to which I belonged.

First the benefits from the nuclear technology we were developing included a source of huge amounts of energy that would have immense potential benefit to the war-ravaged world. We also recognized that there were many uses for radioisotopes to be developed.

Second, with our years of indoctrination, we were convinced that the dangers of excessive exposure to radiation and spread of radioisotopes were real. Nonetheless, we also knew that the methods of detection were so sensitive and the biological effects so well understood that the public health could be well protected. We were concerned that the implementation of the necessary regulations were not in place. Proper radiation protection procedures would be required for all nuclear applications.

Third, we were concerned that proper legislation would be written to move control from the military to the civilian and to develop the great potential.

Let us reflect on our hopes for the legislation by considering the benefits, health physics concerns in detail and our standing with the political and social world in 1945 and 1946.

BENEFITS

As nuclear engineers, we were excited about participating in the development of a technology so powerful that it would change our world in ways we could not even imagine. In order to understand and explore the development of nuclear energy for civilian uses, instead of limiting it to weapons, we held meetings and had extended discussions of the possibilities. During this time almost all of the reactor types were conceived. These included the gas cooled, graphite moderated reactor, the pressurized water reactor, the boiling water reactor, the sodium cooled fast reactor and others that didn't make it, such as the molten salt reactor. Almost every combination of moderator, fuel and coolant was examined. The potential uses of isotopes in medicine, well logging, agriculture, plastics and other industrial processes were proposed and analyzed.

We felt secure and the future was bright. The US not only had the "bomb" but the only industrial complex untouched by the war. It seemed that the endless opportunities dwarfed the problems overall, and even more so for the budding nuclear community.

RADIATION CONCERNS

Before 1942, a few curies of radium were available. Now, the Hanford reactors were producing megacuries of radioisotopes. We were concerned that our proven methods of working safely with these dangers should be

followed. Data on the biological effects of radiation were available from those using x-rays, accelerators and radium. The research programs to develop designs for the Hanford reactors, the processes to separate plutonium and produce the metal had been paralleled with research programs studying the biochemistry of radioisotopes and their effects on living systems.

In the West Stands at the University of Chicago, goats were injected with various radioisotopes. Sometimes they would escape from their cages and come up the stairs to visit us while we were running solvent extraction columns on the graveyard shift. Seeing a goat, known to be radioactive, climbing the stairs to our operating position was an awakening experience. Nonetheless, it was proof of the active research program in the biological effects of radiation. Confinement of the goats was soon perfected.

At Oak Ridge and at Hanford, the research with laboratory animals, aquatic and terrestrial plants, and the soil was well financed and was carried out by the best scientists in the world.

The effects of radiation were correlated with the best estimates of the doses and dose rates from the time of the discovery of x-rays and radium to the beginning of the project. Based on these data, limits were set to assure that all those working with radiation received less than one tenth of the dose producing clinically observable effects.

You may not know that when the medical use of x-rays became widespread, but before the dangers of high exposures were recognized, you could identify a pediatrician by the radiation damage to his hands and arms such as scarring and loss of fingers from holding infants in x-ray beams. The lessons were learned so that now, fifty years later, no clinically observable health effects have been found at or below the levels established in the early forties.

From the very beginning, research programs, safety procedures and extensive biological testing of those of us working with high levels of radioisotopes was a standard daily routine.

Back in the early 40's, there was the almost paranoic concern at Stagg Field and throughout the project that the nuclear work might be discovered. In spite of that, the concern for the public safety outweighed the security concerns and when the rods were pulled for that first chain reaction, a technician was stationed outside of Stagg Field's West Stands to measure radiation levels and protect anyone that might be passing on the sidewalk.

One of the first actions the new nuclear community took which showed their public concern was to eliminate the use of x-rays to fit shoes. I was not aware of any widespread health effects from the foot x-rays but we knew that they were many times higher than we would permit in our work. We also promoted the monitoring and regular calibration of medical and industrial x-ray machines.

We recognized and understood the hazards of nuclear radiation. We had developed the knowledge to control it and had assumed the responsibility to do so.

LEGISLATION

Nonetheless, we were determined to do all we could to avoid irresponsible exploitation of nuclear energy that could result in wide spread radioactive contamination. We also believed that every effort should be extended to prevent the proliferation of nuclear weapons. The scientists and engineers engaged in this work agreed that legislation was needed to

assure the public health, to promote the development of this technology and to keep it from those that would miss use it. Looking back, I don't know how much we influenced the Congress at that time, I expect more than any of us here do now, but the results were what we thought they should be. Looking back from today's perspective, we might want things to be somewhat different.

The first attempt at enabling legislation was the May-Johnson bill which failed because of concern that the military would retain too much control. Senator Brien McMahon ofConnecticut introduced a compromise bill which established the United States Atomic Energy Commission with five civilian members, a general advisory committee and a military liaison committee. This, the most important single piece legislation in the development and use of nuclear energy, was enacted into law August 1, 1946. This act recognized and confirmed the responsibility of the United States not only to develop nuclear energy for the general benefit but to regulate its use to protect the health of all the people. Thus began many decades of cooperation among the Congress, through the Joint Committee on Atomic Energy, the Atomic Energy Commission and the people of the United States.

There were some interesting aspects in the law many which have now been changed. For example, mining of uranium was excluded from the AEC control, hence there was no federal responsibility for the control of mill tailings. There have been modifications of this law and its basic structure was discarded in 1974.

The McMahon Act emphasized the concerns of the nuclear community for the public health. Standards to limit exposure were extremely conservative for those working in the field and they were reduced by a factor of ten for the general public. This may have been a disservice, in that even these levels were much lower than those required for safety, and now, they have been reduced even further. Often the reduction is made only because the lower levels were readily measured and procedures could be easily developed and used without high costs. No one anticipated the costs of lost electrical production and costs from extending construction schedules to implement lower limits and the hearings held in response to public reactions to a perceived need to meet such low levels.

From the end of WW II until Rachel Carson's "The Silent Spring", which did not address nuclear radiation, nuclear operations were considered to be well within politically and socially acceptable standards or values. Almost all of the public recognized the benefits of nuclear energy and had confidence in the competence and responsibility of the nuclear community. It's a bitter irony is that this was lost, not through failure to live up to promised performance, but to a revolutionary change in political views, social values and expectations.

After these fifty years since the McMahon Act became law, I have seen nothing that would change my belief that nuclear technologies still offer the best promise for the energy needed to bring all nations to acceptable levels of living. Since nuclear reactors generate no carbon dioxide, sulfur dioxide or nitrogen oxides, displacing fossil fuel electrical generating plants will actually decrease carbon dioxide, sulfur and other pollutant emissions and improve the atmosphere we breath. More available electrical energy, produced by nuclear reactors can promote the economies of many of the less privileged peoples and begin to stabilize the relations among nations. People that are well fed and secure don't start wars and breed terrorism.

Session 07 -- POSTER -- HLW
Co-chairs: Werner Lutze, UNM
Ted McIntosh, USDOE
7-1

FIELD AND LABORATORY TESTING OF SEAL MATERIALS PROPOSED FOR THE WASTE ISOLATION PILOT PLANT

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ABSTRACT

The Small Scale Seal Performance Tests (SSSPT) were a series of in situ tests designed to evaluate the feasibility of various materials for sealing purposes. Testing was initiated in 1985 and concluded in 1995. Materials selected for the SSSPT included salt-saturated concrete, a 50%/50% mixture of crushed salt and bentonite, bentonite, and crushed salt. This paper presents a summary of the SSSPT field program, results of the in situ testing, and a discussion of post-testing laboratory studies of salt-saturated concrete. Results of the SSSPT support the use of salt-saturated concrete, compacted bentonite clay, and compacted crushed salt as sealing materials for the WIPP.

BACKGROUND

The Waste Isolation Pilot Plant (WIPP), located in southeastern New Mexico, is being developed by the U. S. Department of Energy (DOE) as a potential disposal site for transuranic wastes. The disposal site is located approximately 650 meters below ground surface and consists of a repository mined in a bedded salt deposit. Disposal of wastes in salt is considered advantageous because over time the salt will creep, closing around the waste and thereby encapsulating and isolating hazardous constituents from the accessible environment. Prior to certification as a disposal site and initiation of disposal activities, the DOE must demonstrate that the WIPP will meet all regulatory requirements. As part of this demonstration, Sandia National Laboratories (SNL) is working with the DOE to develop a sealing system design for the four shafts that provide access to the repository.

This paper discusses the Small Scale Seal Performance Tests (SSSPTs), a series of in situ tests designed by SNL and conducted by SNL and Westinghouse Electric Corporation, Waste Isolation Division to evaluate the feasibility of various materials for sealing purposes. The discussion is focused on the results of the field program, with a brief summary of the post-test laboratory analysis of concrete seal materials. The positive results of the SSSPTs support the inclusion of salt-saturated concrete, compacted crushed salt, and compacted bentonite clay in the current WIPP shaft sealing system design.

The system design (1) is based on the use of effective seal materials and the reduction of uncertainty through functional redundancy. Seal components must maintain structural integrity and act as barriers to fluid flow throughout their design life. Redundancy is obtained through the use of multiple components and materials. Salt-saturated concrete,

asphalt, and compacted bentonite clay components will serve as barriers to fluid flow for several hundred years following shaft seal construction, allowing time for the primary long-term crushed salt seal to become fully effective. Crushed salt and compacted bentonite clay materials comprise the long-term components for the shaft seal system. These components are expected to remain functional for thousands of years. A conceptualization of the proposed WIPP shaft sealing system design is depicted in Fig. 1.

Fig. 1

SMALL SCALE SEAL PERFORMANCE TESTS: PROGRAM GOALS AND DESCRIPTION

To assess the performance of candidate seal materials in situ, the SSSPTs were initiated several years before the current WIPP sealing system design. Five series of tests were conducted on seals up to 1m in diameter. Essential SSSPT goals were to 1) use seal emplacement technologies that are compatible with standard industrial practices, 2) demonstrate reduction in fluid flows, and 3) verify that seal components remain structurally sound. Test results were used to evaluate materials in terms of the sealing program goals. The small scale of the field tests (hence the name) allowed performance of numerous tests under a range of conditions and configurations.

A series of five SSSPT tests (A-D and F) was initiated in 1985 under the guidance of the original Test Plan. Series F consisted of a grating demonstration and is not discussed in this paper. (The fourth test was later expanded into two phases.) Details on the development, construction, and proposed testing sequence for each series were documented as addenda to the original Test Plan. Each test series evaluated a different seal material and/or seal configuration. Seal materials were emplaced in either horizontal or vertical boreholes at the repository horizon. Figure 2 presents a schematic of a typical SSSPT for the emplacement borehole, seal structure, and fluid flow test apparatus configuration. Many of the emplaced seals were equipped with internal stress and strain gages.

Fig. 2

Table I lists the seal material, configurations, and construction dates of SSSPT Series A-D. A brief description of the seal materials is presented in Table II.

Table I

Table II

The SSSPT field program verified that the candidate materials could be emplaced using standard industrial equipment and practices. The in situ fluid flow and structural behavior of the seal materials were also assessed. In addition to the field test program, samples from a salt-saturated concrete seal (emplaced in 1985) and the surrounding host rock were extracted from the field configuration for laboratory analysis in 1994. The analysis included visual observation of the seal/rock interface, quantitative evaluation of the gas permeability of the seal material, and petrographic analysis of the seal material and the surrounding host rock.

EMPLACEMENT TECHNIQUES

A primary goal of the SSSPT program was to use or adapt existing technology to the construction of repository seals. Methods used to construct seals for SSSPT Series A-D were all direct applications of widely used construction practices. Emplacement configurations for each series are summarized.

Test Series A consisted of concrete seals emplaced in vertical-down boreholes. Salt-saturated concrete was emplaced in selected seal intervals using gravity flow. No forms were used. In some cases, vibration was used to promote even concrete distribution around structural performance gages (2).

Series B seals were also comprised of concrete, but were emplaced to plug horizontal boreholes. Construction of these seals employed a small pump to fill pre-formed intervals with concrete. Vibration was not applied (3).

Series C seals consisted of precompacted salt and salt/bentonite block structures that were constructed to plug horizontal boreholes. One-meter intervals of the round boreholes were enlarged to create square chambers into which the blocks were emplaced. The blocks were manufactured using a modified pressed-earth adobe block machine and emplaced to form a seal structure with minimal void spaces. No mortar was used between blocks (4).

Series D seals were also constructed of precompacted salt, salt/bentonite, and pure bentonite blocks, but the seal structures were built to plug vertical-down boreholes. Phase 1 seals consisted of salt blocks. Phase 2 seals consisted of a 100% bentonite core with layers of 50%/50% crushed salt/bentonite and/or 100% crushed salt above and below the core. Emplacement techniques were essentially the same as for Series C except that blocks were trimmed to conform to the cylindrical geometry. Portions of blocks were excavated as necessary to accommodate gages placed for monitoring structural performance (5). In some cases, crushed salt was compacted directly into the seal interval using a hand-operated tamper to achieve densities approximating those achieved using precompacted blocks.

FLUID FLOW PERFORMANCE

Seal system permeabilities for the SSSPTs, derived from a series of in situ fluid flow measurements, are presented in Table III. The concrete seals have been subjected to the most extensive fluid testing program. All seals were either gas- or brine-flow tested immediately following seal construction in 1985 to 1987, and a selected subset was retested from 1993 to mid-1995. Fluid injection pressures of up to 2.3 MPa were used in the field testing program. Tracer-gas testing, constant-pressure gas- and brine-flow testing, and constant-volume brine-flow testing were conducted on the concrete seal systems. The seal system consists of the seal material, the seal/host rock interface, the zone of rock immediately surrounding the seal, and the far-field host rock. These tests do not provide information regarding the flow paths of the test fluid. Therefore, the calculated system permeability represents a composite fluid barrier presented by the seal system. From an engineering perspective, these tests provide an excellent means for evaluating the entire seal system.

Table III

Seal system permeabilities for the concrete SSSPT were derived through numerical computer simulations of brine- and gas-flow behavior. Fluid injection was simulated for a conceptualized model of the seal system. Estimated ranges for the formation and seal permeabilities, formation storativity, and seal porosity were used to generate pressure-decay and mass-flow-rate curves. The generated curves were then compared to the field data. Hundreds of computer simulations were used in the analysis, and a best fit to the field data was obtained using an optimization

routine. This technique produced a best estimate of the permeability for each seal system. Details of the analysis tool and conceptual model may be found in Pickens et al. (6) and Beauheim et al. (7).

To improve confidence in the data interpretation, additional testing was conducted to characterize the actual flow paths. Several boreholes were drilled in the formation surrounding a Series A seal. Gas flow testing was conducted in these boreholes to assess the extent and permeability of the disturbed rock zone in the test area. The permeability of the disturbed zone was found to be approximately 10-20 m2, which was consistent with the permeability of the complete seal system.

Brine flow testing was conducted on 50%/50% crushed salt/bentonite seals. A high rate of brine injection led to failure of the first seal, so the brine injection rate was reduced for the second and third test sequences. An average pressure differential of less than 0.013 MPa was maintained throughout the testing period. The steady-state flow rate of brine into the test interval was monitored for approximately 800 days. Brine that seeped through the front of the seal was recirculated during this period (8). Assuming that all flow occurred through the seal and that one-dimensional Darcy flow approximations are valid, the permeability of the 50%/50% salt/bentonite seal material was derived from Darcy's Law. Attempts to conduct gas flow testing were unsuccessful because of a small separation at the seal/host rock interface. The brine permeability of the 50%/50% salt/bentonite seal material was two orders of magnitude higher than laboratory values of similar materials.

The crushed salt seals constructed for Series D, Phase 1 were not expected to present a barrier to flow until the salt had reconsolidated to at least 90% of the density of intact halite. At the time of test termination in mid-1995, the seal material was determined to have a porosity of at least 12% (i.e., a fractional density of 88% of the density of intact halite compared to 82% at the start of the test). Attempts to develop gas pressure in the seal test interval confirmed that the permeability of the crushed salt was too high to constitute an effective fluid barrier. This finding was consistent with the expected behavior of reconsolidating crushed salt.

Brine injection testing was conducted on the two vertically emplaced bentonite core seals. Testing was initiated in 1990 for one seal and in 1994 for the second seal. Throughout the testing period, the pressure differentials across the seals were maintained at approximately 0.72 and 0.32 MPa, respectively. During the test period, no brine was observed at the top of either seal. A conservative estimate for the seal system permeability was made by assuming that 1) the seal material was saturated; 2) test conditions were at steady state; 3) all flow was into the seal material; and 4) formation brine contributions were negligible. The permeability was then derived from Darcy's Law in a manner similar to that used for the 50%/50% crushed salt/bentonite seals, and is listed in Table III.

Gas flow testing was conducted on one of the bentonite core seals to evaluate the gas threshold pressure of the seal material. The test interval was pressurized in 0.67-MPa intervals. After each increase in pressure, the interval was shut in to allow the system to come to equilibrium. As shown in Fig. 3, the seal exhibited negligible gas flow until the test interval pressure exceeded 4 MPa. This test was conducted to provide qualitative information on the behavior of bentonite seals subjected to high gas pressure.

Fig. 3.

STRUCTURAL PERFORMANCE

Many of the SSSPTs were equipped with internal stress and strain gages. Measurements taken from these gages were used to monitor the seal structural performance. Structural failure of a seal material would be detected through seal stress- and strain-state monitoring, fluid flow measurements, and visual observation. The creep behavior of the host rock would also influence the structural response of a seal material. Quantification of the salt creep was derived from borehole closure measurements, which were made in both open and closed boreholes. Gage type, orientation, and location are specified in the applicable test plan addendum. A representative sampling of the SSSPT gage data is presented here.

The structural response of the concrete seals can be evaluated from inspection of radial and circumferential strain (Fig. 4 and 5), radial stress (Fig. 6), and borehole closure measurements (Fig. 7). The general strain gage trend depicts a compressive strain rate that was relatively rapid after seal emplacement but that monotonically decayed with time and continued to increase, but at ever slower rates. This behavior reflects the concrete strains under the compressive loading applied by the surrounding salt. The stress gage data showed that seal compressive stresses rose rapidly immediately following seal construction, attaining steady-state after 100 to 200 days. Creep closure of the borehole was inhibited by the presence of the concrete seal, as illustrated in Fig. 7.

Fig. 4

Fig. 5

Fig. 6

Fig. 7

Visual observations of the concrete seals revealed no spalling or other external evidence of structural degradation. The relatively smooth trend of stress and strain gage data, coupled with the results of fluid flow testing, visual observations, and laboratory analysis (presented in the next section) corroborate the conclusion that the salt-saturated concrete maintained structural integrity throughout the testing period. The unconfined compressive strength of the salt-saturated concrete exceeds predicted formation in situ stresses by more than a factor of three (9). The constitutive model for reconsolidation of crushed salt (10) predicts very little resistance to porosity reduction for porosities greater than 10%. Borehole closure measurements from the 100% crushed salt seals (Fig. 8) support this prediction. These seals were emplaced with an initial porosity of approximately 18%. Seven years later, closure measurements indicate that the porosity may have been reduced to approximately 12%. These internal closure measurements differ only slightly from closure in an empty borehole. Measures of stresses internal to these seals are nominally between 0.1 and 0.2 MPa, as shown in Fig. 9. This finding is also consistent with model predictions that the crushed salt will offer little resistance to closure when the material porosity is greater than 10%.

Fig. 8

Fig. 9

Nominal internal stresses for the 100% bentonite core seals are shown in Figs. 10 and 11. The increasing trend shown in Fig. 10 could be the result of compression resulting from borehole closure, bentonite swell pressure, or a combination of the two processes. The reduction in stress

at approximately 1400 days, shown in Fig. 11, may be related to ion-exchange with permeant brines in the bentonite fabric. These issues are currently being addressed by a separate laboratory program.

Fig. 10

Fig. 11

Structural performance data (as derived from stress, strain, and borehole closure data) were consistent with current understanding and expectations of the respective seal materials. To the extent necessary for demonstrating sealing design adequacy, these data will be incorporated into the material models used to evaluate the shaft sealing design.

LABORATORY TESTING OF SSSPT CONCRETE

In late 1994, approximately nine years following construction of the vertically-emplaced concrete seals, a "post-mortem" analysis of the sealing system was conducted. Cores of intact host rock and samples of the concrete seal material and seal/host rock interface were extracted from the test region. Petrographic studies of the phase assemblages of the concrete seal material and the microstructure of the intact rock were conducted. Gas permeability testing was conducted on concrete core samples.

Preliminary results of the laboratory tests indicate that, although the concrete seal was exposed to native brines during the testing period, the material experienced no deterioration in performance capabilities. Phase assemblage analysis of the concrete showed minimal degradation of the seal material. This result is consistent with a more extensive analysis of concrete specimens extracted from the WIPP underground in 1991 (11). The gas permeability of the concrete core samples was found to be approximately 10-20 m2, which is consistent with the field measurements. The seal/host rock interface was visually inspected for signs of brine migration and deterioration along the interface zone. No signs of degradation or separation were found. In addition to visual inspection of the interface, field observations of the retrieval were recorded. During extraction of the seal/host rock interface material, breakage occurred preferentially through the seal material rather than along the interface. This observation was consistent with direct shear testing of concrete specimens extracted from the WIPP underground in 1991 (12).

Microstructural analysis of the intact rock showed dilation along the salt grain boundaries for specimens taken from the immediate vicinity of an open borehole. Specimens taken in the immediate vicinity of the concrete seal, as well as those from a far-field location, showed no dilation. The calculated gas permeability for these regions also showed that, in the immediate vicinity of the concrete seal, a disturbed zone did not exist. The absence of a disturbed zone in this vicinity indicates that either the seal prevented the formation of a disturbed zone or that, through the process of creep closure, any disturbed rock zone that had formed around the seal was subsequently eliminated. The post-mortem testing sequence provided additional confidence in the ability of salt-saturated concrete to retain fluid flow and structural performance capabilities in WIPP salt.

SUMMARY AND CONCLUSIONS

The goals of the SSSPT series to use standard industrial practices for seal emplacement and to evaluate seal materials for their ability to retard fluid migration and remain structurally sound were met. No new technologies were developed for the emplacement of the SSSPTs. Fluid flow testing of the sealing systems provided strong evidence that both salt-

saturated concrete and bentonite would function effectively as barriers to fluid flow. The testing also supported model predictions of the behavior of a crushed salt seal and resulted in elimination of 50%/50% crushed salt/bentonite as a viable seal material. Stress, strain, and borehole closure data, when coupled to visual observations and fluid flow test results, demonstrated that all seal materials maintained structural integrity throughout the testing period. A post-mortem analysis of the concrete sealing system provided additional confidence in the ability of this material to function effectively at the WIPP horizon.

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EFFECTS OF A HYPOTHETICAL WATER INGRESS INTO A SALT DOME REPOSITORY ON THE ALUMINUM CLADDING OF DIRECTLY STORED MATERIAL

TEST REACTOR FUEL ELEMENTS

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ABSTRACT

The direct final disposal of spent material test reactor fuel elements (MTR-FE) is being discussed in the Federal Republic of Germany as an alternative to reprocessing (1). A possible repository under consideration is a salt mine and the accident scenario for long-term safety analysis is a hypothetical water ingress.

A water ingress involves the formation of highly concentrated brines which first come into contact with the storage casks. The cask design ensures a lifetime of 500 years in Q brine.

After this period, the brine may penetrate into the cask and act on the MTR-FE so that a radionuclide release would be conceivable.

It is therefore of interest to know whether the MTR fuel element cladding materials (high-purity aluminum, AlMg1 and AlMg2) have a barrier function with respect to radionuclide release when exposed to highly concentrated brines. Relevant corrosion experiments have been carried out at the Institute for Safety Research and Reactor Technology of KFA Jlich (2). In a first test series, the above materials were exposed to differently concentrated repository-relevant brines. Interesting correlations were found, in particular, by varying the test temperature and adding different metallic alloys. The results were incorporated into mass loss calculations in order to obtain a first picture of the corrosion behavior involved.

Furthermore, electrochemical studies were carried out providing an insight into the mechanical processes of corrosion.

On the basis of these studies, it may be concluded even at this stage that the cladding materials of high-purity aluminum, AlMg1 and AlMg2 do not exercise a barrier function for radionuclide release. Further investigations on the release of radionuclides from spent fuel element sections in brines will serve to determine the source term for release calculations.

INTRODUCTION

After a decision has been made against the construction of a German reprocessing plant, the direct final disposal of spent fuel elements is being discussed in the Federal Republic of Germany as an alternative to reprocessing abroad. In order to protect man and nature from the harmful effects of radioactive waste, final disposal is to be effected in underground stable geological formations ensuring safe containment of the radionuclides involved. Possible repository formations are salt mines in northern Germany (Wendland), which have remained almost unchanged since their formation approx. 200 million years ago because they are isolated from water-bearing strata. Safety analyses for such repository formations have already been carried out for the emplacement of heat-generating waste from reprocessing, and they demonstrated the basic suitability of a salt mine for final disposal.In these long-term analyses, a water ingress into the repository was assumed as a hypothetical accident. Since spent fuel elements greatly differ in their chemical and physical properties from vitrified reprocessing waste, the effect of such an accident on the fuel elements must be examined. These investigations are being carried out at the Research Centre Karlsruhe for fuel elements from UO2 pellets

and at the Institute for Safety Research and Reactor Technology of KFA Jlich for fuel elements from research and high-temperature reactors. The experiments on the long-term safety of HTR-FE are nearly completed.

The problems of a direct final storage of spent fuel elements from material test reactors (MTR) have been dealt with since 1993. These fuel elements consist of a metallic uranium-aluminum alloy as the fuel kernel enveloped in aluminum. An ingress of water leads to the formation of highly concentrated extremely corrosive brines. The storage casks of the "Pollux" type are designed for a lifetime of 500 years in such brines. After this time, the brine may penetrate into the cask and act on the MTR fuel elements so that radionuclide release would be conceivable in the salt mine. Since the fuel kernel is enclosed in an aluminum cladding, the question arises whether this cladding has a barrier function with respect to release. In a first investigation series, the corrosion of aluminum was therefore studied under repository-relevant conditions.

EXPERIMENTAL

The chemicals used in the present study were of analytical-grade quality. The following equipment was used for the determination of mass loss:

drying cupboard from Heraeus

ultrasonic cleaning device from Bandelin GmbH & Co, Berlin

The following equipment was used for electrochemical work:

recorder "DIGICASS 600" from Laumann, Selb

potentiostat, Wenking model LB 81 M, from Intelligent Controls GmbH, Clausthal-Zellerfeld

voltage scanner, Wenking MVS 87, from Intelligent Controls GmbH

DC measuring amplifier, type 72 WC, from Knick.

The relevant MTR-FE cladding materials used were high-purity aluminum (Al 99.5; material no.3.0255 from Honsel AG) as well as the alloys AlMg1 (material no. 3.3315.10 from Honsel AG) and AlMg2 (material no. 3.3325.10 from Honsel AG). After pickling with nitric acid and subsequent neutralization, these materials had the same surface as the cladding material of the MTR fuel elements. Test strips of the dimensions 20 mm*40mm*2.4mm were used to determine the mass loss due to corrosion. All material samples were cleaned with acetone prior to testing and then exposed to the respective brines in laboratory flasks using different test parameters. At fixed intervals, the samples were withdrawn, freed from adhering corrosion products, dried, weighed and exposed again to the brine. Corrosion products were removed with 5% nitric acid in an ultrasonic bath.

High-purity aluminum (Al 99.5) was used for the electrochemical investigations. The metal test strip was screwed into a specifically manufactured electrochemical measuring cell. The reference electrode was a calomel electrode (Hg/Hg2Cl2/Cl) with saturated potassium chloride solution (4.2 mol/l) as the electrolyte solution. The potential of the calomel reference electrode relative to the standard hydrogen electrode was +242 mV at 25C. The electrolyte solution of the reference electrode was continuously added through a needle to compensate local concentration changes of the electrolyte at the interface. For measurements at 90C the calomel electrode was brought to a temperature of 25C to maintain functionability. In order to determine the potentials of local surface areas, the reference electrode was provided with a Haber-Luggin capillary which had to be brought close to the sample surface. This capillary reduces the ohmic drop in voltage in the electrolytes (3). For measuring the rest potential, the electrode potential was measured in comparison to

the reference electrode using a voltmeter of very high internal resistance (>1012 ohm) to ensure that no current flows in this part of the circuit.
The corrosion behavior of the materials was examined in the following repository-relevant salt solutions:
Table I
Table II
The reference solution for all experiments was bidistilled water.
RESULTS AND DISCUSSION
In order to determine the mass loss, the aluminum materials were exposed to the three different repository-relevant brines in a first test series at a temperature of 90C.
Fig. 1
Figure 1 shows the percentage mass decrease of the materials for solution 2 plotted against the time of exposure. It can be clearly seen that Al 99.5 significantly differs from AlMg1 and AlMg2. Whereas the alloys lose between 1.5 and 2.5% of their mass after 120 days, the mass loss of aluminum 99.5 exceeds 25%. The same trend can be seen for solution 1, although less pronounced. For solution 3 the percentage mass losses of the materials amount to less than 1% after 120 days and are thus about the same as for distilled water. The high mass loss in solution 2 may be attributed to the extremely high magnesium chloride fraction. The hydrolysis of magnesium chloride increasingly leads to a release of hydrogen protons lowering the pH value. The increased proton concentration now has an essential influence on the metal/hydrogen redox pair. In comparison to the normal hydrogen electrode, the potential for aluminum is at more negative values. This means that aluminum is oxidized and hydrogen protons are reduced. The increased proton concentration now shifts the equilibrium of the redox pair towards oxidation of the metal. This leads to an increased release of aluminum cations and enhanced metal corrosion. Moreover, the chloride ions can penetrate into the oxide scale of the metal causing structural changes of this protective layer which contribute towards accelerating corrosion.
The second test series served to examine the influence of temperature on the corrosion rate. The corrosion behavior of Al 99.5, AlMg1 and AlMg2 was studied in solution 1 at room temperature and at 90 degrees Celsius. It can be seen from Fig. 2 that the temperature increase has a greatly accelerating effect on the corrosion rate of Al 99.5. The corrosion rate for Al 99.5 increases by about 109-fold. A similar increase was found for AlMg1, whereas even a 390-fold increase was observed for AlMg2. An increase of the reaction temperature changes the proton concentration. Thus, for example, a pH value of 4.52 is found for solution 1, whereas a pH value of 3.61 was measured at 90 degrees. The increased corrosion rate may also be explained here by the increase in proton concentration. The equilibrium is shifted in favor of metal oxidation and thus towards metal ion reselease. Moreover, kinetically induced processes cannot be ruled out, which are accelerated by the temperature increase and, in their turn, increasingly contribute towards corrosion.
Fig. 2
Furthermore, the influence of metallic additives was studied using those metal alloys (Hasteloy C4; stainless steel 1,4541 and GGG 40) which serve as cask materials.
The time-dependent variation of percentage mass decrease can be seen from Fig. 3. On the whole, the corrosion resistance of the cladding materials

significantly decreases in the presence of metallic additives. The mass of Al 99.5 decreased by 98% within less than 30 days when GGG 40 was added both to solution 1 and solution 2 at 90 degrees Celsius. The corrosion behavior of the cladding materials was only insignificantly influenced by the addition of GGG 40 in distilled water and in solution 3. In the presence of stainless steel 1,4541 and Hasteloy C4, the percentage mass decrease of Al 99.5 was 72% and 84%, respectively, in solution 1 after 50 days. The increase of the corrosion rate is to be attributed here to local element formation. Thus, for example, the aggressive solutions 1 and 2 lead to the release of iron cations from the cask materials in the first phase, which then react with aluminum in the second phase. The iron cations on the aluminum surface are thus reduced, involving simultaneous oxidation of the aluminum metal and dissolution of aluminum cations. The formation of such local elements is the reason for very rapid material corrosion.

Fig. 3

Additional experiments were carried out to study the influence of oxygen concentration and irradiation on the corrosion rate.

Experiments in an argon atmosphere did not show any change in comparison to the reference sample in air. No unambiguous result was achieved with materials exposed to gamma irradiation (8.7*103 rad per hour) during the experiment.

For the electrochemical studies the potential/time behavior of the Al 99.5 cladding material was examined in various brines at different temperatures. The time-dependent variation is shown in Fig. 4 for solution 1.

Fig. 4

The potentials initially decreased towards more negative values at elevated temperatures. The rapid decrease may be explained by the fact that the thin surface scale of the metal was not stable enough at elevated temperatures leading to deterioration. The subsequent increase of the potential values means that the surface scale is built up again in the medium establishing a more stable equilibrium, which is reflected in constant potential values. It is interesting to note that these stable potential values are lower for solutions 1 and 2 than for solution 3, which implies the presence of a thinner protective oxide scale on the metal surface in these media. In the absence of sufficient protection by a stable oxide scale, corrosion can then proceed much more rapidly in solutions 1 and 2.

CONCLUSION

In summary, it may be derived from the mass loss experiments that an increase in reaction temperature and the addition of GGG 40 to the aggressive repository-relevant brines 1 and 2 causes a drastic increase of the corrosion rate especially for the material Al 99.5. AlMg1 and AlMg2 did not show any corrosion resistance either.

The electrochemical experiments made it possible to quantitatively describe the different behavior of the oxide scales under the respective experimental conditions by recording potential-time curves. The correlation between the corrosion reaction, i.e. metal dissolution, and the oxide scale as the decisive electrochemical parameter of influence was established. The more stable the oxide scale on the metal surface, the slower is the corrosion rate. This result is in conformity with the results from mass loss experiments, and it confirms that the cladding

materials tested do not have a sufficient barrier function to avoid radionuclide release.
In the further course of experiments, spent fuel element sections will be exposed to the brines. The aim is to determine the source term so that release calculations can be established for the respective radionuclides.
These studies were supported by the Federal Ministry of Education, Science, Research and Technology under reference number 02E86547.
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7-4
RADIONUCLIDES REMOVAL FROM HIGH SALT CONTENT LOW LEVEL LIQUID WASTE FROM THE REPROCESSING PLANT - EXPERIMENT RESULTS OF ACTUAL LIQUID WASTE -

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ABSTRACT
Power Reactor and Nuclear Fuel Development Corporation (PNC) has been developing radionuclides removal technique to get higher volume reduction for the low level liquid waste from the Tokai Reprocessing Plant (TRP). Experiments were done using actual concentrated liquid waste from the TRP. Radionuclides removal process mainly consists of coprecipitation with ultrafiltration and ion exchange. Results of decontamination factors obtained from this experiment are as follows;
1) Decontamination factor >1E+6 for alpha nuclides.
2) Decontamination factor for ruthenium and total beta nuclide is about 1E+2

INTRODUCTION

Power Reactor and Nuclear Fuel Development Corporation has been developing radionuclides removal technique to get higher volume reduction for the low level liquid waste from the reprocessing plant since 1986.(1) Experiments were done using liquid waste from the TRP. The radioactivity of the liquid waste is about 1E+4Bq/ml. Dominant radionuclides are plutonium, uranium, ruthenium and cesium. This liquid waste contains much amount of salt as sodium nitrate and sodium carbonate. Radionuclides removal process mainly consists of coprecipitation with ultrafiltration and ion exchange. This report focuses on the result of the examination about the characteristics of coprecipitation.

WASTE COMPOSITION AND SELECTION OF THE PROCESS

Waste Composition

Two types of low level liquid waste were used in the experiments; (2) the low level liquid waste (LAW), which is generated from the separation-refinement process, the analysis for quality control, etc., and the concentrated liquid waste (MAW). Typical MAW contains many kinds of alpha nuclides (plutonium, uranium, americium, etc.) and beta gamma nuclides (iodine, cerium, ruthenium, cesium, etc) in order of 1E+4Bq/ml. These test solutions also contain many kinds of salt at high concentration and a lot of impurities such as calcium, magnesium.

Selection of the Process

It is important that the process is composed of few unit to minimize the plant and reduce the secondary waste. However, the liquid waste contains many kinds of alpha nuclides (plutonium, uranium, americium, etc.) and beta gamma nuclides (iodine, cerium, ruthenium, cesium,strontium, etc.) so that it is necessary to combine some unit processes according to the chemical characteristics of each radionuclide. After many radionuclides removal techniques were investigated, the following techniques were selected.3) This process are shown in Fig. 1.

1) Iodine is precipitated as AgI by adding AgNO3 and filtrated by ultrafilter.

2) Alpha nuclides and most of the beta gamma nuclides are coprecipitated with Fe(OH)3 and filtrated by ultrafilter.

3) Strontium and cesium are adsorbed by sodium titanate resin and potassium cobalt ferrocyanide resin, respectively.

Fig. 1

OUTLINE OF THE PROCESS

Iodine Removal

After adding sodium sulfite to prevent volatilization of iodine, the pH is adjusted to 7 by nitric acid. At pH7, sodium sulfite is added again to convert iodate into iodide ion. After then, silver nitrate is added to precipitate iodine as silver iodade, which is filtrated by ultrafilter.

Decarbonation

Liquid temperature is kept at 80C with aeration after adding nitric acid to adjust pH to 1, so that carbonate ion and nitrite ion are removed to off-gas as CO2 and NOX gas. This process prevents plutonium and uranium from formation of carbonate complex at the next coprecipitation process.

Coprecipitation I

Ferric nitrate is added to the liquid waste and pH is adjusted to 10. The flocks of Fe(OH)3 generated are filtrated by ultrafilter. Alpha nuclides and most of the beta gamma nuclides are removed with iron flocks.(4,5)

Coprecipitation II

Ferric nitrate is added to the liquid waste and pH is adjusted to 7. The flocks of Fe(OH)3 generated are filtrated by ultrafilter. Amophoteric nuclides, such as antimony, are removed with iron flocks.

Ion Exchange

After the coprecipitation, strontium is adsorbed by sodium titanate resin and cesium is adsorbed by potassium cobalt ferrocyanide resin. Because calcium and magnesium are also adsorbed, sodium titanate resin is needed not only for strontium but also for calcium and magnesium.

EXPERIMENTAL RESULTS

Optimization of the Amount of Coprecipitant

The experiments were carried out to optimize the amount of the coprecipitant. The results are shown in Fig. 2. The more the coprecipitant is added, the bigger the decontamination factor for alpha nuclides becomes. The coprecipitation effect for two times is more

effective than for one time, but the second coprecipitation effect is much smaller than the first one. For the case of ruthenium, the increase of the coprecipitant makes decontamination factor slightly large, but the effect of repeating the coprecipitation are hardly observed. To obtain a large volume reduction ratio, the amount of coprecipitant should be as small as possible to get enough decontamination factor. Moreover, the concentration limit by ultrafilter is 15,000mg/l for iron. To get enough decontamination factor and 1/100 volume reduction by ultrafilter, the amount of coprecipitant is determined as 150mg/l for iron. The frequency of coprecipitation process is determined as one time for the following reasons; 1) small effect of the second coprecipitation, 2) not disturb the volume reduction ratio, and 3) avoid the increase of equipment. Axis of abscissas shows the amount of the coprecipitatant converted into the iron concentration. Axis of ordinates shows decontamination factor get by the coprecipitation and the ultrafiltration. The white points show the data of the coprecipitation for two times, while the black points show the data for one time.

Fig. 2

Effect on Alpha Nuclide Removal

Decarbonation process

Carbonate ion in the liquid waste is considered to form stable carbonate complexes with plutonium and uranium, which are hard to coprecipitate with Fe(OH)3, so it must be removed before the coprecipitation. The experiments using MAW were carried out to confirm the effect of the removal of carbonate ion on the coprecipitation of alpha nuclides. The results are shown in Table I. Decontamination factor of the coprecipitation after the decarbonation is more than the order of 1E+2, while decontamination factor is nearly equal to 1 without decarbonation. The decarbonation process is necessary for effective coprecipitation of alpha nuclides.

Table I

Phosphate ion

The experiments using the liquid waste which had highly phosphate content (max.8g/l) were carried out. The results are shown in Table II. Compared with the results of the experiments using MAW, which has low phosphate concentration (<90mg/l), phosphate ion affects on the removal of alpha nuclides by coprecipitation.

Table II

Effect on Ruthenium Removal (6,7,8,9)

Addition of non-radioactive ruthenium

The experiments were carried out to confirm the effects of addition of non-radioactive ruthenium. Before the experiment of coprecipitation for the radioactive nuclide, 135mg/l of non-radioactive ruthenium is added as Ru(NO3)3. The results are shown in Table III. As it shows, the addition of non-radioactive ruthenium is not so effective to remove ruthenium in the liquid waste.

Table III

Coprecipitation with CoS

Ruthenium removal by coprecipitation with CoS was attempted by using LAW, which already treated in the process shown in Fig. 1.(10) After addition of cobalt nitrate, pH is adjusted to 9.5 and 4% thioacetamide is added at 60C with stirring to generate CoS flock. After cooling, CoS flock is filtrated by the ultrafilter. The results are shown in Table IV.

Ruthenium is still detected after the coprecipitation, so that this treatment is not so effective, too.

Table IV

Adsorption by the zinc electrode posited charcoal

Ruthenium removal by adsorption by the zinc electrode posited charcoal (Zn-C) was attempted using treated LAW same as the method described in the section 4.3.2. The liquid waste passed through the Zn-C column. The results are shown in Table V. The concentration of Ru becomes below the detection limit and more than 1E+3 of decontamination factor values of the whole process are obtained in all the experiments. The zinc electrode posited charcoal is effective for removal of ruthenium, but it is necessary to confirm the continuation of effective removal condition. According to the literature,11) 12) such good condition does not last so long. It is also necessary to obtain the optimized the operation condition to minimize the yield of secondary wastes.

Table V

TOTAL RADIONUCLIDE REMOVAL EFFICIENCY

Finally, the MAW was treated through the nuclide removal process in the following conditions; the 150mg/l of coprecipitant as iron, without the mixture of highly phosphate ion content liquid waste, and the amount of the coprecipitant are the same as the condition described in the section 4.1.-4.3. The results are shown in Table VI. Good decontamination factor were obtained about almost all the radionuclides, especially about alpha radionuclides, however decontamination factor of ruthenium was about 1E+2.

Table VI

CONCLUSION

By using coprecipitants with ultrafilter and ion exchange resin, decontamination factor over 1E+6 for total alpha nuclides were obtained for the low level liquid waste from the TRP. And from now, research and development for higher volume reduction, less secondary waste and improvement of the beta gamma radionuclides removal efficiency especially for ruthenium will be executed.

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7-5

K BASINS SPENT NUCLEAR FUEL REMOVAL AND CONDITIONING

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ABSTRACT

In February 1995, the U.S. Department of Energy (DOE) committed to relocate more than 100,000 individual spent nuclear fuel (SNF) assemblies from wet storage facilities on the Hanford Site in Eastern Washington State, to a dry storage facility. The scheduled start date for fuel retrieval operations is December 1997, and completion is scheduled for December 1999. Parsons Infrastructure & Technology Group, Inc. (Parsons) was tasked by Westinghouse Hanford Company (WHC) to develop a trade study that compared alternatives for SNF retrieval from wet storage and SNF conditioning prior to dry storage. The results of the trade study were used by WHC to support selection of the alternative that would accomplish SNF relocation operations in the safest and most cost-efficient manner. Each alternative focused on two major functions: The SNF Retrieval Operations and SNF Conditioning. The SNF Retrieval Operations consisted of evaluating four distinct alternatives for fuel handling, fuel cleaning, and fuel packaging. The three SNF Conditioning alternatives included vacuum drying the fuel at ambient temperature, vacuum drying the fuel using the decay heat of the fuel, and vacuum drying the fuel by applying supplemental heat to the fuel at a temperature of 350oC. Conceptual designs of the equipment required to perform various functions were developed. The quantity and types of equipment selected was expected to support a schedule for placement of the SNF in dry storage within a two-year period. Time-motion studies were performed to evaluate schedule impacts and expected radiation dose to workers. A life-cycle cost estimate was developed, and included construction and operating costs for each of the alternatives evaluated.

After all data was fully developed, a summary evaluation was performed to compare the fuel retrieval and conditioning alternatives in terms of cost, dose, schedule, and complexity. The evaluation showed that tradeoffs were such that three of the four retrieval concepts, coupled with SNF drying using the supplemental heating alternative, were nearly equivalent. These three retrieval alternatives are leaving the fuel intact in their present storage canisters, modifying their present

storage canisters to allow limited fuel flushing, and removing the fuel from their canisters for cleaning. The trade study results and recommendations were used by WHC to support the selection of the recommended "Path Forward," which is the strategy for placement of the SNF in dry storage.

The Path Forward recommendations made by WHC, subsequent to issuing the trade study, were to remove the SNF from their canisters, thoroughly clean the SNF, remove the water from the fuel using a vacuum drying process, and then remove the residual and bound water using the supplemental heat vacuum drying process.

INTRODUCTION

In February 1995, the DOE approved the Spent Nuclear Fuel Project Path Forward recommendation for resolution of the safety and environmental concerns associated with the deteriorating SNF stored in the Hanford Site's K Basins. The K Basins consist of fuel pools within two Production Reactor buildings near the Columbia River. The Path Forward recommendation described the facilities and processes required to safely retrieve, package, transport, condition, and dry store the K Basin Fuel. In addition to approval of the Path Forward, the DOE committed to an accelerated schedule to begin fuel removal from the K Basins by December 1997, and complete it by December 1999.

In response to DOE's goals, WHC initiated a project to further develop the technical baseline for the Spent Nuclear Fuel Removal and Conditioning Project. Parsons was tasked to develop concepts for removing the spent nuclear fuel from the K Basins, conditioning the fuel for dry storage, and evaluating and documenting the results in a Trade Study Report. The study goal was to delineate the systems and processes required to perform those functions, and to evaluate the life-cycle costs, personnel exposure levels, schedule for fuel removal, risks and uncertainties, and impacts of these interfacing systems.

Approximately 2,100 metric tons of uranium in the form of more than 100,000 individual SNF assemblies are presently stored in canisters in the K Basins. Most of the SNF originated in the N Reactor facility and is low-enriched, metallic uranium, clad in Zircaloy. A typical SNF assembly is approximately 26 in. Long and 2.4 in. In dia. A SNF assembly is made up of on inner SNF element with cladding, and an outer SNF element with cladding. The inner SNF element has a small-diameter hole in the center and is held in place with springs; however, many of the springs have corroded and may not be in place. There are four types of canisters in the K Basins. Each canister has two barrels made of either stainless steel or aluminum, and holds up to 14 fuel assemblies; however, not all of the canisters are full and some of the SNF elements are broken. A significant fraction of this fuel has become degraded caused by cladding breaches during reactor discharge and subsequent corrosion during underwater storage.

Both of the K Basins are constructed of reinforced concrete and were placed in service in the early 1950s. They are located within 0.25 mile of the Columbia River. K West Basin was cleaned, refurbished, and epoxy-coated in 1981. The SNF in K West Basin is stored in closed, stainless-steel or aluminum canisters. Visually, these canisters appear to be in good condition.

K East Basin was not refurbished or epoxy-coated; it has been used to store N Reactor fuel since 1975. Fuel is stored there in open, stainless-steel or aluminum canisters. Degradation of the aluminum canisters and

breached fuel appears to be substantial. The water in the K East Basin is radioactively contaminated, and area dose rates near the pool surface are relatively high. Significant quantities of sludge to a depth of nearly 1 ft. are visible in some pool locations.

ALTERNATIVE CONCEPTS DEVELOPED

The Trade Study addressed many aspects of the fuel retrieval and conditioning operations outlined in the original Spent Nuclear Fuel Project Path Forward. The concept scope included developing equipment that would safely retrieve and handle the fuel bearing canister from racks at the bottom of the fuel pools. Equipment was conceptualized that would clean the sludge and corrosion products from the SNF elements. Variations in the equipment configurations allowed cleaning the fuel while it remained in the canisters, or after removal from the canisters. Systems were designed to minimize the amount of sludge released into the fuel pools during the fuel retrieval and cleaning operations. Fuel handling equipment and methods were developed for loading the fuel into the primary containment vessel called the Multi-Canister Overpack (MCO). The MCO is a 24-in.-dia., 160-in.-tall, stainless-steel vessel. After the fuel is loaded into the MCO, the MCO is welded closed and the water is removed. Residual water left in the MCO will be removed by one or more vacuum conditioning processes. The basic functions addressed in the trade study are shown in Fig. 1.

SNF Retrieval Concepts

The alternative methods for SNF and canister retrieval, cleaning, and MCO loading required developing methods for performing, handling, and machining operations in an underwater environment. The four separate concepts evaluated were as follows:

Concept 1A, As-Is, Direct loading of the SNF canisters into the MCOs;

Concept 1, Limited Cleaning, Limited flushing of the SNF in the canisters prior to MCO loading;

Concept 2, Slotting and Flushing, Canister machining and rigorous SNF flushing in the canister prior to MCO loading; and

Concept 3, Separate Fuel from Canister, Removal of the SNF from the canisters, and rigorous SNF and canister cleaning prior to MCO loading and fuel reracking in baskets.

Fig. 1

Spent Nuclear Fuel Conditioning Concepts

S-1, Dewatering and Drying, The S-1 conditioning effort was in the initial conditioning step required for all scenarios. Step S-1 is accomplished by draining/pumping the free water from the MCO, followed by low-temperature "cold" vacuum drying.

S-2, Cold Vacuum Conditioning, This step was one of the two "complete drying" steps to further dry the contents of the MCO. Step S-2 is accomplished by using only the fuel decay heat over time to heat and dry the fuel.

S-3, Hot Vacuum Conditioning, This step was the second "complete drying" steps to further dry the contents of the MCO. Step S-3 drying would be accomplished by using supplemental heating of the MCO (to approximately 300oC), and holding the fuel at this temperature and under vacuum to remove nearly all retained moisture from the fuel and corrosion products within the MCO.

Figure 2 shows the three alternatives for dewatering and vacuum drying the fuel in the MCO, and includes the calculated number of stations required to condition all fuel within a two-year period.

Trade Study Development

Equipment concepts were developed that could perform the various functions of the process, from fuel retrieval through final storage. Equipment layouts were made showing equipment in the K Basin Facilities and in new facilities. Figure 3 shows a typical process equipment layout in a K Basin pool. With the time restriction of two years for complete fuel retrieval, some of the systems required redundant equipment items. All fuel retrieval alternatives will be performed in the two K Basin Facilities. The current process logic for fuel conditioning allows conditioning steps to be performed at one or more of four distinctly different facilities in different locations. These four facilities are the K Basin(s), the K Basin Annex(es), the Canister Storage Building (CSB), and the conditioning facility. The CSB Annex was added as a fifth facility, and although considered a separate alternative, it would actually be a part of the CSB. Depending on a number of different factors (including facility space available to perform conditioning operations, shipment conditions of the MCOs, location of the facilities themselves in relation to Hanford Site boundaries, MCO transportation constraints, etc.), a given conditioning step may not be amenable to perform at a given facility, or may be done much more efficiently and logically at one facility, versus another.

Figure 4 shows option paths evaluated. These were based on 24 (18 paths plus 6, S-1 at the CSB Annex) feasible processing paths. Each path has one of two choices for MCO loadout. These consist of either remote loading or water-shield loading, which brings the total number of options to 48.

After the equipment concepts and layouts were produced, time-motion studies were performed for all operations, from fuel retrieval in the K basins through final placement of the MCOs containing the fuel into the CSB. Activities were defined and the crew sizes required to carry out each activity were established. These studies revealed that the Crane Availability in each of the K Basins was the limiting factor for meeting the two year fuel removal schedule. The results of the time-motion studies were then used as input to define the approximate total person-dose received by workers during the fuel removal operations. Doses were calculated using the computer dose modeling code Microshield.

A cost estimate for each of the alternatives was developed. As the basis for the cost estimates included in the trade study, it was assumed that construction and modifications to facilities would be performed by the onsite construction contractor, ICF Kaiser Hanford, and the operation of facilities and equipment would be performed by WHC personnel. Sketches, drawings and descriptions were used to establish quantities of equipment required, building areas, configurations, and construction types, as well as installation complexity. Because the level of detail of the design documents lacked the final design detail, a contingency factor was applied to the estimates.

Fig. 2

Fig. 3

Fig. 4

EVALUATION CRITERIA

The fuel removal and conditioning concepts developed were evaluated and compared to ascertain their potentials for meeting cost, schedule, technical, and other requirements. The evaluation used a systems engineering approach for assessing the fuel removal and conditioning

alternatives. It facilitated a side-by-side comparison of functions and requirements of the fuel handling and processing alternatives. Key criteria were established and used to perform the evaluation. They consisted of: 1) Cost--Total of both acquisition and operating costs in dollars; 2) Exposure risk--(dose) in person-rem; 3) Complexity - this criterion included the potential for leaving residual moisture in the MCO after drying; and operability and maintainability considerations, including operations steps, special handling, and operations simplicity. The rankings were based on the professional judgement of the evaluators and ranged from 1 (least potential) to 4 (most potential). Each alternative was included in an option description worksheet. The data was then plotted and compared by concept to show the options that minimized cost, dose complexity, and schedule. (Table I).

Table I

TRADE STUDY EVALUATION RESULTS

Comparisons between the various alternatives (Table I) were made in terms of cost, schedule, dose to workers, and operational complexity. The results of the evaluation showed that Concept 2 with S-1, S-2, or S-3 conditioning was not a favorable option because of a higher cost, schedule, and dose to workers than any other options. The evaluation results also show that the tradeoffs of cost, schedule, dose, and complexity in Concepts 1, 1A, and 3 with S-1/S-3 conditioning are such that these concepts are nearly equivalent.

Concept 1A, As-Is

Concept 1A operations consist of essentially three steps: 1) Retrieving the canisters; 2) Removing the canister lids; and 3) Loading the MCOs. S-1/S-3 conditioning at the new conditioning facility offers the most efficient method of conditioning, with the least amount of time and equipment required per MCO. Based on the evaluation data, this concept scored favorably in both cost and dose received by workers when conditioning Step S-3 was used. Concept 1A required the minimum amount of equipment and the minimum number of operations to perform fuel removal operations in the K Basin. The capital and operating costs of the alternative with S-3 conditioning are relatively low because of the absence of complex equipment for fuel retrieval. Fuel handling activities by remote or underwater methods limit direct exposure to personnel. The minimization of fuel handling activities in the K Basin process area translates into less worker dose.

The estimated time it takes to perform fuel removal operations was moderate in relation to the other concepts. The primary driver behind the fuel removal schedule is the availability of the crane to perform various functions during MCO loadout, shield plug welding, and cask transfer operations. A disadvantage to Concept 1A is the unknown state of the fuel and sludge after the MCO is sealed. Concept 1A fuel removal with S-2 conditioning shows a substantial increase in acquisition costs because of the increased operations and equipment costs of the conditioning process.

Concept 1, Limited Cleaning

Concept 1 operations consisted of essentially four steps: 1) Retrieving the canisters; 2) Removing the canister lids; 3) Flushing the canister/fuel; and 4) Loading the MCO. Compared to Concept 1A with either S-2 or S-3 conditioning, the major difference in the data shown on the evaluation table is that this concept has generally higher total costs and dose to workers. The higher costs and higher personnel doses are attributed to additional equipment costs and additional operating times

respectively, to perform the flushing activities. Schedule and complexity score relatively the same as Concept 1A.

Concept 2, Slotting and Flushing

Concept 2 operations consist of five steps: 1) Retrieving the canisters; 2) Removing the canister lids; 3) Machining the canister; 4) Flushing the canister/fuel; and 5) Loading the MCO. This concept is shown to be generally not a favorable option. Total costs are relatively high because of the increased complexity of canister and fuel handling equipment required for machining the canisters, and the increased time it takes to perform all fuel removal operations. Dose rates are unfavorable and reflect the doses operators would received while performing basin operations for greater than 100 weeks. This concept with either S-2 or S-3 conditioning has generally the worst scores in all categories the criteria evaluated. An advantage to this concept is that a quantity of sludge and corrosion presently in the canisters and on the fuel will be removed, and will provide improved conditioning products.

Concept 3, Separate Fuel from Canister

Concept 3 with either S-1/S-2 or S-1/S-3 conditioning at various locations was evaluated. Concept 3 operations consist of six steps: 1) Retrieving the canisters; 2) Removing the canister lids; 3) Removing the fuel from the canisters; 4) Cleaning the fuel; 5) Reracking the fuel in baskets; and 6) Loading the MCO (this doubles the fuel packing density of Concept 1A, 1 and 2). Dose rates encountered in this process are approximately 30% higher than the doses calculated for Concepts 1A and 1. The reason for the increased doses for the concept is because of the increased number of manhours actually spent in the basin area for fuel repacking. The schedule for fuel removal is improved (approximately 55 weeks or approximately one-half of Concept 1A and 1) because of concurrent processing of "good" and "bad" fuel, and the 50% reduction in the number of MCOs needed for complete fuel removal from the Basins. Costs for the option using Concept 3 are generally less than the other options because of lower acquisition costs for MCOs, and lower labor costs for MCO and cask handling activities. The schedule advantage of this concept seems to present an attractive alternative. Improved knowledge of sludge and corrosion product inventories in the MCOs after fuel rod cleaning will prove advantageous during the fuel conditioning operations. This knowledge will also improve confidence in the determination of the fuel condition future years, when the fuel is processed for final disposal.

CONCLUSION

Subsequent to the trade study completion, WHC refined the Path Forward to reflect Parsons recommended approach for fuel retrieval and conditioning. That approach shows repacking the bare fuel in storage baskets (Concept 3) and applying a two-step fuel drying and conditioning process (S-1/S-3).

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AT-LINE DETERMINATION OF VOCs IN WASTE DRUM HEADSPACE WITH FOURIER TRANSFORM INFRARED SPECTROSCOPY

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ABSTRACT
Transuranic (TRU) radioactive wastes have been retrievably stored in waste drums at Department of Energy (DOE) sites since the 1970's. Ultimately, these waste drums are destined for final disposition in the Waste Isolation Pilot Plant (WIPP). Current requirements for acceptance of waste into the WIPP dictate that a representative drum headspace sample be acquired and analyzed prior to the transport and disposal of waste in the WIPP. Analysis results of the headspace sample are to be used for waste characterization, verification of process knowledge, assigning Environmental Protection Agency (EPA) hazardous waste codes, determining the potential for flammability, and as input to gas generation and transport models. Because of the very large number of waste drums and the rate at which they will need to be processed, a rapid, simple and reliable analysis method for waste drum headspace that can be performed "at-line" is necessary. Fourier transform infrared (FTIR) spectroscopy was selected because the analysis times are short, operation of the instrumentation is simple and reliable and because FTIR systems are rugged and can be easily configured to function for "at-line" analysis. Drum headspace samples are pulled directly into a cell mounted on an FTIR spectrometer and a spectrum recorded. From each infrared spectrum, 29 volatile organic compounds, the C1-C3 hydrocarbons, and some interferences are identified and quantitated. To evaluate the analytical performance of the FTIR system and methodology on real samples, over 200 gaseous samples of actual TRU waste drum headspace and the headspace of other inner layers of confinement within the waste drums have been analyzed by an "at-line" FTIR system. Analytical results are available within 5-6 minutes of sample collection. The FTIR analysis results were compared to the results from duplicate samples that were collected in SUMMA canisters and analyzed by the standard laboratory gas chromatographic (GC) methods. The FTIR analysis results agree well with the chromatographic analyses and will meet the program required limits for accuracy and precision for the analytes of interest. To date, the results indicate that FTIR spectroscopy is a viable, cost effective alternative to the laboratory based GC methods currently specified for the analysis of TRU waste drum headspace.
INTRODUCTION
Transuranic (TRU) wastes have been retrievably stored at Department of Energy (DOE) facilities since the 1970s. In 1980, the Waste Isolation Pilot Plant (WIPP) was authorized and funded to provide a research and development facility focused upon demonstrating the safe disposal of radioactive wastes. Because of the wide variety of activities that generated DOE's TRU waste, some of the waste also contains components that are regulated by the Resource Conservation and Recovery Act (RCRA). To ensure that the waste to be admitted to the WIPP meets a variety of compliance programs, including RCRA, the TRU Waste Characterization Program (TWCP) was initiated. Waste characterization data from the TWCP will be, and currently are being used, to support the assessment of WIPP repository performance, the application for permits and variances issued

by regulatory agencies, and necessary revisions to transportation restrictions.

The EPA in its conditional no-migration determination for the WIPP,(1) indicated that before DOE submits a petition for the actual disposal phase at the WIPP, real sampling and analytical data would be required to characterize the waste. Among the chemical analyses requested is the determination of volatile organic compounds (VOCs) in the headspace of the waste drums and all additional layers of confinement within each waste drum. The VOC data are necessary to evaluate the potential for migration of these compounds beyond the WIPP boundary above EPA determined health based risk limits (2) and to confirm that the waste drum headspace is representative of the drum contents. The EPA is also requiring quantitative data for the flammable VOCs, hydrogen and methane prior to disposal at the WIPP. Due to the potential for explosive mixtures in waste drum headspace, EPA has currently imposed a 500 ppmv concentration limit for total flammable VOCs in the headspace of containers to be sent to the WIPP during the test phase.

A list of 29 VOCs was compiled from examination of available records associated with the descriptions of the activities that may have generated waste within the DOE complex and from actual records describing the stored wastes. Exposure and lower explosion limits of these compounds are indicators of what concentrations might be of reasonable concern for waste handling and storage. Some characterization of stored TRU waste which included some headspace analysis has been performed and reported.(3) Generally, fewer then 5 of the identified VOCs are found in any particular sample.

At the Idaho National Engineering Laboratory (INEL) alone, there are as many as 140,000 containers of TRU waste. Most of these containers are 55-gallon drums with sealed lids. Due to the potential buildup of flammable gases in these drums from radiolysis, these drums are to be vented and appropriate filters installed in the lid to allow the drum to aspirate. At the time of filter insertion, it is desirable to collect and analyze a sample of the headspace for characterization and classification purposes. Because of the number of drums that need to be vented, the INEL has designed and built an automated Drum Venting Facility (DVF) that will vent drums at the rate of one every 5-10 minutes. As part of this facility it is desirable to have a rapid VOC analysis technique to provide near-real time feedback of analysis results to dictate the immediate storage and handling requirements for each drum. In the Waste Characterization Area (WCA) located at Argonne National Laboratory-West (ANL-W), waste drums are opened for detailed characterization and examination. A rapid VOC analysis technique is also desirable at the WCA since all layers of confinement within each drum need to be sampled for VOCs, CH4 and H2 analysis.

The currently accepted methods for the analysis of waste drum headspace components are gas chromatography with thermal conductivity detection (GC/TCD), gas chromatography with flame ionization detection (GC/FID) and gas chromatography with mass spectrometric detection (GC/MS).(4) A typical sampling and analysis of waste drum headspace consists of collecting the gaseous sample in a SUMMA canister, surveying for radioactive contamination, transporting it to the laboratory, an initial dilution, screening via GC/FID analysis, the final dilution, GC/MS analysis for VOCs, GC/TCD analysis for H2 and CH4, and finally reporting in data packages. The total cost for each analysis is ~$1000 and the

turnaround time is as much 1 month, primarily because of the data reduction and reporting requirements. Minimum actual analysis times would be 15-45 minutes per chromatogram once the sample is received in the laboratory.

Because of the time necessary to develop a typical chromatogram, chromatography is not a preferred technique for the at-line analyses outlined above. This is particularly true if dilution is required to get the sample into the working analytical range. Some "high-speed" chromatography applications exist; however, they suffer from reproducibility problems without special sample inlet systems, must sacrifice chromatographic resolution for analysis time, and often more than one column/chromatographic system must be used to quantitate multiple analytes.(5,6) Separate columns are also necessary for the analysis of permanent gases such as H_2, CH_4, and CO_2.

A reasonable alternative to at-line GC instrumentation for the analysis of waste drum headspace is Fourier transform infrared spectroscopy (FTIRS). Rugged FTIRS instrumentation is commercially available and has been used for several on-line applications.(7,8) Depending upon the instrument/detector used, analysis times using FTIRS can be as short as 1 second, however, analysis times of 1-5 minutes are more common. Most VOCs and permanent gases can be analyzed with infrared spectroscopy. Exceptions include diatomic molecules without dipoles such as H_2 and O_2. Because of the nature of absorption spectroscopy, calibrations/standardizations can be universal within certain limitations, i.e once a calibration/standardization is established it is possible to transfer this calibration from instrument to instrument, provided that the data are collected under similar conditions.(9)

The goal of the work outlined in this paper was to evaluate and determine if FTIRS is a suitable alternative to GC methods for the "at-line" analysis of waste drum headspace in order to support various decisions about whether the waste drums meet TWCP requirements and certain WIPP Waste Acceptance Criteria. The basic questions lie in whether analysis by FTIRS produces statistically equivalent results as the GC procedures and/or whether it meets other TWCP criteria for precision (25%) and accuracy (30%).(4)

EXPERIMENTAL

Bomem furnished the FTIR based VOC system as specified by the INEL. Because the "turn-key" FTIR system is located in a "suspect radiation contamination zone" near where the waste drums are handled and mounted into the waste characterization hot cell at the WCA, the hardware is housed in a NEMA 12 enclosure (48 in. tall x 36 in. wide x 16 in. deep) for protection from damage and from contamination by radiation. Figure 1 is a schematic of the components comprising the FTIR based VOC analysis system. In the NEMA 12 enclosure, a Bomem MB 100 series FTIR is vertically mounted and equipped with a specially designed top plate. The optical bench is purged with hydrocarbon and CO_2 free dry air which is vented into the NEMA 12 enclosure to help maintain a slight positive pressure within the enclosure. A 20 cm gas cell with zinc selenide windows with an antireflection coating to reduce the refractive index and a DTGS detector are mounted on the top plate. All sample transfer lines and the sample cell are maintained at 110C. Transducers are mounted in the cell to record the temperature and pressure of each sample. Because of the heat load supplied by the instrumentation and the other heated components, the NEMA 12 enclosure is cooled and maintained at ~28C with a

closed cycle air conditioner. Operation of the valves and the FTIR are controlled via RS422 from a 486 based PC located >60 feet away at the WCA control center.
Fig. 1
Sampling and analysis is initiated when the start signal is received from the GSS computer. At this point, the three way valve (V1) rotates toward the sample line, and the cell and lines are evacuated to <2.5 Torr by opening V4. The cell and lines are backfilled to ~625 Torr with hydrocarbon and CO2 free dry air by closing V4 and opening V3. The cell and lines are reevacuated by closing V3 and opening V4. Once evacuated to <2. 5 Torr, V4 is closed and the "ready evacuated" signal is sent to the GSS computer, which then opens V0, and the lines and cell begin to fill with the sample. When the pressure has stabilized, spectral acquisition is initiated. Each spectrum is the result of 10 coadded scans. A second spectrum of a "diluted" sample can be acquired by rotating V1 toward the air line and then opening V2 until the desired lower pressure is obtained, then the cell is backfilled to 640 Torr with hydrocarbon and CO2 free dry air.
Because quantitative IR spectra of several of the VOC's of interest were not available at all or not available at the conditions at which the sample spectra were to be recorded, it was decided to record the quantitative spectra at the sampling conditions, i.e., at 110C with a nominal pressure of 640 Torr (near ambient pressure in eastern Idaho). Bomem assembled a second system at their facilities with a gas sampling manifold that included a cold finger to trap VOCs and a MB 100 series Fourier transform infrared spectrophotometer with an identical top plate to that provided the INEL. To collect a pure component spectrum of a particular VOC, a sample of the neat analyte was put into the cold finger and then the finger was attached to the manifold and the sample was frozen with liquid nitrogen. The space over the frozen sample in the cold finger, manifold, and heated gas cell were evacuated using a mechanical and a turbo pump in series. Once a stable vacuum was reached, the cold finger containing the sample was isolated by closing the valve, the liquid nitrogen was removed, and the sample heated to room temperature. The valve to the sample was then opened and the sample allowed to "evaporate" into the evacuated manifold and cell until the desired partial pressure of the analyte was reached (usually ~0.64 Torr). The total pressure was then brought to 640 Torr with nitrogen (i.e. 1000 ppmv) and the spectrum was acquired with 50 coadded scans at 1 cm-1 resolution. Many of the more polar and less volatile VOC's presented problems that were likely due to adsorption of the analyte onto the walls of the cell and manifold. The addition of nitrogen to the sample to bring it to 640 Torr seemed to enhance the instability of the analyte partial pressure. For these cases, additional nitrogen was not added to bring the total pressure to 640 Torr. There were no apparent differences in the spectral features of the analyte spectra recorded at low and high pressure. Linearity of the samples created in this manifold system was verified by using carbon tetrachloride to construct a calibration curve from 0 to 1000 ppmv. Carbon tetrachloride was selected because of its very high absorbtivity coefficient.
Examination of IR spectra of actual waste drum headspace indicated that interferences that were not included in the original calibration set were often encountered. These interferences included NH3, N2O, trimethylamine, >C6 hydrocarbons and very wide concentration ranges of CO2. To compensate

as much as possible for very wide ranging CO2 concentrations, a series of six CO2 standards were prepared that when combined with the existing 2500 and 25000 ppmv CO2 spectra covered the range from 2500-50,000 ppmv. Certified 1000 ppmv standards of N2O and NH3 were purchased from Scott Specialty Gases (Plumsteadville, PA). A certified 1000 ppmv standard of trimethylamine was purchased from SO-CAL Airgas (Los Angeles, CA). A hydrocarbon standard was prepared at 955 ppmv from decane (333 ppmv), undecane (322 ppmv) and dodecane (300 ppmv). Spectra of each of these standards were collected on the "turn-key" system installed at the WCA or an identical system intended for installation at the DVF.

Quantitative analysis of the VOC's from the infrared spectra collected on the "turn-key" FTIRS system was performed using PLS.(10) The PLS analysis methods on the "turn-key" FTIRS system were generated using Galactic Industries PLSplus add-on package to Grams/386. Because of software limitations and the cumbersome calibration sets required for a single method that would quantitate all 29 target VOC's and the C1-C3 hydrocarbons, individual PLS methods were used for each analyte in an optimally selected region of the spectrum for that analyte. Calibration sets consisted of the original calibration set supplied by Bomem and the supplemental interference spectra collected at the WCA (n=110). All spectra in the set were background corrected to avoid extraneous factors that might affect quantitation of an analyte. Artificial spectra containing offsets and sloping lines were added to the set for simple background factor definition. Frequency regions for each analyte were selected after evaluating the correlation spectra for that component calculated by a development aid in the PLSplus package and the actual spectrum of the analyte. An optimum number of factors for each analyte method were selected from the evaluation of the predicted residual error sum of squares (PRESS) values determined using the cross-validation procedure included in the PLSplus package. A summary of the frequency regions and factors used in these methods is given in Table I. Evaluation of an unknown spectrum for the 29 VOC analytes and methane, ethane, and propane using 32 separate PLS methods takes ~10 seconds. A complete analysis and report are provided to the operator within 3-6 minutes of sample introduction to the system.

Table I

RESULTS AND DISCUSSION

The suitability of using FTIRS for the at-line determination of VOCs in waste drum headspace was evaluated by several means. Routine daily FTIR analysis of a check standard was used to assess long term precision, accuracy and determine subsequent working detection limits. Direct comparisons were made between the FTIRS results and results from "identical" samples collected in SUMMATM canisters. Further assessments of accuracy, precision and error rates were accomplished by participation in the TWCP Performance Demonstration Program.(11)

During operations at the WCA, prior to any sample collection or analysis on a given day, a check standard was analyzed by the FTIRS and occasionally sampled into a SUMMATM canister for later analysis by GC methods. A summary of the results from the analysis of these check standards is presented in Table II. Two separate standards were used over a period of ~6 months, each for ~3 months. The current sampling manifold design at the WCA requires that the check standards be transported through ~25 feet of unheated tubing prior to reaching a heated portion of the manifold. Polar VOCs tend to interact with the walls of the tubing

and are not efficiently transported without heat or the presence of a more polar compound to displace it from the active sites on the surface. This problem is clearly reflected in the recovery and precision for the analysis of methanol by both the FTIRS and GC analyses of the first standard which was dry. Addition of water to a low pressure gas standard containing polar VOCs which was then sampled through the same tubing demonstrated that methanol could be quantitatively transferred through this tubing if a more polar compound was present to compete for binding sites. A second standard was ordered that contained water, however methanol was still no efficiently reaching the manifold. At 500 psig, some of the water apparently condensed in the cylinder and some of the methanol was lost in the condensation.

Table II

Examination of Table II indicates that aside from the problem with the methanol transport, the FTIRS method performed well and was within the TWCP requirements for 30% accuracy and 25% precision. Long term precision for the FTIRS method is most typically <10%. Accuracy for all analytes in Table II, except methanol, is within the TWCP requirement, however they appear to be 5-15% negatively biased. The GC analyses for some analytes also appear to be biased, but in a positive direction, e.g. 11-dichloroethane and 11-dichloroethene.

Because FTIRS spectra of headspace VOC samples are highly overlapped, direct analysis of the spectrum is difficult at best and special data processing like PLS is required to simplify the analysis. Each compound has a unique IR spectrum, which is the fact that allows the PLS algorithm to easily extract the necessary qualitative and quantitative information from a highly overlapped spectrum. The results can be slightly affected by the number and identity of the analytes present in the sample spectrum. Because of this, the determination of appropriate detection limits is not necessarily obvious. The components in the check standards were selected primarily because of how frequently they occur in the actual waste drum headspace samples. The were secondarily selected to represent classes of compounds. Rarely are more than 5 VOCs found in a given headspace sample. The analysis of the check standards for all VOCs of interest should therefore be representative of the samples and the long term precision should be representative of the actual sample analysis. Because of the representativeness of this standard and analysis, the precision times 3 is also an appropriate working detection limit for the FTIRS analysis. The detection limits shown in Table I were calculated from the repeated analysis of these check standards. These detection limits compare well with and are only slightly higher than published detection limits(12), as expected sine the published detection limits were obtained under optimum conditions.

During operations at the WCA, headspace samples from waste drums and all layers of confinement within the waste drums were sampled in to 250 mL SUMMA canisters for later analysis by standard GC methods. If the headspace volume was deemed sufficient, a second sample was collected directly into the at-line FTIRS system for immediate analysis. In all, 231 samples were collected and analyzed by both the GC methods and FTIRS. Of these 231 samples, 15 represent direct comparison of the FTIRS and GC/GC-MS for the two reference standards. In the actual headspace samples, 111-trichloroethane was found in nearly all samples. Trichloroethene, 11-dichloroethene, acetone, toluene, methanol and carbon tetrachloride were also found in some samples. Only 111-trichloroethane

and trichloroethene were found at significant concentrations in the samples.

Overall, for the samples that were above the detection limits for both GC/GC-MS and FTIRS the correlation line for 111-trichloroethane was FTIRS=(0.9960.014)*GC-(217) with an R2 of 0.958. The y-intercept implies a negative offset of ~20 ppmv. For trichloroethene, the regression line when both techniques were above the detection limit is given as FTIRS=(0.880.03)*GC-(910) with an R2 of 0.934. The ~12% negative bias noted in the slope is consistent with the accuracy associated with the analysis of the reference standard (see Table II). With few exceptions the two analysis techniques track each other quite well for 111-trichloroethane and trichloroethene. Correlation for 11-dichloroethene does not appear be good; however, if one considers the 15% positive bias for the GC-MS method (see Table 5-2) and the ~10% bias for the FTIRS, this result is not surprising.

The regression line for acetone when results for both the GC and FTIRS techniques are above their detection limits is FTIRS=(0.830.04)*GC-(03) with an R2 of 0.900. Examination of the overall data for acetone indicates a large number of false negatives for FTIRS analysis relative to the GC. Most alarming were the false negatives when the GC result was at 70-100 ppmv. Examination of the data one of these samples is interesting. For this sample, the GC indicated a concentration of 100 ppmv acetone and 1900 ppmv 111-trichloroethane. The FTIRS analysis indicated that acetone was <18 ppmv, however 1-butanol, 2-butanone, ethyl ether and 111-trichloroethane were at 101, 413, 95, and 1800 ppmv respectively. Because of the complexity of the sample spectrum, direct examination after 111-trichloroethane had been subtracted, could not rule out the presence of acetone or, confirm the presence of 1-butanol, 2-butanone, or ethyl ether because the presence of an unidentified compound present that has spectral characteristics similar to alcohols and/or ketones. As many as four other compounds were known to coelute with acetone in the GC-FID method. The compounds that are known to coelute with acetone are freon 113, ethanol, carbon disulfide and dimethoxymethane. Because of the high concentration of 111-trichloroethane, a very large dilution was required for the second analysis by GC-MS which raised the detection limits. However, reanalysis of the stored GC-MS data confirmed acetone. To determine the final dilution factor for the GC-MS analysis, a screening GC-FID is run at ~3.5x dilution of the sample. This analysis indicated a variety of peaks that were not identified. The FTIRS analysis was performed with no dilution of the sample.

Correlation between the FTIRS and GC analyses for toluene indicates a significant number of false or severely positively biased results for the FTIRS method. The main absorption band for toluene is interfered with severely by 111-trichloroethane and CO2. The magnitude of spectral residuals for the toluene method could be directly correlated to the positively biased toluene results and the presence of very high CO2 concentrations. Because the CO2 concentration was so high, nonlinear absorption of light affected this region of the spectrum. Because toluene is also found in nearly all of the actual headspace samples at low concentrations, toluene was likely positively identified but the quantitation was likely biased because the calibration for toluene is linear, however the absorbance is now nonlinear due to the excessive CO2 and spectral residual resulted. The CO2 was likely correctly compensated

for since the series of CO2 spectra in the calibration set were designed to identify the nonlinear behavior associated with high concentrations. The overall analysis results of the FTIRS compared to the GC methods is shown qualitatively in Table III as the number of true and false identifications are given along with the calculated total error rates ((false positives + false negatives)/total analyses), sensitivities and selectivities. The overall total error rate is generally in the 8-9% range with only 1% of these errors being false negatives. The specificity indicates that the FTIRS will discriminate against the analytes that are not present ~92% of the time. Likewise, the FTIRS method will identify the presence of the components ~90% of the time. The results are somewhat better when the methane, trimethylamine and hydrocarbon results are dropped. Trimethylamine and hydrocarbons are tentatively identified compounds in the GC/GC-MS methods and may not have been reported if their concentrations were low. Methane was only seen in the reference standards and the GC method failed to identify it even though the concentration was above the detection limit. The sensitivities listed should be viewed with caution, since the results may be biased by the low number of true positive for most analytes.

Table III

Examination of Table III indicates a large number of the cis-12-dichloroethene analysis results were false positives. The detection limits for cis-12-dichloroethene in the samples was usually in the ~12 ppmv. Of the false positives, 91% were <20 ppmv and all were less than 27. Most compounds, false positives could be similarly categorized. The major exceptions have been noted in the previous discussions. For example, many of the false positives for ethyl ether and 1-butanol could be due to the interferences which caused the acetone to be a false negative as described above. At two times the detection limits, i.e. a practical quantitation limit, the total error rates drops to <1%, sensitivities are ~95% and selectivities are ~98%.

Among of the quality assurance objectives for the analysis of gas samples for the TWCP is the analysis of blind audit samples as part of the PDP. (11) These blind samples are distributed, analyzed, and evaluated every 263 weeks. Performance is evaluated with blanks where no target compounds exceeding 50% of the PRQL, precision within 25%, accuracy within 30% and qualitative identification of non target analytes. In the most recent series of PDP samples, the FTIRS scored a 93% for the critical target compounds and 81% overall for the target compounds. The overall score was considered a pass and the critical target compound score was considered a conditional pass as a score of 95% is considered optimal. The major problem that may have caused this somewhat low score was that for some of the samples, the target concentrations were at or near the FTIRS detection limit and therefore may have slightly missed the accuracy or precision requirement. The only exception to this was for 1,2-dichloroethane which was ~35% negatively biased. This was the primary error that caused the score for the critical target compounds to be <95%.

CONCLUSIONS

Based upon the work presented in this paper, the analysis of waste drum headspace for VOCs with FTIRS appears to be quite feasible. Qualitative and quantitative determinations of VOCs with FTIRS are generally well within the requirements of the TWCP program, i.e. 30% accuracy and 25% precision. Long term precision of the technique and instrumentation is very good as it is generally <10%. The technique does appear to be

slightly negatively biased however. The reason for this bias has not been positively identified; but, maybe, in part, be due to the number of factors necessary for the calibration to adequately model the component of interest. In any case, the present systems with the 20 cm pathlength cell and DTGS detector appear to be quite capable of quantitative analysis of the 29 target VOCs, the C1-C3 hydrocarbons, and 6 known interferences in 3-6 minutes. The concentration range covered, depending upon the analyte, is from low ppmv to several percent. The use of longer pathlengths and more sensitive MCT detectors will reduce the detection limits and speed the analysis; however, there may be a sacrifice due to increased interference from low concentration unknown compounds in the samples and in the concentration range that can be covered. The calibration sets have been used on at least three different instruments appear and they do appear to be "universal", provided that the frequency accuracy of the instrument is within tolerance.

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ASSESSMENT OF GAS FLAMMABILITY IN TRANSURANIC WASTE CONTAINER

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ABSTRACT

The Safety Analysis Report for the TRUPACT-II Shipping Package (Transuranic Package Transporter-II (TRUPACT-II) SARP) set limits for gas generation rates, wattage limits, and flammable volatile organic compound (VOC) concentrations in transuranic (TRU) waste containers that would be shipped to the Waste Isolation Pilot Plant (WIPP). Based on existing headspace gas data for drums stored at the Idaho National Engineering Laboratory (INEL) and the Rocky Flats Environmental Technology Site (RFETS), over 30 percent of the contact-handled TRU waste drums contain flammable VOC concentrations greater than the limit. Additional requirement(s) may be imposed for emplacement of waste in the WIPP facility. The conditional no-migration determination (NMD) for the test phase of the facility required that flame tests be performed if significant levels of flammable VOCs were present in TRU waste containers. This paper describes an approach for investigating the potential flammability of TRU waste drums, which may increase the allowable concentrations of flammable VOCs. A flammability assessment methodology is presented that could allow more drums to be shipped to WIPP without treatment or repackaging and reduce the need for flame testing on drums. The approach includes experimental work to determine mixture lower explosive limits (MLEL) for the types of gas mixtures observed in TRU waste, a model for predicting the MLEL for mixtures of VOCs, hydrogen, and methane, and revised screening limits for total flammable VOC concentrations and concentrations of hydrogen and methane using existing drum headspace gas data and the model predictions.

INTRODUCTION

The U. S. Department of Energy (DOE) intends to begin operation of the Waste Isolation Pilot Plant (WIPP) beginning in 1998. Waste characterization requirements for shipping transuranic (TRU) waste from DOE sites to the WIPP facility are set forth in the Safety Analysis Report for the TRUPACT-II Shipping Package (Transuranic Package Transporter-II (TRUPACT-II) SARP) (NRC 1994). The TRUPACT-II SARP, in evaluating gas generation rates and potential flammability of the wastes, requires that gas phase concentrations of flammable volatile organic compounds (VOCs) not exceed 500 ppmv and the hydrogen and methane concentration not exceed 5 percent. Additional requirement(s) may be imposed for emplacement of waste in the WIPP facility. In the conditional no-migration determination (NMD) for the test phase of the WIPP facility, the U.S. Environmental Protection Agency (EPA) required that each waste container emplaced underground at the WIPP facility have no layer of confinement containing either flammable mixtures of gases or mixtures of gases that could become flammable when mixed with air. The conditional

NMD required that flame tests be performed if significant levels of flammable VOCs (500 ppmv) were present in TRU waste containers. Based on existing drum headspace gas data for over 800 drums stored at the Idaho National Engineering Laboratory (INEL) and the Rocky Flats Environmental Technology Site (RFETS), over 30 percent of the contact-handled TRU waste drums contain greater than 500 ppmv of potentially flammable VOCs. Under current requirements, these drums cannot be shipped. Moreover, the volume of drums that are in accessible storage that can be shipped will be significantly reduced under these requirements and near-term shipment schedules will be impacted. However, based on the existing drum data and preliminary calculations, it is anticipated that only 10 percent of the drums are potentially flammable and 90 percent are safely shippable without treatment or repackaging. Costs associated with treatment or repackaging the waste for shipment will be substantial and flame testing, if required, will also have cost impacts. Preparing the waste for shipment and flame testing will both impact the scheduling of shipments to the WIPP facility. The flammability assessment methodology given in this paper is intended to increase the allowable concentrations of potentially flammable VOCs and thereby allow more drums to be shipped without treatment or repackaging, reduce the number of drums requiring flame testing, reduce costs, and minimize delays in waste shipments.

This paper describes an approach that includes investigating the potential flammability of TRU waste drums, increasing the allowable concentrations of flammable VOCs, and performing flammability assessments on drums. Estimates based on existing drum data indicate that when the proposed methodology is implemented, the number of drums that can be shipped without treatment or repackaging would be substantially increased. Because the methodology incorporates experimental data, validated predictive modeling of mixture lower explosive limits (MLEL) for the drums, and conservative screening of flammable gas concentrations, it is technically defensible and appropriate. The methodology takes into account the presence of all flammable gases in the wastes in determining acceptable screening limits for flammable gas concentrations in drums. Specifically, hydrogen and methane are considered in addition to VOCs.

Both empirical and theoretical models for predicting MLELs are considered. A model will be selected for use in flammability assessments that performs well relative to experimental data, adequately accounts for gas mixture compositions in drums, and demonstrates an acceptable level of conservatism. Screening limits are based on population statistics for flammable gas concentrations in the innermost layers of confinement in the drums, including the model-based maximum permissible hydrogen and methane concentration (MPHMC). The methodology for evaluating drums involves comparisons with screening limits, comparisons with less conservative drum-specific limits as necessary, and flame testing for drums that exceed these limits.

The MPHMC will be used to establish maximum allowable generation rates for hydrogen and methane gas. These rates, which will be calculated using the TRUPACT-II SARP methodology, must be determined because total concentrations of flammable VOCs may exceed the limit of 500 ppmv previously used in determining gas generation rates. The results could support preparation of an application to the Nuclear Regulatory Commission (NRC) to modify the current restrictive limits. This may

ultimately allow more waste to be shipped to the WIPP facility without repackaging or treatment.

This paper provides the technical basis for and description of the flammability assessment methodology as developed to date. First, a description of the testing to determine MLELs of TRU waste container gas mixtures is provided. Models that have been identified for assessing mixture flammability are described next. Following this, a description of the approach for determining the screening levels based on statistical analyses is presented. Finally, the steps of the flammability assessment methodology are given.

FLAMMABILITY TESTING PROGRAM

Considerable experimental data exist on the flammability of gas mixtures found in industrial and mining applications, such as mixtures comprised of hydrogen, methane, carbon monoxide, carbon dioxide, nitrogen, and oxygen. However, no experimental data are available for the types of gas mixtures observed in TRU waste containers. The objective of the flammability testing is to experimentally determine the lower explosive limit (LEL) for various TRU waste container gas mixtures.

The gas mixtures to be evaluated are composed of flammable and nonflammable VOCs listed in the Transuranic Waste Characterization Quality Assurance Program Plan (DOE 1995a) and hydrogen. To facilitate the experimental design, the flammable VOCs have been organized into functional and LEL groups. Functional groups for flammable VOCs are groups of VOCs with similar chemical structural characteristics. LEL groups for flammable VOCs are groups of VOCs with LELs that fall within a prescribed range of LEL values. The functional groups and associated functional group numbers (FGN) to be considered are aromatics (FGN = 1), ketones (FGN = 2), alcohols (FGN = 3), and alkanes/alkenes (FGN = 4). The LEL ranges and associated LEL group numbers (LGN) to be considered are 0.9%-1.3% (LGN = 1), 1.4%-2.6% (LGN = 2), and 5.6%-6.7% (LGN = 3). There is a general correlation between functional groups and groups based on LEL. That is, VOCs of a functional group tend to have LELs in a particular range. Table I summarizes flammable VOCs by functional and LEL groups.

Table I

The flammability testing will be done in two phases and will be based on a factorial experimental design that utilizes the LEL groupings. The factors are the presence or absence of each of three flammable VOC LEL groups in the mixture, the presence or absence of hydrogen in the mixture, and the presence or absence of nonflammable VOCs in the mixture. The objective of the first phase is to provide the necessary data to develop an empirical equation that expresses the flammable gas MLEL as a function of the concentration of each of the factors (i.e., concentrations of each of the three flammable VOC LEL groups, the concentration of hydrogen, and the concentration of nonflammable VOCs). The first phase will involve the testing of approximately 40 mixtures at ambient room temperatures (i.e., approximately 70F). These mixtures will be composed of varying ratios of 1,1-dichloroethane, methyl ethyl ketone, toluene, hydrogen, and carbon tetrachloride.

The objective of the second phase is to investigate the effects of VOC substitution within a group and to evaluate the effects of concentration on the predicted LEL. The second phase will involve the testing of an additional 30 mixtures, half of which would be tested at an elevated temperature of 146F. The second phase will also involve additional

chemicals, possibly acetone, benzene, cyclohexane, 1,1-dichloroethene, cis-1,2-dichloroethene, methanol, or methane, depending on the results of the first phase.

The results of the flammability testing will provide the empirical data needed to develop an empirical model for predicting flammable limits for mixtures of flammable gases in TRU waste containers. The results will also be used to evaluate the performance and adequacy of other predictive models.

MODELS FOR ASSESSING MIXTURE FLAMMABILITY

Four models for assessing mixture flammability have been identified through literature searches and discussions with flammability experts. The literature search identified documents that pertain to flammability limits and predictive methods. The search identified three theoretical models that are applicable to the problem. An empirical approach to modeling MLELs that would directly utilize experimental results was also identified. The four models being considered are an empirical model, the Le Chatelier rule, the group contribution method, and the adiabatic flame temperature method.

Empirical Model

The data obtained from the flammability testing will be used to develop an empirical model for predicting lower flammable limits for mixtures of VOCs and flammable inorganic gases in TRU waste containers. The empirical model is an equation that expresses the flammable gas MLEL as a function of the concentrations of each compound tested. The coefficients in the equation are obtained through standard least-squares statistical techniques and can be tested for their significant contribution towards predicting the MLEL. Experimental errors can be used to determine confidence limits for the predictions.

Le Chatelier's Rule

The Le Chatelier rule is an empirical equation developed by Le Chatelier in the late 19th century that enables the flammability limits of a mixture to be calculated if the flammability limits of individual components of a mixture are known. The effects of a few inert or nonflammable compounds (i.e., carbon dioxide and nitrogen) on the MLEL can be evaluated using a graphical method. The Le Chatelier rule has been tested for many mixtures that are important in transportation, industrial applications, and mining.

Group Contribution Method

The group contribution method provides an estimate of the flammability limits of a mixture based on knowledge of the chemical structure of each flammable compound in the mixture. The method does not account for the presence of inert (i.e., nonflammable) compounds that may be present in the mixture. Several group contribution methods have been proposed by various researchers (Shebeko et al. 1983; Season 1991; ASTM 1994; AIChE 1994) for estimating the LEL of individual compounds. However, no group contribution method has been proposed for mixtures of flammable gases. Based on an extension of the method for estimating the LEL of pure compounds (Procedure B) of the American Institute of Chemical Engineers (AIChE) Data Prediction Manual (AIChE 1994), the LEL was estimated for each of the gas mixtures and compared with the corresponding LEL estimated using the Le Chatelier rule. The absolute average error between the two methods was approximately 2 percent, with the group contribution method predicting a higher LEL in almost all cases.

Adiabatic Flame Temperature Method

The adiabatic flame temperature method is based on calculating and comparing the adiabatic flame temperature of a potentially flammable gas mixture with the critical or limiting adiabatic flame temperature. In the event of an explosion, energy is released by the combustion of the flammable compounds. Initially, the energy is absorbed by (1) unreacted reactants, (2) the combustion products, and (3) inert or nonflammable gases. Eventually, however, the energy will be dissipated from the system by various heat transfer processes. If a flammable gas mixture explodes in an adiabatic system (one in which there is no transfer of heat to or from the system), then it is possible to calculate an adiabatic flame temperature that corresponds to the temperature of the system after the explosion. The minimum temperature at which a flame can be sustained is referred to as the critical or limiting adiabatic flame temperature.

A number of computer codes are available to perform the complex thermodynamic chemical equilibrium calculations, including the American Society of Testing and Materials (ASTM) CHEETAH code (ASTM 1994), the National Aeronautic and Space Administration (NASA) Lewis Research Center CET93/CETPC code (McBride et al. 1994), the Lawrence Livermore National Laboratory (LLNL) CHEETAH code (Fried 1995), the University of Arizona CHEMEQ code (Wendt 1993), and the NASA CET93/CETPC code (NFPA 1988). If the adiabatic flame temperature of a potentially flammable gas mixture calculated by the code is above the critical or limiting flame temperature, then the mixture is flammable.

DETERMINING SCREENING LIMITS FOR FLAMMABLE GASES

A predictive model that performs well relative to experimental data, adequately accounts for gas mixture compositions in drums, and demonstrates an acceptable level of conservatism will be selected for use in determining drum-specific MLELs. The model will be selected from the models previously discussed. The MLELs predicted using the model will account for the presence of flammable VOCs, hydrogen, and methane in the drums. Because all flammable gases are included in the MLELs, the maximum concentration of hydrogen and methane that can be tolerated without the gas phase mixture becoming potentially flammable is the difference between the MLEL and the total concentration of flammable VOCs. This difference is the MPHMC.

To date, over 500 drums stored at the INEL and the RFETS have been sampled and analyzed for a specific suite of VOCs, hydrogen, and methane under the existing Transuranic Waste Characterization Program (TWCP). The samples were obtained from headspace gases under the drum lid with the rigid drum liner punctured. Additional drum sample analyses that are currently being gathered under the TWCP will be used in finalizing the flammability assessment methodology. The analyses will be used in determining screening limits for flammable gases.

The process for determining the screening limits for flammable gases is summarized in Fig. 1. First, the experimental and modeling work must be completed. The results will be used to choose a method for predicting MLELs. The chosen method will be used to obtain drum-specific MLELs using predicted innermost confinement layer headspace concentrations for existing drum data. The MLELs and the sums of flammable VOC innermost confinement layer concentrations will be used to compute drum-specific MPHMCs for existing drum data. Finally, screening values will be statistically determined for the sum of flammable VOCs and for MPHMCs.

Fig. 1

METHODOLOGY FOR ASSESSING FLAMMABILITY OF GAS MIXTURES IN TRU WASTE

Flammability assessments for individual drums are comprised of sequential evaluations, as necessary. The methodology consists of the following steps in evaluating individual waste drums for flammability (see Fig. 2):

1. Analyze drum headspace gas.
2. Predict innermost-layer headspace concentrations of flammable VOCs, hydrogen, and methane.
3. Sum innermost-layer concentrations of flammable VOCs and sum innermost-layer concentrations of hydrogen and methane.
4. Compare the flammable VOC sum with the flammable VOC screening limit and the hydrogen and methane sum to the MPHMC screening limit. If the both sums are below their associated limits, the drum may be shipped; if not, then go to step 5.
5. Compute the drum-specific MLEL, drum-specific MPHMC, and the sum of innermost concentrations of flammable VOCs, hydrogen, and methane.
6. Compare the flammable VOC, hydrogen, and methane sum with the drum-specific MLEL and compare the hydrogen and methane sum with the drum-specific MPHMC. If both sums are below the drum-specific limits, the drum may be shipped.
7. If at least one of the sums exceeds the respective limit, perform a flame test that simulates the gas mixture. If the flame test indicates that the mixture is not flammable, the drum may be shipped; otherwise, the waste must be either repackaged and reevaluated or another option must be taken, such as waste treatment.

Fig. 2

SUMMARY

The methodology may substantially increase permissible concentrations of flammable VOCs and consequently, the number of drums that can be shipped to the WIPP facility without treatment or repackaging. This increase would alleviate schedule impacts associated with near-term shipment of drums in accessible storage. The increase would also reduce schedule and cost impacts associated with waste repackaging or treatment. Furthermore, because the methodology incorporates experimental data, validated predictive modeling of actual drum lower explosive limits, and conservative screening of flammable gas concentrations, it is technically defensible and appropriate.

This work will be integrated with several other ongoing programs. Specifically, the results of the drum headspace gas sampling analysis being performed by the INEL and the RFETS under the TWCP will be used to develop the flammability screening limits. The results of the TRUPACT-II Gas Generation Testing Program and the TRUPACT-II Matrix Depletion Program will be used to establish new wattage limits by waste type. The TRUPACT-II compliance documentation will have to be revised to reflect the revised container wattage limits. Based on the revised wattage limits and the new methodology for determining flammability, an application for an amendment to the TRUPACT-II Certificate of Compliance will be submitted to the NRC for approval.

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7-8

TRANSFORMATION OF SOLID PHASES DURING THE STORAGE OF RADIOACTIVE PULPS IN TANKS

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ABSTRACT
One of the aspects of the problem connected with the treatment of heterogeneous wastes accumulated over the period of the operation of a radiochemical plutonium-producing enterprise is considered. The final goal of the work is the evacuation of radioactive wastes from the storage tanks and the decontamination of the wastes by removing long-lived radionuclides from the solid phase. The assessment of the phase composition of the precipitates is made using an X-ray powder diffraction analysis. The results obtained indicate that during the storage time of several decades, the wastes have undergone significant transformations.

INTRODUCTION

At present, the amount of various radioactive wastes, for which no decontamination or treatment technologies have been developed previously, is very large. As a rule, radioactive pulps were directed into tanks for a temporary storage. These tanks have a limited service life. In addition, the problems of treatment of the radioactive sludge (bottom residue) formed in surface storage reservoirs for liquid radwastes and pools for the storage and handling of fuel elements, remain unsolved.

The development of the strategy of treatment of these wastes involves both technological and ecological aspects, because the pulps and the sludges contain a large amount of long-lived radionuclides, primarily \7a\1-emitters. In addition, any method of treatment of radwastes should allow for the technological capabilities of any specific plant, the economic characteristics of the process, and accepted safety standards. The determination of the most efficient method of treatment will be possible, if true information about the state of the object to be studied is available. For the pulps, the data on the phase composition of the macrocomponents and the chemical forms of the existence of radionuclides by the time of the beginning of waste treatment are required. The information obtained thus far indicates that the precipitates found in the pulps and sludges undergo structural transformations involving both macro- and microcomponents.

This work is one of the stages of the development of a chemical technology of treatment of heterogeneous wastes and is devoted to a study of the compositions of the phases formed during a long-term storage of precipitates in tanks.

TECHNICAL DISCUSSION

Table I shows the results of the chemical analysis of two sludge samples taken from different tanks with a volume of 3000 m3, made of ferroconcrete and having the form of cylinders with diameter 12 m and height 30 m. These tanks are clad with 4-mm thick stainless steel and are installed in a rocky soil. Over 30 years, the first tank (sample 1) was used for the storage of mainly ferrocyanide precipitates, and second tank (sample 2) was used for the storage of hydroxide precipitates (iron hydroxides, aluminum, silica gel, etc.). Prior to analysis the samples were compacted by a 3-day setting.

Table I

The sludge samples (200 cm3) were taken through the hatch in the upper part of the tank, using a special sample-taking device.

As follows from the results of Table I, the solid phase of the sludges is characterized by a comparatively large concentration of a-emitting elements (uranium and plutonium) and a significant level of b-and g-activity.

The samples of the precipitates designed for X-ray powder diffraction studies were washed 8 - 10 times with hot distilled water, the samples were centrifuged, and the liquid phase containing water-soluble impurities was decanted. The solid phase was dried in air. Note that, as result of this washing, virtually all uranium was removed from the solid phase, which was confirmed by the spectral analysis data. The concentration of water-soluble impurities in the samples was monitored using an X-ray diffraction analysis. For this purpose, the X-ray diffraction patterns of the precipitates formed after the evaporation of the decantates were recorded. These X-ray diffraction patterns were compared with the X-ray diffraction patterns of the washed sludge samples. The latter did not show reflections characteristic of water-soluble compounds.

Table II shows the sets of interplane distances for the sludge samples investigated. The X-ray diffraction patterns of the sludges have a complicated character and indicate the presence of several crystalline phases in the precipitate. All X-ray diffraction patterns show an intense halo in the low-angle range, which is characteristic of amorphous phases.

Table II

The X-ray diffraction pattern of sample 1 indicates the absence of crystalline ferrocyanide phases in the precipitates.

On the basis of the chemical composition of the solid phases and the conditions of their storage (long-term action of high temperatures and fields of ionizing radiation), a phase possessing the set of interplane distances characteristic of the hematite mineral (a-Fe2O3) was identified (1). Note that this phase is present in the sludge samples taken from different tanks. The color of natural hematite changes from red-brown to black. The color of real sludges also changes in the same manner. Theoretically, the composition of the hematite mineral is described by the formula Fe2O3 powever, natural hematite always contains impurities. The presence of impurities in the hematite structure leads to a change in the crystal lattice parameters, which is observed in our case.

In addition, the X-ray diffraction pattern also shows the set of interplane distances consistent with the crystal structure of the takovite mineral (2). This mineral belongs to the group of hydrotalcite and has composition Ni5Al4O2(OH)10'6H2O. The takovite phase disappeared after the treatment of sample 1 with a solution of hydrochloric acid.

The set of interplane distances characteristic of hematite was identified for sample 2 after the subtraction of the reflections characteristic of the kyanite or disthene mineral (3). This mineral (Al2SiO5) belongs to the class of silicates. The reflection intensities indicate the prevalence of the hematite phase in the precipitate.

Table III

The X-ray powder diffraction, IR spectroscopic, chemical, and radiometric analyses that the sludges do not contain the phases that were initially disposed into the storage tanks. As a result of a long-term action of fields of ionizing radiation and, as a consequence, increased temperatures, the chemical forms found in this heterogeneous system underwent transformations. This was accompanied by the formation of M-OH-M and M-O-M bonds, the decomposition of the initial crystalline phases, and the formation of new crystalline phases.

The processes of the transformation of the solid phases are , to a certain extent, similar to those that take place during the formation of hydrogeneous-mineral deposits in a geological medium.

The transformation of the macrophase was accompanied by the transformation of the chemical forms of long-lived radionuclides, which is particularly important for plutonium present in both ionic and colloidal forms. Thus, the formation of mixed phases with the macrocomponents of the precipitate and an increase in the crystallinity of the plutonium hydroxide species were observed. The indirect evidence that similar processes do take place are the results of the experiments on the solvent extraction of plutonium with nitric acid containing hydroquinone (4). In the presence of hydroquinone, a simultaneous increase in the silicon and plutonium concentrations in the liquid phase was observed. Experiments on model systems were carried out, in order to refine the character of the process and to determine the effect of iron (II) ions.

The results of the study of the system composed of a mixture of silica gel and plutonium hydroxide after heating for 200 hours at 100o confirmed the assumption about a possible interaction between hydroxide, leading to the formation of a common phase, in the course of aging. Upon the subsequent treatment of the resulting precipitate with a nitric acid

solution of hydroquinone, Pu(IV) was reduced to Pu(III), plutonium compounds with silicon decomposed, and these elements passed to solution. The redox processes, which took place in the pulps, led to the formation of new phases with crystal structures of minerals such as hematite, kyanite etc. The properties of these phases in these heterogeneous systems somewhat differ from those of pure phases of the minerals identified. The chemical stabilities of these phases were determined, and methods for the selective dissolution of individual phases were developed.

CONCLUSIONS

The experiments on modeling the processes occurring during a long-term storage of sludges in tanks allowed us to determine the behavior of plutonium fission products and various valent forms of plutonium, including colloidal species, during the transformation of individual phases of the macrocomponents in these heterogeneous systems.

Based on these data, we developed a technology of the chemical decontamination of the sludges. With a minimal dissolution of the macrophase, this technology allows a high-efficiency removal of long-lived radionuclides from the sludges. As a result, the amount of secondary radwastes, both liquid and solid, decreases. The information about the structures of the precipitates at different stages of their treatment helps to optimize the process of the subsequent solidification of these phases.

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7-9

PRIMARY STUDY OF SEPARATION OF CESIUM FROM WASTE SOLUTION OF SPENT FUEL IN CHEMICAL PREDISPOSAL WITH CRYSTAL AMMONIUM MOLYBDPHOSPHATE (AMP) BEFORE VITRIFICATION

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ABSTRACT

Ammonium Molybdphosphate(AMP) is one of significant inorganic exchangers for separation of cesium from high level acid liquid waste (HLLW) (1,2), but the formerly prepared AMP were microcrystals, which was very difficult to apply in a column test. For the first time, we obtained granular crystal of AMP by taking advantage of equilibrium between PO43- and P2O74- in acid solution. The static ion exchange capacity for cesium was 0.63mmol/g and the distribution coefficient Kd was 350ml/g in 1 mol/l HNO3. The dynamic ion exchange capacity for cesium was 0.55mmol/g in simulated HLLW. After absorbing 2106 Gy dose from 60Co, the ion exchange properties did not change.The results of experiments indicated granular

AMP was one of the best chemical pretreatment material before vitrifying HLLW.

INTRODUCTION

In reprocessing a large amount of HLLW was produced. The Chinese HLLW which came from Purex process is a concentrated acid solution which contains fission products and has a very high salt content, especially nonradioactive nuclides such as Na, Al, Fe, Ni, and Cr. After a period of time in storage, the short lived nuclides decayed. The radioactivity comes from the long half-life nuclides in which the activity of 137Cs and 90Sr is 1.31012 and 9.31011 Bq/L. Table I gives the main composition of Chinese HLLW in a typical storage tank.

There are two ways for disposing of this type of HLLW, chemical treatment and vitrification. Chemical treatment could be divided into two steps. The first step is separating cesium and strontium. The second step is extracting transuranics from HLLW (3). After chemical treatment, the activity of the HLLW will be reduced 103 or more. The waste becomes a lower level radioactive waste stream that could be solidified in cement and disposed of in near surface with an engineered barrier. The TRU elements and 99Tc, which is only about 1% of salt content of original Chinese HLLW, can be transmuted to either stable or short half-life nuclides. They can also be incorporated into a glass matrix and disposed into a deep geologic repository. The separated 137Cs and 90Sr could be beneficially used or encapsulated for near surface storage.

Table I

The characteristic of chemical treatment are:

- the radioactive nuclides can be recovered and reused
- disposal processes such as ion exchange and extraction is easier
- the disposal cost is lower

The research work of inorganic ion exchanger for separating cesium-137 and strontium-90 was very extensive and mature. Some inorganic ion exchangers have good selectivity. When the waste solution is passed through the column, only cesium or strontium are absorbed. Although inorganic ion exchanger is an ideal material for recovering cesium and strontium, applying them is very difficult in a column test because they have some drawbacks and most of them are microcrystals. We synthesized granular crystals of AMP for recovering cesium, which solved the problem successfully. The preparation and behavior of granular AMP is described in this paper.

EXPERIMENT

Synthesis of AMP

Potassium pyrophosphate solution (0.025 mol/l in 1 mol/LHNO3) and ammonium molybdate solution(0.2 mol/l in 1 mol/LHNO3) are mixed at room temperature, in a volume ratio of 1:1. Aging about two weeks, yellow crystal precipitated and were separated with filter and dried at room temperature. After crushing up and sifting this crystal, the sample of AMP was obtained.

Analysis Composition and Structure of AMP

The content of NH4+ in sample was determined by decomposing NH4+ from AMP with 10% Sodium hydroxide solution and changing NH4+ to NH3, which was absorbed by standard sulfuric acid, then titrating this acid solution with standard alkali. The content of phosphor and molybdenum was determined by spectrophotometric method.

The structure analysis of AMP was determined by FT-IR Spectrometry (Nicolet 170SX U.S.A.), X-ray diffraction (Rigacu DMAX-2000 Japan), and Electron-microscope (T-605 Japan).

Determination of Ion Exchange Behavior of AMP Crystal

Took 0.1 g AMP and 20 ml 0.005 mol/l cesium or various ion nitrate (in 1 mol/l HNO_3) in small bottle, and shaked in thermostat at 20C. Determining the concentration of cesium in the solution at different time obtained the equilibrium curve and distribution coefficient Kd and static ion exchange capacity Q. The Kd and Q value are calculated by the following formula:

Eq. 1

Eq. 2

C0, C: concentration of original and equilibrium solution.

V: volume of equilibrium solution(ml).

M: weight of exchanger(g).

1.0g AMP was put in a column (diameter 0.36 cm, h:d is 4), operating temperature was 20C, flow rate was 1BV/hr. The feed solution was simulated Chinese HLLW.

RESULTS AND DISCUSSION

Composition of AMP Sample

The analytical results of sample were compared with theoretical value of $(NH_4)_3PMo_{12}O_{40}4H_2O$ as Table II shows.

Table II

The composition of AMP sample was in accordance with the theoretical composition of $(NH_4)_3PMo_{12}O_{40}4H_2O$ within an acceptable error.

Structure Analysis of AMP Sample

The IR and X-ray diffraction analysis of samples were in accordance with standard literature illustrations of $(NH_4)_3PMo_{12}O_{40}4H_2O$. The figures are omitted in this paper.

Composition analysis and structure analysis confirmed the sample was $(NH_4)_3PMo_{12}O_{40}4H_2O$. From the electron microscope photo (Fig. 1), the sample was cubic crystal.

Fig. 1

The Ion Exchange Behavior of AMP Crystal

The equilibrium curve of cesium was measured by static method with sampling at different times as shown in Fig. 2.

Fig. 2

From the curve the ion exchange equilibrium time of cesium was about 24 hours. The static ion exchange capacity Q was 0.63 mol/g, distribution coefficient Kd was 350 ml/g. The Q and Kd of various ions was determined by static method. The concentration of these ion was measured with ICAP instrument. The results as following Table III.

Table III

In dynamic tests of AMP sample with simulated Chinese HLLW, the concentration in the effluent vs effluent volume V(ml) gave the breakthrough curve as in Fig. 3. From the curve, the 1% breakthrough capacity was 0.51 mmol/g, 50% breakthrough capacity was 0.55 mmol/g.

Fig. 3

The radiolytic stability of the sample was tested with ^{60}Co, no significant change in exchange properties occurred until a total dose of 2106 Gy was absorbed.

CONCLUSION AND APPLICATION

Both the crystal AMP and microcrystal AMP have same behavior of ion exchange.

Crystal AMP is a good material for column testing.

AMP column removal of cesium-137 from HLLW could be a predisposal step for vitrification.

The crystal AMP will be a good material as an addition to vitrification, if the loaded AMP column does not elute cesium-137 in the process.

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7-11

THE NEPTUNIUM-PROBLEM: CORRELATION OF NP(IV), NP(V) AND NP(VI) WITH HNO2 IN NITRIC ACID SOLUTIONS - POSSIBLE ADJUSTMENT OF THE NEPTUNIUM-VALENCE-STATE IN AN ADVANCED PUREX-PROCESS FOR PARTITIONING AND TRANSMUTATION (P&T)

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ABSTRACT

One of the mostly striking problems of modern reprocessing based on the PUREX-Process is the uncontrolled spread of neptunium over the different product streams of a reprocessing plant, which is due to the rather complicated redox chemistry of this actinide in nitric acid solutions. According to the ideas of P&T, a strategy has been developed of quantitatively directing the neptunium into the fission product waste stream from which it can be isolated together with other actinides in order to provide them for future transmutation. Realizing this strategy the neptunium must be prevented from being coextracted together with uranium and plutonium in the first extraction cycle of the PUREX-Process. For that purpose the feasibility of a complete redox conversion of the element to unextractable Np(V) has been theoretically and experimentally investigated. As a first approach the behavior of neptunium in nitric acid systems of different acidities has been studied in order to gain more information about the initial valence-state distribution of the element after the dissolution of spent nuclear fuel. According to the results of these studies the major part of neptunium (80-90%) is already present as Np(V) at the end of the dissolution process. Because of the obvious influence of nitrous acid on the neptunium valence state equilibrium, a complete redox conversion of the actinide seems to be possible by controlling the concentration of nitrous acid in nitric acid solutions. First attempts started with a valence state composition as estimated for the feed solution of the PUREX-Process. They arrived at Np(V)-conversion rates of more than 94%. Thereby a new method of

controlling the concentration of nitrous acid has been applied having the advantage of excluding the production of any secondary waste. In order to avoid a later rearrangement of the neptunium-valence-state-equilibrium several reducing reagents have been tested for their ability of stabilizing Np(V). Some rather promising results have been obtained on employing urea and hydrazine. In the presence of these reagents the Np(V)-percentage could be enhanced to nearly 100%.

All chemicals used to treat the aqueous waste solution correspond to the CHNO-principle. According to that an increase in secondary waste is avoided by the use of reagents that are completely incinerable, producing only non-toxic gases and solutions such as CO_2, N_2 and H_2O.

INTRODUCTION

Several years ago the idea of Partitioning and Transmutation (P&T) was born, with the purpose of minimizing the long term risk of nuclear waste by a selective separation of long lived isotopes,actinides and fission products, and their later destruction by irradiation (1). As an objective the supporters of this vision wanted to achieve a drastic reduction of the radiotoxicity of HLW having in mind the reduced public acceptance of its long term storage. Due to the restricted possibilities of separation techniques as well as irradiation technologies in the past, the idea of P&T seemed to fail. Thanks to the rapid development of modern science, e.g. new accelerator driven neutron sources attaining rather high neutron flux densities, P&T has gained new importance in recent years (2). Thus new P&T-programs have been launched in several countries of the nuclear community in order to investigate either the necessary chemical separation techniques or suitable irradiation assemblies. Referring to the problem of Partitioning, many flow sheets have been developed and sometimes tested in cold or hot runs proposing different separation methods for actinides and fission products (3). Most of them are based on conventional reprocessing suggesting a combined isolation of transuranic elements after a complete removal of uranium and plutonium from the spent fuel. As far as these proposals for the waste handling employ liquid-liquid-extraction as a separation means, they normally refer to the PUREX-Process. Until today this process has been successfully implemented in many countries in order to recover uranium and plutonium from spent nuclear fuel and to make them available for further fuel fabrication (4). The extracting reagent used is Tri-n-butyl-phosphate (TBP), an organophosphorous reagent that exhibits good extraction properties towards tetra- and hexavalent actinides in nitric acid solutions. Thus, a nearly quantitative recovery of uranium and plutonium can be achieved. Unfortunately there is one other actinide - neptunium - which exhibits a similar redox chemistry and therefore is also accumulated in the first decontamination cycle (4).

Neptunium is mainly generated from U-238 by neutron capture during irradiation of fuel elements in nuclear reactors. On dissolving the spent fuel in nitric acid at the head end of the reprocessing plant the neptunium, initially present as NpO_2, is partly oxidized and an equilibrium of three valence states is established including Np(IV), Np(V) and Np(VI) (5). The extraction behavior of these three oxidation states is quite different. With respect to the extraction properties of TBP as described above, Np(IV) and Np(VI) are removed together with uranium and plutonium from the feed solution of the PUREX-process. In contrast to that hardly any reagent exists which can separate Np(V) from aqueous solutions. As a result the neptunium spreads over the whole

PUREX-process being finally contained in the uranium- and the plutonium-fraction as well as in the fission product waste stream.
Fig. 1
Regarding the radiotoxicological long term risk caused by neptunium, this uncontrolled spread is unacceptable. According to the requirements of a safe nuclear waste management and meeting the idea of P&T it has to be the objective of any basic research work aiming at a controlled recovery of actinides and fission products from spent nuclear fuel to develop a strategy of advanced reprocessing which will include a defined routing for neptunium during the process. This defined routing can either mean a coextraction of the element together with uranium and plutonium in the first extraction cycle of the PUREX-process or its prior transfer to the waste from which it could then be isolated together with the other transuranium elements providing them for further transmutation. Referring to the process-handling as well as the production of secondary waste, the latter proposal seems to be more favorable and thus has been chosen as the objective of our research work since about two years now.
Due to the fact that the extraction behavior of neptunium towards TBP fundamentally depends on the valence-state in which the actinide is present in the feed solution of the PUREX-process, a method of treatment had to be found that allows to control the redox-chemistry of the element in nitric acid solutions. Thus a strategy was investigated facilitating a total transfer of neptunium to Np(V) and thus hindering it from being extracted by TBP in the first extraction cycle of the reprocessing plant. In order to realize this idea, the composition of the neptunium valence-state distribution as well as its dependence on several influencing factors in nitric acid solutions were determined.
ADJUSTMENT OF THE NEPTUNIUM VALENCE STATE EQUILIBRIUIM AT THE HEAD END
In the Absence of Plutonium
In order to gain more information about the valence-state equilibrium of neptunium in the feed solution of the PUREX-Process the behavior of the actinide in the end of the dissolution of spent nuclear fuel in nitric acid was studied. For simplifying the first theoretical approach the presence of plutonium and iron in the spent fuel was omitted.
At the head-end of a reprocessing plant the irradiated fuel elements are dissambled, shorn and subsequently dissolved in nitric acid of varying molarities (6-2M) under slightly boiling conditions. In the course of this procedure the uranium, initially present as uranium dioxide (U(IV)), is oxidized to uranyl (U(VI)), which forms a water soluble nitrate-complex:

$$2\ UO_2 + 6\ HNO_3 \rightarrow 2\ UO_2(NO_3)_2 + NO_2 + NO + 3\ H_2O \quad (1)$$

As it can be seen from Eq. 1, nitric oxide and nitrogen dioxide are formed in the course of this reaction. In spite of the major part of NOx being released from the solution, the gas will be partly dissolved in the nitric acid leading to the formation of nitrous acid. Like the uranium, the neptunium will also be oxidized during the dissolution process following a reaction (see Eq. 2) quite similar to Eq. 1.

$$2\ NpO_2 + 6\ HNO_3 \rightarrow 2\ NpO_2(NO_3)_2 + NO_2 + NO + 3\ H_2O \quad (2)$$

In contrast to the quite stable uranyl the analogous neptunyl is subjected to a steady reduction mainly depending on the acidity as well as the amount of nitrous acid in the aqueous solution. Thus a valence-state equilibrium of Np(IV), Np(V) and Np(VI) is established.

$$2\ NpO_2^+ + NO_3^- + 3\ H^+ \rightarrow 2\ NpO_2^{2+} + HNO_2 + H_2O \quad (3)$$

$$2\ NpO_2^+ + 4\ H^+ \rightarrow Np^{4+} + NpO_2^{2+} + 2\ H_2O \quad (4)$$

According to literature (5) equation (3) summarizes a rather complicated oxidation-reduction-mechanism, in which the nitrous acid reveals the role of a catalyst. The overall redox reaction is known to be much faster than the disproportionation of Np(V) (see Eq. 4), which will be significant only at higher acidities and elevated temperatures (5,7).

Thus, the scale of the Np(V)/Np(VI)-ratio at the end of the dissolution of the spent fuel can be estimated by focussing on Eq. 3. For that purpose the quantitative development of nitrous acid during the dissolution of uranium dioxide was recorded in some preceding simulation experiments. It was found to shift from about 5E-4 mol/l at the beginning to about 3E-3 mol/l at the end of the process. After cooling the concentration of nitrous acid was reduced again yielding finally about 5E-5 mol/l. This last figure is estimated to correspond to the molarity in the feed solution of the liquid-liquid extraction where the acidity is adjusted to about 2 mol/l. Derived from the equilibrium constant of Eq. 3 the Np(V)/Np(VI)-ratio was calculated. According to this calculation, the oxidation state distribution of neptunium at the end of the dissolution process should reveal a clear excess of Np(V).

In order to compare the theoretical approach with experimental results, experiments were carried out under the same conditions as present at the end of the dissolution process. Thus the concentration of nitric acid was adjusted to 2 mol/l. At the same time the amount of nitrous acid in the aqueous solution was controlled by the introduction of nitric oxide. The neptunium added to the reaction vessel initially consisted of pure Np(IV) as expected to be contained in the nuclear fuel (NpO_2). Samples were taken after certain periods of time and were analyzed for their neptunium valence state composition.

Table I

The results of these determinations are depicted in Table I. According to them a relatively quick development of a valence state equilibrium was observed which could not be surveyed within the time range necessary for the off-line determination of the neptunium oxidation states. As expected this equilibrium revealed a maximum in Np(V) whereas the Np(IV)-portion was drastically reduced. With respect to the obtained results the valence state distribution seemed to depend on the nitrous acid concentration as well as on the temperature. This became evident on comparing the distribution of oxidation states determined in the reaction vessel with and without heating. At elevated temperatures the oxidation and disproportionation of Np(V) was obviously accelerated yielding an increased percentage of Np(IV) and Np(VI) (see Eq. 4).

In the Presence of Plutonium

With plutonium and iron present in the real nuclear fuel, the redox chemistry of neptunium in the dissolver as well as in the feed solution of the PUREX-Process might be basically influenced. In order to investigate this influence the experiments as described under a) were repeated in the presence of iron which was used to simulate also the presence of plutonium due to their similar redox potentials. As can be seen from Table I the Np(V)-percentage increased obviously reaching about 92% after cooling. Simultaneously the amount of Np(IV) and Np(VI) was reduced to 1,05% and 7,2% respectively. This unexpected comproportionation may be explained as follows:

In high molar nitric acid solutions plutonium and iron are likely to be oxidized to higher oxidation states (Pu(IV) and Fe(III)). As reported in (7, 8) these ions will oxidize Np(IV) yielding Np(V).

$Np^{4+} + Pu^{4+} + 2H_2O \rightarrow NpO_2^{+} + Pu^{3+} + 4H^{+}$ (5)

$Np^{4+} + Fe^{3+} + 2 H_2O \rightarrow NpO_2^{+} + Fe^{2+} + 4H^{+}$ (6)

On the other hand the reduced species Pu(III) and Fe(II) might cause a reduction of Np(VI) (see Eq. 8).

$NpO_2^{2+} + Fe^{2+} + 2 H_2O \rightarrow NpO_2^{+} + Fe^{3+}$ (7)

Thus, finally the presence of plutonium and iron leads to a comproportionation and further increase in Np(V) in the aqueous solution.

NEPTUNIUM VALENCE STATE COMPROPORTIONATION

As it has been shown, the valence state equilibrium of neptunium in nitric acid solutions strongly depends on the amount of nitrous acid present. In solutions with an acidity of not more than 2-3M an excess of Np(V) is to be expected. This excess is even enhanced in the presence of iron and plutonium. Thus, the idea of a complete comproportionation of the neptunium valence states to Np(V) seems to be realizable by a temporal limited control of the concentration of nitrous acid.

Once a complete comproportionation has been achieved, a rearrangement of the neptunium valence state equilibrium has to be excluded. This can only be realized by the application of a reducing reagent which will prevent any Np(V) from being reoxidized to Np(VI). Therefore our further efforts followed two different objectives.

Np Valence State Comproportionation by Controlling the HNO_2-concentration

The comproportionation experiments were carried out under similar conditions as the studies of the neptunium redox-behavior during the dissolution of spent fuel. Therefore a certain amount of neptunium was dissolved in 2M nitric acid. The valence state composition in the solution was adjusted to the distribution estimated for the feed solution of the solvent extraction according to our preceding investigations. Nitric oxide was bubbled through the reaction vessel in order to generate nitrous acid. The concentration of nitrous acid was controlled by temperature as determined in some preceding cold runs. Samples were taken from the solution in certain time-intervals and were analyzed for their composition of the neptunium valence state distribution.

Table II

Generally speaking the valence state conversion of neptunium has been successful even though the obtained Np(V)-percentages were not totally satisfying as yet. As depicted in Table II the measured conversion-rates were in a range of 80-94% in Np(V) depending on both the temperature (amount of nitrous acid) and the reaction time. A decrease of the Np(V)-percentage with increasing temperature was obvious. This was due on the one hand to the lower molarity of nitrous acid in the reaction vessel and on the other hand to the kinetics of the disproportionation reaction Eq. 5. Referring to the valence state distribution as obtained after different periods of time a maximum of Np(V)-comproportionation seemed to be attained after 1h. With proceeding reaction time the Np(V)-percentage was reduced. This reduction was mainly caused by a steady rise in nitric acid in the reaction vessel with a continued introduction of nitric oxide.

$2 NO + O_2 \rightarrow 2 NO_2$ (8)

$2 NO_2 \rightarrow N_2O_4 \quad DH > O$ (9)

$N_2O_4 + H_2O \rightarrow HNO_2 + HNO_3$ (10)

As can be seen from the Eq. 8-10 the immediate oxidation of nitric oxide in the oxygen-containing aqueous solution forms nitrogen dioxide (see Eq. 8) which will partly dimerize to dinitrogen tetraoxide (see Eq. 9). In the presence of water, the dimer itself disproportionates to nitrous acid

as well as nitric acid (see Eq. 10) and thus leads to a higher acidity. Caused by this rise in acidity the oxidation (see Eq. 3) and the disproportionation of Np(V) also increase (see Eq. 4), predominantly resulting in the formation of more Np(VI). If higher conversion-rates of neptunium shall be obtained the increase in acidity has to be avoided. This could be achieved either on working at lower concentrations of nitric acid or on reducing the hydrolysis of dinitrogen tetraoxide by controlling the reaction temperature.

Np Valence State Stabilization After the Valence-State Adjustment

With respect to the preceding results the neptunium valence state equilibrium is probably to be rearranged even if a comproportionation of the actinide has been achieved. To exclude any further oxidation of Np(V) several organic and inorganic reagents have been tested for their ability to stabilize Np(V) in nitric acid solutions due to their reducing character. All reagents used were in good correspondence to the CHNO-principle which proposes to employ only those materials for the treatment of nuclear waste that are completely incinerable in order to avoid any increase in secondary waste. Substances meeting this requirement should therefore contain only carbon, hydrogen, nitrogen and oxygen in their chemical structure.

The experiments started with an initial Np(V)-percentage of about 96%. The molarity of the nitric acid was varied from 1 to 3M. After the addition of the reducing reagent the composition of the neptunium valence state equilibrium in the solution was determined after certain periods of time. The promising results for two reagents that have been successfully tested (urea and hydrazine) are shown in Table III.

Table III

As expected they reveal an obvious increase in the Np(V)-percentage (99,7% in the maximum) due to the reduction of Np(VI). A further formation of Np(IV) could not be observed. Possible mechanisms for the reactions of urea and hydrazine with Np(VI) are presented by Eq. 11 and Eq. 12.

$$CO(NH_2)_2 + 6\ NpO_2^{2+} + H_2O \rightarrow CO_2 + N_2 + 6\ NpO_2^{+} + 6\ H^{+} \quad (11)$$

$$N_2H_4 + 4\ NpO_2^{2+} \rightarrow 4\ NpO_2^{+} + N_2 + 4\ H^{+} \quad (12)$$

Evidence for these mechanisms has been given by the following observation: According to Eq. 11 and Eq. 12 the reduction of Np(VI) yields a plain decrease in pH. Thus, on the other hand, the reduction rate should decrease with rising acidity. By comparing the reduction rates obtained at different concentrations of nitric acid as revealed in Table III, this correlation seems to be proved.

CONCLUSION

With respect to our studies on the behavior of neptunium during the dissolution of spent nuclear fuel, the major part of the actinide will be present as Np(V) in the feed solution of the PUREX-Process. Thus, a complete conversion of the element to its unextractable pentavalence state seems to be possible by taking advantage of the close correlation between the neptunium valence state equilibrium and the nitrous acid formed in the aqueous solution. The conversion rates obtained in some preliminary tests attained a maximum Np(V)-percentage of 94,2%. Thus, the general feasibility of a neptunium comproportionation could be shown. A clear dependence of the valence state conversion on thermodynamic and kinetic influences makes us believe that an optimization of these results can be achieved in the future. Even if a quantitative formation of Np(V) has not been achievable yet due to the ambivalent character of nitrous

acid, this might be realized by the application of some reducing reagents which exhibit stabilizing as well as enhancing features towards Np(V) in nitric acid. The most promising results have been obtained by employing urea or hydrazine as a reductant yielding a Np(V)-percentage of nearly 100%.

Although a complete comproportionation of neptunium in the feed solution of the PUREX-Process seems to be possible, some further research work is still required as several questions have not yet been answered. One of the most important aspects in this context will be the influence of iron and plutonium in the real feed. According to our theoretical as well as experimental studies on the dissolution process, their presence will initially cause a larger amount of Np(V). But it has to be kept in mind that any manipulation of the neptunium valence state equilibrium will also effect the redox chemistry of these two elements. Especially the use of strongly reducing reagents could lead to a disadvantageous reduction of Pu(IV) to the unextractable Pu(III) as it is known for hydrazine (7). Thus the coextraction of uranium and plutonium in the first decontamination cycle of the liquid-liquid extraction would be disturbed. Furthermore the possibility of a later separation of neptunium from the HLW has to be investigated with respect to the final goal of P&T. Therefore an oxidation of the actinide is proposed, aiming at a complete conversion to the extractable species Np(VI).

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INVESTIGATING THE FEASIBILITY OF USING OXALIC ACID TO CLEAN CARBON STEEL HIGH-LEVEL WASTE TANKS

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ABSTRACT
As a result of former reprocessing operations, two types of high-level radioactive waste (HLW) were generated at the Western New York Nuclear Service Center (WNYNSC) located in West Valley, NY. These were plutonium uranium extraction (PUREX) waste and thorium extraction (THOREX) waste. In 1980, the West Valley Demonstration Project (WVDP) Act was passed authorizing the Department of Energy (DOE) to conduct a HLW management project at the WNYNSC. As part of the Act, the approximate two million liters of HLW generated are to be processed and solidified by vitrification.
As a result of vitrification operations, residual amounts of the HLW described above and cesium (Cs)-loaded zeolite, a high-activity ion exchange waste stream generated from vitrification pretreatment operations, will be present in underground carbon steel tanks located in an on-site Tank Farm. Research is currently being conducted at the Pacific Northwest National Laboratory (PNNL) on the use of oxalic acid, which has been proposed to facilitate the removal of residual amounts of these high-level and high-activity wastes.*
Laboratory-scale tests performed by PNNL have been used to determine the optimum conditions in terms of acid-to-waste ratio, contact time, and the temperature of acid. In addition, the effects of multiple contacts, long-term contacts, the presence of corrosion products, the lack of agitation, and the temperature of tank contents were evaluated. The objectives of this research are to maximize the removal of the residual wastes from the carbon steel tanks, minimize the dissolution of iron, minimize corrosion, and provide a feed that is compatible with the current vitrification waste form specification.
INTRODUCTION
Approximately two million liters of HLW were generated during 1966 - 1972 at a commercial nuclear fuel reprocessing plant located at the WNYNSC in West Valley, NY. The waste, which consisted of alkaline sludge, alkaline supernate, and acidic liquid, was placed into two tanks located on the site for storage.
The largest volume fraction of waste was generated during reprocessing of spent uranium fuel using the PUREX process. Waste from the PUREX process was neutralized with sodiumhydroxide (NaOH) for storage in a carbon steel tank designated 8D-2. Neutralization resulted in a precipitated hydroxide sludge that settled to the bottom of the tank and a supernatant salt solution. The acidic liquid originated from the processing of thorium (Th) fuel from the Indian Point I reactor using the THOREX process. This waste was stored in a stainless steel tank designated 8D-4. Additionally, Cs-loaded zeolite, a high-activity waste, was generated from the WVDP pretreatment operations, which used zeolite in an ion-exchange process

designed to remove radioactive cesium from liquid HLW. The ion exchange columns and the spent zeolite are stored in a carbon steel tank designated 8D-1.

In preparation for vitrification operations, the spent zeolite is being removed from Tank 8D-1 and transferred to Tank 8D-2. Residual Cs-loaded zeolite, however, will be left behind. As a result of vitrification operations, a residual mixture of PUREX waste, THOREX waste, and spent Cs-loaded zeolite will remain in Tank 8D-2. The decision to facilitate removal of residual wastes by eluting the radioactive Cs from zeolite and dissolving the sludge, which will result in a transfer of the radionuclides to the Vitrification Plant and a decrease in the radioactivity remaining in the tank, is being evaluated as part of the stabilization objectives for these tanks.

Oxalic acid has been studied as a decontamination reagent for nuclear reactors and equipment. It may also be used to elute Cs from zeolite and to dissolve sludge. Previous laboratory studies show that oxalic acid can be used to elute Cs from zeolite at a level of 88-90% over a period of 11 days when 0.8 molar (M) oxalic acid is added to zeolite at ~20 liters (L) acid/kilogram (kg) zeolite.** These studies have also found that a maximum of 66 weight % (wt%) of washed sludge can be dissolved using ~0.7 moles of oxalic acid per unit of washed sludge (equal to one liter of PUREX sludge and supernate plus THOREX precipitated waste). In addition, it has been reported that over 95% of the sludge from the Savannah River Plant's Tank 16H was dissolved using 8 wt% oxalic acid at 85C with agitation in a two-step dissolution process (50 hours per step) with an initial reagent-to-sludge volume of 20:1.

The purpose of this paper is to discuss the research that is being conducted at PNNL. The results from a series of experiments that vary the strength of oxalic acid, the number of contacts, and the duration of contacts will be discussed in terms of the following primary objectives: maximize the removal of HLW from the carbon steel tanks, minimize the dissolution of iron, minimize tank corrosion, and provide a feed that is compatible with the current vitrification waste form specification.

CESIUM ELUTION EXPERIMENTS

These Cs elution experiments were performed in 500 milliliter (ml) Pyrex reaction vessels, with ports for thermocouples, stir shafts, and sampling, as shown in Fig. 1. Nonradioactive Cs-loaded zeolite was placed in the bottom of the reaction vessel and covered with 10 M NaOH to simulate the basic environment in Tank 8D-1. The 8 wt% oxalic acid was then added. The vessel was sealed using vacuum grease and placed in a heating mantle with a temperature controller programmed to hold the temperature at 50C (unless otherwise noted), which is the expected temperature of the material in Tank 8D-1 . A stirring shaft was placed in each vessel and the contents of the vessel were agitated throughout the test period (unless otherwise noted) to simulate the effects of the mobilization pumps. Tests were run for 50 hours (unless otherwise noted) and sampling was performed at 2, 4, 20, 28, and 50 hours of contact time (unless otherwise noted). When sampling was being performed, agitation of the material in the vessel was stopped and the material was first allowed to settle for five minutes. After five minutes, a 5 ml sample was drawn from the supernate and analyzed by Inductively Coupled Plasma-Mass Spectroscopy (ICP-MS) for Cs content.

Fig. 1

Some of the Cs elution experiments performed included: 1) determination of optimum process conditions, 2) multiple contact tests, and 3) tests to determine the effects of additional conditions such as the presence of corrosion products and carbon steel coupons, no agitation, and the temperature of tank contents.

The determination of the optimum process conditions for Cs elution from zeolite involved six tests. The variables in this experiment were the ratio of oxalic acid to zeolite, the temperature of the acid, and the contact time.

The multiple contact experiment involved four tests. Using the optimum process conditions previously determined, Cs-loaded zeolite was contacted with fresh oxalic acid three times for two hours. Two-hour contact times were chosen because previous experiments indicated that the majority of the Cs was eluted after two hours. After each contact, a sample of approximately 5 ml of supernate was drawn from each reaction vessel for analysis and the rest of the supernate was removed from the vessel. Two types of tests were performed for this experiment, one involving a water rinse of the zeolite after each contact and one with no water rinse.

The effects of several additional conditions on Cs elution from zeolite at the optimum process conditions were also evaluated. The presence of corrosion products in Tank 8D-1 and their effect on Cs elution was examined by adding iron oxide (Fe2O3) as a simulant. An amount of Fe2O3 equivalent to 2400 kg of rust at the bottom of the tank was added to the vessel.

The effect of carbon steel coupons on Cs elution was examined. Two carbon steel coupons were pre-weighed and suspended inside the reaction vessel so that they were completely submerged in the oxalic acid. The pair of coupons used for this test were prepared from larger U-bend coupons from earlier West Valley Nuclear Services (WVNS) corrosion tests. One of the coupons, after being measured and weighed, was left "as received" with a rusted layer on the surface. The other coupon was cleaned in an inhibited acid cleaning solution to remove the rust layer. Once cleaned the "clean" coupon was measured and weighed. When the test was complete, the coupons were removed from the vessel, rinsed in deionized water, dried at room temperature, and then re-weighed to determine weight loss and estimate corrosion rates.

The effect of no agitation was examined using a vessel set up in the usual manner, but without a stir shaft. The effect of a 25C bath temperature over 50 hours, rather than the 50C bath temperature, was examined. The 25C bath temperature reflects conditions under which the tank contents are at ambient temperature; while the 50C bath temperature reflects the effects of both radiolytic decay heat and heat from the pumps used to agitate the tank contents. The final test for this set of experiments was a control run at optimum process conditions. In addition to ICP-MS analysis for Cs concentration, the samples in this set of tests were also analyzed for iron (Fe) and aluminum (Al), which are the major oxide components of the zeolite.

SLUDGE DISSOLUTION EXPERIMENTS

As with Cs elution, the sludge dissolution experiments were run in 500 ml Pyrex reaction vessels, with ports for thermocouples, stir shafts, and sampling, as shown in Fig. 1. An appropriate amount of sludge simulant was added to the reaction vessel, followed by an appropriate amount of 8 wt% oxalic acid. The vessel was then sealed using vacuum grease and placed in a heating mantle with a temperature controller programmed to

hold the temperature at 50C (unless otherwise noted), which is the expected temperature of the material in Tank 8D-2. A stirring shaft was placed in each vessel and the contents of the vessel were agitated throughout the test period (unless otherwise noted) to simulate the effects of pumps in Tank 8D-2. Tests were run for 50 hours (unless otherwise noted) and sampling was performed at 2, 4, 20, 28, and 50 hours of contact time (unless otherwise noted). When sampling was being performed, agitation of the material in the vessel was stopped and the material was first allowed to settle for five minutes. After five minutes, approximately 5 ml were drawn from the supernate and analyzed by ICP-MS. In some of the experiments, the supernate was also analyzed by Ion Chromatography (IC) for oxalate concentration. At the completion of most of the experiments, the solids were separated from the supernate by vacuum filtration then dried and weighed to determine the total solids dissolved during the experiment.

The determination of the optimum process conditions for sludge dissolution involved six tests. The variables in this experiment were the ratio of oxalic acid to sludge, the temperature of the acid, and the contact time.

The multiple contact experiment involved two tests. Using the optimum process conditions previously determined, sludge simulant was contacted with fresh oxalic acid three times for two hours. After each contact, a 5 ml sample of supernate was drawn from each reaction vessel and sent for analysis. Two types of tests were performed: in the first test the oxalic acid in the vessel from each contact was removed after the contact and fresh oxalic acid was added; in the second test the oxalic acid was not removed after each contact.

The effects of several additional conditions on sludge dissolution at the optimum process conditions were also evaluated. The effect of the presence of corrosion products in Tank 8D-2 was examined by adding Fe2O3 as a simulant. An amount of Fe2O3 equivalent to 2400 kg of rust at the bottom of the tank was added to the vessel.

The effect of the presence of carbon steel coupons was examined. Two steel coupons were pre-weighed and suspended inside the reaction vessel so that they were completely submerged in the oxalic acid. The pair of coupons used for this test were prepared from larger U-bend coupons from earlier WVNS corrosion tests. One of the coupons, after being measured and weighed, was left "as received" with a rusted layer on the surface. The other coupon was cleaned in an inhibited acid cleaning solution to remove the rust layer. Once cleaned the "clean" coupon was measured and weighed. When the test was complete, the coupons were removed from the vessel, rinsed in deionized water, dried at room temperature, and then re-weighed to determine weight loss and estimate corrosion rates.

The effect of no agitation was examined using a vessel set up in the usual manner, but without a stir shaft. The effect of a 25C bath temperature over 50 hours, rather than the 50C bath temperature was examined. The final test for this set of experiments was a control that was run using the optimum process conditions.

In addition to the experiments described above, at the completion of the third, fourth, and fifth experiments, a determination of the solids remaining was made. At the completion of each of these experiments, most of the supernate was drained off and collected. The remaining material was poured through a vacuum filtration apparatus and the solids were collected on a piece of filter paper of known weight. The solids were

dried at room temperature for 24 hours and then weighed. Percent solids dissolved during the experiment could then be calculated using the total weight of simulant used in the experiment and an initial weight percent of 21.24% solids, which includes solids that were dissolved in the supernate as well as undissolved solids.

CESIUM ELUTION RESULTS

The experiments performed show that for Cs elution no one set of parameters tested resulted in a distinguishably higher efficiency, however, a multiple contact process appears to provide higher percentages of elution compared to a single contact. Addition of corrosion products, the presence of carbon steel coupons, and the lack of agitation all produced approximately the same level of Cs elution, which was no different from the control, after 50 hours of contact with oxalic acid. Cs elution was decreased slightly, however, when a 25C bath was used instead of the 50C bath used in most of the tests. Although the addition of Fe_2O_3 did not appear to have a significant effect on the elution of Cs, almost all of the Fe_2O_3 added to the system was dissolved. These results suggest that rust present in the tank may be extremely soluble in oxalic acid. Other tests in which carbon steel coupons were present suggest that oxalic acid may be corrosive to carbon steel since higher amounts of Fe appeared in solution than would be expected from the zeolite.

Determination of the optimum process conditions for elution of Cs from zeolite involved six tests. The variables examined were: 1) the amount of acid (20, 40, or 80 L of 8 wt% oxalic acid per kg of Cs-loaded zeolite) 2) the initial temperature of the added oxalic acid (25C or 80C) and 3) the total contact time (2, 4, 20, 28, or 50 hours).

The results of this set of tests are summarized in Fig. 2. As Fig. 2 shows, no one set of conditions consistently produced the highest elution level at each contact time. Cs elution ranged from a minimum of 52.2% (at 40 L acid/kg zeolite, 25C acid addition, 2 hour contact time) to 90.4% (at 20 L acid/kg zeolite, 80C acid addition, 50 hour contact time). In the interest of keeping plant operations simple (i.e., heating the oxalic acid to 80C would add difficulty to the process and may not significantly improve the Cs elution) and minimizing waste, 40 L acid/kg zeolite added at 25C were chosen as the optimum process conditions. The majority of the Cs elution appears to occur during the first two hours of contact, with ~60% of the Cs eluted in this time, therefore a two hour contact time was chosen as optimum for the laboratory-scale tests. A contact time this short, however, may not be feasible in full scale operations.

Fig. 2

In the multiple contact experiment Cs-loaded zeolite was contacted with fresh oxalic acid three times using the conditions of 40 L 8 wt% oxalic acid/kg zeolite, 25C acid addition, and three two-hour contact times. Two hour contacts were used because it was previously identified that >50% of the Cs was eluted after two hours contact time. Two types of tests were performed; one involving a water rinse after each contact, and one with no water rinse.

The results of the multiple contact tests show that close to 100% Cs elution was observed using two or three consecutive two hour contacts with oxalic acid. A water rinse did not appear to significantly improve or hamper Cs elution.

The effects of several additional conditions on Cs elution at the optimum process conditions of 40 L 8 wt% oxalic acid/kg zeolite added at 25C were

also evaluated. The additional conditions were: 1) addition of Fe2O3 to simulate corrosion products (rust) in the tank 2) addition of carbon steel coupons (one pre-corroded, one clean) to simulate the sides of the tank and to obtain preliminary carbon steel corrosion data 3) no agitation and 4) a 25C bath temperature instead of 50C. The supernate samples obtained from these tests were analyzed for Cs concentrations as well as concentration of Fe and Al.

After 50 hours contact time, all conditions except the 25C bath resulted in approximately the same amount of Cs eluted, an average of 90.6%, with a standard deviation of 2.5%. As would be expected, the 25C bath condition resulted in lower levels of Cs elution at longer contact times; however, the difference is within experimental error. The no-stir condition showed relatively low levels of Cs eluted at short contact times, but by 20 hours contact time, the Cs elution level had risen from 29.1% to 78.2%, and by 50 hours it was at 88.4%. Neither the addition of Fe2O3 nor the presence of carbon steel coupons appeared to have any effect on Cs elution at contact times greater than 2 hours.

The addition of Fe2O3 to simulate corrosion products in the tank did not appear to have a significant effect on the elution of Cs from zeolite or on the dissolution of Al in zeolite. It does, however, have an effect on the amount of Fe present in solution. In Fig. 3, which shows the amounts of Fe dissolved, 100% Fe dissolved means 100% of the Fe in the system. For the test in which Fe2O3 was added, this includes the Fe coming from the Fe2O3. Hence, although this test and the control both appear to have dissolved the same percentage of Fe, the total amount of Fe in solution is much higher for this test. This additional Fe may increase the number of glass logs produced by vitrification. The data indicate that 92% of the Fe in the system--including loose rust at the bottom of the tank--is in solution after 50 hours contact time.

It is interesting to note that it appears to take some time for the Fe concentration in solution to rise. At 20 hours contact time, only about 5% of the Fe in the system is in solution (corresponding to about the amount one would expect to come from the dissolution of Fe from the zeolite). By 28 hours, this amount had risen to 63%.

The presence of carbon steel coupons in the oxalic acid solution had no significant effect on either the elution of Cs from zeolite or on the dissolution of aluminum oxide (Al2O3) in zeolite. Figure 3 shows amounts of Fe greater than 100% in solution at longer contact times. In this test, 100% Fe is based on the amount of Fe in the zeolite; therefore, amounts greater than 100% would have to be attributed to Fe dissolved from the carbon steel coupons. The control test indicates that at 50 hours contact time, about 91% of the Fe from the zeolite is in solution; this suggests that amounts of Fe above the 91% in the coupon test are probably coming from the dissolution of the coupons themselves.

Fig. 3

A lack of agitation in the reaction vessel contents only appeared to affect Cs elution at short contact times. At 2 and 4 hours, 15.1% and 29.1% of the Cs was eluted, respectively, but after 20 hours, the amount of Cs eluted increased to 78.2% and by 50 hours had reached the same level as the control. Similar effects were observed in the dissolution of Fe and Al from the zeolite. These results suggest that when contact times are long (i.e., greater than 28 hours), a lack of agitation should not be detrimental to Cs elution. Cs elution, however, may be limited by

diffusion if the zeolite is in a pile and thus has less surface area exposed to the oxalic acid.

The 25C bath temperature resulted in lower levels of Cs elution at most contact times. This suggests that the temperature of the system could have an impact on the efficiency of Cs elution operations. If tank temperatures fall below the 50C used in these tests, it may be necessary to extend the duration of contact times and/or implement multiple contact treatments.

SLUDGE DISSOLUTION RESULTS

The experiments performed show that for sludge dissolution no one set of parameters tested resulted in a distinguishably higher efficiency, however, multiple contacts were not found to be advantageous. In addition, contact time did not appear to be a factor in sludge dissolution. Significant changes in the percentage of each ion dissolved were not observed after 4 hours. As with Cs elution, the addition of corrosion products, presence of carbon steel coupons, and the lack of agitation did not have a significant effect on sludge dissolution. Although the addition of Fe2O3 did not appear to have a significant effect on the dissolution of sludge, almost all of the Fe2O3 added to the system ended up in solution. These results suggest that rust present in the tank may be extremely soluble in oxalic acid. Other tests in which carbon steel coupons were present suggest that oxalic acid may be corrosive to carbon steel, since higher amounts of Fe appeared in solution than would be expected from the sludge.

Determination of the optimum process conditions for sludge simulant dissolution in 8 wt%oxalic acid consisted of six tests. The variables studied were: 1) the ratio of oxalic acid to sludge simulant (20, 40, and 80 L acid/kg sludge simulant) 2) the temperature of the acid as it was added (25C or 80C) and 3) the contact time (2, 4, 20, 28, or 50 hours). The concentration of soluble oxalate was measured by ion chromatography (IC) after each contact time in each of the six tests. Throughout the experiment, changes in the oxalate concentration were found to be less than the analytical error of 10%. These results suggest that within experimental error, no precipitation of oxalate was detected.

The concentration of each ion in solution was measured by ICP-MS after each contact time in each of the six tests. Figure 4a shows the ranges of dissolution for each ion. In Fig. 4a each dark circle represents one data point and each data point for a given ion represents a different set of conditions (i.e., L acid/kg, temperature of acid addition, contact time). The analytical error associated with each data point is 10%. For most ions, a number of data points lie within the experimental error of the maximum dissolution, making it difficult to state with any certainty that one particular set of conditions was the most effective for a given ion. However, it was possible to observe trends in the data. Upon examination of the results obtained, it becomes apparent that 20 L acid/kg sludge and 25C acid addition seem to lead to lower dissolutions, while 40 or 80 L acid/kg sludge and 80C acid addition seem, in general, to lead to higher dissolutions. Contact time did not appear to play a major role in the dissolution of most of the ions; significant changes were not observed after 4 hours contact time and, in most cases, at least 70% of the ion had dissolved after 2 hours contact time.

For most of the ions analyzed (zirconium (Zr) and cerium (Ce) being the only exceptions), maximum dissolutions were found to be over 70%. The two high data points for Ce dissolution occur at 40 L acid/kg sludge and 25C

acid addition, at 28 and 50 hours contact time. Since Ce was chosen as a surrogate for the radionuclides uranium (U) and thorium (Th), the low dissolution of Ce from the simulant is a cause for some concern. The ICP-MS measurements of the initial simulant failed to detect more than 4% of the amount added to the simulant. Therefore, it is entirely possible that these low percentages are due to the inability of the ICP-MS to detect Ce under these conditions. It is also possible that Ce simply does not dissolve, or if it does dissolve, it may form a cerium oxalate type of compound and thus not be found in solution. It is difficult to say, based on these results, whether or not Ce was a good choice for a U and Th surrogate. These results may suggest that U and Th will not dissolve well under the conditions tested or they may simply be an indication that Ce does not dissolve well under the conditions tested.

Optimum process conditions for sludge dissolution were chosen based on the results of these six tests and on the ease of use during plant operation. The optimum process conditions chosen were 40 L 8 wt% oxalic acid/kg sludge added at 25C. Forty L 8 wt% acid/kg sludge was chosen over 80 L 8 wt% acid/kg sludge in the interest of waste minimization and because 80 L acid/kg sludge was not significantly better than 40. Twenty-five degrees Celsius acid addition was chosen because 80C acid addition did not uniformly increase dissolution and in cases where it did increase dissolution, the increase was on the order of a few percent. In addition, heating the acid would present additional operational requirements in full scale operations.

In the multiple contact experiment, sludge simulant was contacted with oxalic acid three times using the conditions of 40 L 8 wt% oxalic acid/kg sludge, 25C acid addition, and a two hour contact time. Two hour contacts were chosen because the first experiment indicated that >70% of most of the ions had dissolved after two hours. Two tests were run: one in which the oxalic acid from each contact was removed from the reaction vessel before the next contact, and the other in which the fresh oxalic acid was added to the acid from previous contacts.

In the test where the oxalic acid was removed after each contact, most of the dissolution occurred during the first two hour contact. Additional contacts increased the total dissolution of ions by only a few percent. In the test where the "used" oxalic acid was left in the vessel, for many ions, after the initial contact, total dissolution appeared to decrease with subsequent contacts. The data indicate that multiple contacts with oxalic acid are not advantageous to the sludge dissolution process. The amounts of ions in solution remain constant within experimental error over the three contacts for both the case where the acid is removed after each contact and the case where fresh acid is added to that which is already present.

The effects of several additional conditions on sludge dissolution at the optimum process conditions of 40 L 8 wt% oxalic acid/kg sludge added at 25C were also evaluated. These additional conditions were: 1) addition of Fe_2O_3 to simulate the presence of corrosion products (rust) in the tank 2) addition of carbon steel coupons (one pre-corroded, one clean) to simulate the sides of the tank and to yield preliminary corrosion data 3) no agitation and 4) a 25C bath temperature instead of 50C, as stated in the initial description of the sludge dissolution experiments.

The conditions in this experiment seemed to affect the dissolution of strontium (Sr), nickel (Ni), and Fe the most. In the case of Sr, the data shows that the 25C bath decreases the Sr dissolution to well below the

control level after 50 hours contact time. Nickel dissolution is dramatically decreased by the presence of steel coupons and, to a somewhat lesser extent, the dissolution of Ni is lower than the control at 50 hours under both the no-stir and the 25C bath temperature conditions.

Fig. 4a

Fig. 4b

The addition of Fe_2O_3 to simulate corrosion products in the tank did not appear to have a significant effect on the dissolution of the ions analyzed. However, it did have an effect on the amount of Fe in solution. In Fig. 4b, 100% Fe dissolved means 100% of the Fe added to the system. For the test in which Fe_2O_3 was added, this includes Fe from Fe_2O_3. Although this test and the control both appear to have dissolved the same percentage of Fe, the total amount of Fe in the system is much higher for the test in which Fe_2O_3 was added. The data indicate that 97% of the Fe in the system, including loose rust at the bottom of the tank, is in solution at 50 hours contact time.

The data show that the presence of carbon steel coupons in the oxalic acid solution had an effect on the dissolution of both Ni and Fe in the sludge dissolution experiments, but did not have a significant effect on the dissolution of any of the other ions present in the sludge simulant. The results from the tests in which carbon steel coupons are present show that the dissolution of Ni decreases dramatically with time (from 70% after 4 hours contact to 30% after 50 hours contact) when the coupons are present. A possible explanation for this is that Ni was plated out on the surface of the coupons. Analysis of the coupon surfaces and the stripping solution might confirm whether this is the cause of the dramatic decrease in Ni in solution when carbon steel coupons are present. If this is the case, it is possible that once the oxalic acid cleans the oxide off the tank wall other metals will also be reduced and plate out on the tank surfaces. These other metals might include radionuclides, which would not be a desirable situation.

Figure 4b shows that amounts of Fe greater than 100% are in solution at all contact times except 4 hours. In this test, 100% Fe is based on the amount of Fe in the sludge simulant alone; therefore, amounts greater than 100% would have to be attributed to Fe dissolved from the carbon steel coupons. The control test indicates that at 50 hours, approximately 96 wt% of the Fe from the simulant is in solution. This suggests that amounts of Fe above 96% 10% in the coupon tests are probably due to the dissolution of the coupons themselves.

A lack of agitation of the reaction vessel contents did not appear to have a significant effect on the dissolution of any of the ions analyzed in these tests except for Ni, which exhibited slightly lower dissolution than the control at longer contact times. A 25C bath temperature appeared to affect the dissolution of both Sr and Ni. Both of these ions exhibit somewhat lower dissolution than the control in a 25C bath.

The solids remaining at the end of most of the experiments were collected and weighed to determine the amount of solids that had been dissolved by treatment. An average of 74% (with a standard deviation of 2.6%) of the solids were dissolved in the tests where the determination was made. The remaining solid material from one of the tests was analyzed by X-ray diffraction. The material was found to be a complex mixture of phases. The main phases identified are listed in Table I along with a semi-quantitative wt %. Additional phases are likely to be present, but were

not identifiable, and the semi-quantitative analysis ignores the presence of additional unidentified phases. The presence of SiO_2 (quartz) and ZrO_2 (Baddeleyite) is not surprising since these are both difficult compounds to dissolve. Also, the large percentage of $C_6H_{20}Ce_2O_{22}$ (cerium oxalate hydroxide) present agrees with the ICP-MS analysis for Ce in the supernatant, which indicated that very little of the Ce initially present in the sludge had been dissolved.

Table I

CONCLUSIONS

The original objectives of Cs elution and sludge dissolution for which the experiments discussed above were conducted included maximizing Cs elution and sludge dissolution, and investigating the effect of tank corrosion products. In addition to these objectives, the test condition of oxalic acid wt % was set at 8 wt%. After reviewing the results obtained, however, a modified set of objectives was identified that added minimizing the dissolution of iron, minimizing tank corrosion, and providing a feed compatible with the current vitrification waste form specification to the objectives already defined. With this new set of objectives, additional Cs elution tests that used 4 and 1 wt% oxalic acid solutions and corrosion tests were performed.

The following conclusions are based on results of work presented in this report as well as some of the follow-up testing indicated above:

The data show that lower percentages of Cs may be eluted when the wt% of oxalic acid is lowered (i.e., from 8 wt% to 4 or 1 wt%). Using 8 wt% oxalic acid, as described above, a maximum of 99.9% Cs was eluted with an oxalic acid-to-zeolite ratio of 40 L/kg, a temperature of 25C, and three 2-hour contacts. Approximately 60-70 % Cs was eluted, however, using 4 and 1 wt% oxalic acid. (Note: A higher oxalic acid-to-zeolite ratio was used for the 4 and 1 wt% tests.)

The data show that by using 8 wt% oxalic acid, an oxalic acid-to-sludge ratio of 40 L/kg, an acid temperature of 25C, and a single contact, 70-75 wt% of the solids present in the sludge can be dissolved. The dissolution of the radionuclides may be considerably lower, <10%, based on the dissolution of Ce that was used as a surrogate for U and Th.

The data show that larger amounts of Fe_2O_3, which represents corrosion products present in the tank, are solubilized when 8 wt% oxalic acid solutions are used. By using 8 wt% oxalic acid with the conditions specified in the first bullet, ~ 90% of the Fe_2O_3 was solubilized compared to ~ 15-20 % solubilized for 4 and 1 wt%. (Note: A higher oxalic acid-to-zeolite ratio was used for the 4 and 1 wt% tests.)

Oxalic acid corrosive tests were performed using A516 Grade 55 mild steel test specimens, which has essentially the same chemical composition and microstructure as the type of steel used in fabricating the tanks.*** These tests showed that even at relatively modest temperatures (50C) and short exposures to 4 and 8 wt% oxalic acid solutions (1 to 3 weeks), the corrosion rate is quite high. The corrosion appeared as localized pitting and crevice corrosion for submerged test specimens; however, vapor space corrosion was negligible at the conditions tested.

The experiments performed and the results obtained have provided a basis for declaring that oxalic acid can elute Cs from zeolite and dissolve sludge. Questions and issues still exist, however, that need to be resolved before oxalic acid can be identified as being feasible to clean carbon steel HLW tanks. Some of the questions and issues, which were briefly mentioned in the conclusions stated above, are:

Iron present at too high a level is detrimental to vitrification. How can the amount of iron dissolved be minimized?

The goal in maximizing sludge dissolution relates to mobilizing the radionuclides present enabling them to be transferred to vitrification. If Ce is an accurate surrogate for U and Th, what impact will precipitated U and Th oxalate compounds have on this goal i.e., can they still be mobilized and transferred?

As stated above, the data show that oxalic acid is corrosive to the mild steel test specimens. For purposes of safety, how can corrosion to the tanks be minimized?

In support of the WVDP's effort to evaluate options to meet stabilization objectives for the HLW tanks on site, the investigation of the feasibility of using oxalic acid will continue. The basis of information obtained to date will be reviewed and used to determine the path forward that will enable the determination of feasibility to be made.

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DISPOSITION OF PUREX CONTAMINATED NITRIC ACID THE ROLE OF STAKEHOLDER INVOLVEMENT

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ABSTRACT

What does the United States space shuttle and the Hanford PUREX facility's contaminated nitric acid have in common. Both are reusable. The PUREX Transition Project has achieved success and minimized project expenses and waste generation by looking at excess chemicals not as waste but as reusable substitutes for commercially available raw materials. This philosophy has helped PUREX personnel to reuse or recycle more than 2.5 million pounds of excess chemicals, a portion of which is the slightly contaminated nitric acid. After extensive public review, the first shipment of contaminated acid was made in May 1995. Removal of the acid was completed on November 6, 1995 when the fiftieth shipment left the Hanford site. This activity, which avoided dispositioning the contaminated acid as a waste, generated significantly more public input and concern than was expected. One of the lessons learned from this process is to not underestimate public perceptions regarding the reuse of contaminated materials.

The quantity of radioactivity in the contaminated acid (each individual shipment) met the criteria for a low specific activity shipment under the Department of Transportation regulations. In the total volume of acid there was less than 0.3 grams (0.01 ounces) of plutonium and 7,400 kilograms (16,300 pounds) of uranium, of which approximately 72 kilograms (158 pounds) was U-235 (i.e. fissile material). Low specific activity shipments are routinely shipped across the country. The handling (including transportation) of hazardous liquid chemicals occurs daily in the continental United States. In 1993 alone, approximately 5.3 X 109 liters (1.4 billion gallons) of nitric acid was produced and transported internationally; without substantial incident. However, the transport and disposition of the PUREX contaminated acid generated more interest and comments than was expected for an activity of this type. In comparison, spent fuel shipments from PUREX generated little interest from reviewers.

PUREX DEACTIVATION

In December 1992, DOE directed that the PUREX Plant be shut down and deactivated because it was no longer needed to support the nation's weapons-grade plutonium production. The scope of the deactivation project involves many activities necessary to place the PUREX Plant in an environmentally safe and stable state for long-term surveillance and maintenance. Removing major hazards from the PUREX Plant, such as excess chemicals, spent fuel, and residual plutonium are major goals of the deactivation project. This will reduce the risk of exposure to both onsite workers and members of the general public. Stakeholder involvement has played a major role in the formation of deactivation plans, implementation of selected strategies and accomplishment of specific goals.

BACKGROUND INFORMATION

Historically, nitric acid was used at the Hanford Site's PUREX Plant to dissolve irradiated fuel elements, and for the separation and purification of uranium, plutonium, and neptunium in solvent extraction operations. The nitric acid was recovered and reused during processing. A specific result of cessation of PUREX Plant operations is that excess chemicals are available, including approximately 692,000 liters (183,000 gallons) of slightly radioactively contaminatednitric acid.

DEACTIVATION PLANNING

The original project baseline for disposition of the excess 10 molar nitric acid was to sugar denitrate the material to approximately 1 molar acid in the PUREX canyon. This process would generate between 300 to 400 metric tons of nitrogen oxides or NOx as gaseous effluent. Denitration would reduce the liquid volume to be transferred for disposal to tank farms by about 33 percent. This action eliminates the acid but provides no beneficial use for the material, while having potential present and future environmental impacts.

Westinghouse sought ways to beneficially use the material to avoid processing the acid as waste. Brainstorming sessions were held to seek innovative ways to use the material. With no use for the surplus acid identified within the DOE complex, private sector interest was solicited. An expression of interest was received from British Nuclear Fuels private limited company (BNF plc), the sole respondent to the Commerce Business Daily advertisement.

Abandoning the treatment option to sugar denitrate the acid and pursuing the beneficial reuse of the material, along with other changes to the project, resulted in saving $37 million and shortened the duration of the Deactivation project by 10 months. Beneficial reuse of the acid is the most economical and cost effective solution for disposition of the acid.

DOCUMENTATION

The concept of shipping the acid to England for use in a process similar to PUREX was previously addressed under the National Environmental Policy Act (NEPA) as a Categorical Exclusion or (CX). Nuclear proliferation became a potential concern with interest groups. Later DOE determined that the action would meet conditions of the regulations that require additional NEPA review. Therefore an Environmental Assessment was prepared to provide a quantitative analysis of potential risks and environmental impacts associated with the proposed action and alternatives, in the continental U.S. and on international waters, and to allow a determination of whether or not an Environmental Impact Statement is required.

Preparation, review, and approval of the Environmental Assessment took many months. An Ad Hoc stakeholder committee, consisting of representatives from three local interest groups, DOE, and Westinghouse, was formed to facilitate document preparation and review. The Ad Hoc stakeholder committee was used to improve the nitric acid Environmental Assessment and address many different opinions prior to issuing the document for comment. Subsequently, the draft document was sent to more than 200 individuals, states, Indian Nations, interest groups and affected public for public comment. Public meetings were held on the east coast at the three proposed shipping ports; Portsmouth, Virginia, Baltimore, Maryland and Newark, New Jersey. Figures 1 and 2 show the posters used at the public meetings. During the public comment period more than 50 inquiries for information, clarification, or comment were made. All comments were addressed in the final Environment Assessment. Comments included a wide range of topics and issues. Some comments were specific to the activity and some comments were unrelated to the project. Comments ranged from concerns about potential spread of contamination to proliferation issues. Comments on proliferation were addressed in the "Environmental Assessment, Disposition and Transportation of Surplus Radioactive Low Specific Activity Acid, Hanford Site, Richland, Washington," DOE/EA-1005, as follows:

In evaluating the nonproliferation policy aspects of the proposed shipment, DOE considered the facts that BNF plc has a readily available supply of nitric acid, which could be procured from any number of U.S. or other commercial sources, and that interested parties such as Ecology, U.S.Environmental Protection Agency (Region 10), Yakima Indian Nation, and the Confederated Tribes of the Umatilla Indian Reservation do not object to the shipments. In addition, the proposed shipment appeared to be a case-specific solution to a material disposition problem, promoting waste minimization and reducing potential emissions to the environment. The export would not make a material contribution to the proliferation of weapons of mass destruction and would be consistent with Executive Order 12114, Environmental Effects Abroad of Major Federal Actions. These facts appeared to support the position that the transfer of nitric acid from the PUREX Plant was a policy-neutral decision, and did not set a precedent from either a technical or policy standpoint.

Many comments were associated with the transport of the acid across the United States. Some states requested advance notice of shipments. The Department of Energy instituted weekly conference calls to keep states and other interested individuals informed on the status of the shipments. A Finding of No Significant Impact or (FONSI) was approved by the Hanford Site Manager in May 1995. Shortly there after the first nitric acid shipments to England were made. Figure 3 shows PUREX employees loading acid into the shipping container. The last shipment left the Hanford site on November 6, 1996. A total of 707,000 liters (187,000 gallons) of acid were shipped.

LESSONS LEARNED

The PUREX Transition Project team developed a good working relationship with state/federal regulators/stakeholders. Early and frequent contact with interest groups is key to project success. Flexibility is also key to success. Working directly with outside groups such as the Ad Hoc committee helped to produce an initial draft of the environmental assessment that was more palatable to the public. However, it is ironic that due to specific concerns, public comment on the disposition of the

slightly contaminated acid generated significantly more interest than the transfer of spent fuel. The lesson here is one of perceptions. In this case the greater risk was not the biggest concern. When addressing unique situations such as the disposition of PUREX contaminated acid, it is essential that "normal procedures" not be totally relied upon. Due to frequent interaction with the regulators and stakeholders deactivation project goals were accomplished without delay to field work.
Fig. 1a
Fig. 1b
Fig. 1c
Fig. 2a
Fig. 2b
Fig. 2c
Fig. 3

7-14
POSITION FOR DETERMINING GAS PHASE VOLATILE ORGANIC COMPOUND CONCENTRATIONS IN TRANSURANIC WASTE CONTAINERS

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ABSTRACT
In the conditional no-migration determination (NMD) for the test phase of the Waste Isolation Pilot Plant (WIPP), the U.S. Environmental Protection Agency (EPA) imposed certain conditions on the U.S.Department of Energy (DOE) regarding gas phase volatile organic compound (VOC) concentrations in the void space of transuranic (TRU) waste containers. Specifically, the EPA required the DOE to ensure that each waste container has no layer of confinement that contains flammable mixtures of gases or mixtures of gases that could become flammable when mixed with air. The EPA also required that sampling of the headspace of waste containers outside inner layers of confinement be representative of the entire void space of the container. The EPA stated that all layers of confinement in a container would have to be sampled until DOE can demonstrate to the EPA that sampling of all layers is either unnecessary or can be safely reduced. A test program was conducted at the Idaho National Engineering Laboratory (INEL) to demonstrate that the gas phase VOC concentration in the void space of each layer of confinement in vented drums can be estimated from measured drum headspace using a theoretical transport model and that sampling of each layer of confinement is unnecessary. This report summarizes the studies performed in the INEL test program and extends them for the purpose of developing a methodology for determining gas phase VOC concentrations in both vented and unvented TRU waste containers. The methodology specifies conditions under which waste drum headspace gases can be said to be representative of drum gases as a whole and describes a method for predicting drum concentrations in situations

where the headspace concentration is not representative. The methodology addresses the approach for determining the drum VOC gas content for two purposes: operational period drum handling and operational period no-migration calculations.

INTRODUCTION

Characterization of transuranic (TRU) wastes destined for the Waste Isolation Pilot Plant (WIPP) will include sampling of gases in the headspaces of waste drums for volatile organic compounds (VOC). The headspace gas VOC concentrations will be used to determine potential flammability of gases and VOC concentrations in drums for transportation and Resource Conservation and Recovery Act (RCRA) compliance purposes (DOE 1995). in the conditional no-migration determination (NMD) for the test phase of the WIPP facility (55 FR 47700), the Environmental Protection Agency (EPA) required that the U.S. Department of Energy (DOE) must ensure that each waste container emplaced underground at the WIPP has no layer of confinement that contains flammable mixtures of gases or mixtures of gases that could become flammable when mixed with air. For purposes of a no-migration demonstration, DOE must also characterize the nonflammable volatile hazardous constituents in TRU waste containers. The EPA stated that all layers of confinement in a container would have to be sampled until DOE can demonstrate to the EPA, based on the data collected, that sampling of all layers is either unnecessary or can be safely reduced. This position paper summarizes a methodology that has been proposed in Connolly et al. (1995) that addresses these conditions and alleviates the requirement of sampling all layers of confinement.

A test program was conducted at the Idaho National Engineering Laboratory (INEL) to demonstrate that the VOC concentration in the void space of each layer of confinement in vented drums can be estimated using measured drum headspace and a model incorporating theoretical diffusion and permeation transport principles and that sampling of each layer of confinement is unnecessary. The model and model validation results provide information that are used to address the headspace VOC concentration representativeness issue raised by the EPA for vented containers.

This report presents the methodology for determining VOC concentrations in both vented and unvented drums for existing waste packaging configurations. The methodology specifies conditions under which drum headspace gases can be said to be representative of drum gases as a whole and describes a method for predicting drum concentrations in situations where the headspace concentration is not representative. For the purpose of predictions of gas phase VOC concentrations, innermost layers of confinement are the polymer bag layers closest to the waste; rigid inner layers of confinement, such as polymer bottles, are not included. The methodology presented in this paper addresses the approach for determining the drum VOC gas content for two purposes: operational period drum handling and operational period no-migration calculations.

VOC TRANSPORT STUDIES

The INEL test program was developed to predict innermost bag gas phase VOC concentrations from the headspace gas VOC concentrations using VOC transport models and data obtained from waste drum sampling. This program consisted of three stages.

In the first stage of the test program, a transient VOC transport model was developed to estimate the VOC concentration within laboratory-scale waste drums as a function of time (Liekhus et al. 1994a). The testing

demonstrated that transient gas phase VOC concentrations can be predicted based on theoretical transport mechanisms. Model equations accounted for three primary mechanisms for VOC transport from a void volume. These mechanisms were VOC permeation across a polymer confinement layer, VOC diffusion across an opening in a layer of confinement, and VOC solubility in a polymer confinement layer.

In the second stage of the INEL program, a steady-state transport model was developed to predict concentrations in laboratory-scale vented waste drums containing simulated waste sludge based on knowledge of drum headspace VOC concentration and waste drum configuration (Liekhus et al. 1994b). Experiments were performed to measure the VOC concentration in laboratory-scale vented waste drums containing two types of simulated waste sludge with differing compositions (Liekhus et al. 1994b). Experimental results were expressed as a ratio between drum headspace VOC concentration and the VOC concentration inside the innermost polyethylene bag. Model equations were used to determine the effect of model parameters on the estimated concentration difference across a transport boundary, such as the rigid drum liner or polymer bag, using permeation as the transport mechanism.

In the third stage of the INEL experimental program, the model developed in the second stage was used to estimate VOC concentrations in 64 actual waste drums from INEL and the Rocky Flats Environmental Technology Site (Rocky Flats). The effectiveness of the model in estimating VOC concentration was examined for vented waste drums containing different waste types and packaging configurations. The model results demonstrated that maximum VOC concentrations within actual waste drums can be estimated from drum headspace gas sampling data. Details of the gas sampling and analysis, equipment used, procedures employed and quality control procedures are provided in Liekhus (1995).

The VOC transport model was said to be unbiased if the 95-percent confidence limits on the mean logarithm of the ratio of the VOC concentration in the innermost layer of confinement to the maximum VOC concentration in the headspace is zero. Otherwise, the model is said to have a positive or negative bias. Transport model predictions were unbiased or positively biased (i.e., over estimated) for the maximum measured concentrations of 11VOCs within vented waste drums. These VOCs are carbon tetrachloride, chloroform, 1,1-dichloroethane, 1,1-dichloroethylene, methanol, methylene chloride, tetrachloroethylene, 1,1,1-trichloroethane, trichloroethylene, 1,1,2-trichloro-1,2,2-trifluoroethane, and toluene. The model exhibited a negative bias for 2 VOCs (p-xylene and acetone). The model was unbiased in estimating the total VOC concentration within the innermost layer of confinement for waste drums containing a maximum of four layers of polymer bags that were all sampled within one day and exhibiting detectable concentrations in inner bags. The model exhibited a positive bias in estimating the total VOC concentration within the innermost layer of confinement of waste drums containing a maximum of two or five layers of polymer bags.

Model precision was characterized for 13 VOCs and three drum configurations by the lower 90/90 tolerance limit (LTL) for the logarithm of the ratio described earlier. Based on the tolerance limits, the VOC transport model predicts an innermost bag VOC concentration that, with 90-percent confidence, will not be less than 50 percent of the maximum measured VOC concentration in at least 90 percent of the waste drums for 8 VOCs and two of the waste drum configurations. These VOCs are acetone,

1,1-dichloroethane, 1,1-dichloroethylene, methylene chloride, toluene, 1,1,1-trichloroethane, trichloroethylene, and 1,1,2-trichloro-1,2,2-trifluoroethane. The VOC transport model predicts that the VOC concentration will not be less than 33 percent of the maximum measured VOC concentration in at least 90 percent of the waste drums for 3 VOCs and waste drums with a maximum of four layers of polymer bags. The three VOCs are chloroform, tetrachloroethylene, and p-xylene. For carbon tetrachloride and methanol, model estimates will not be less than 20 percent of the maximum measured VOC concentration in at least 90 percent of the waste drums.

DRUM AGE CRITERION FOR HEADSPACE GAS SAMPLING

There is a certain age criterion that must be met by a drum of TRU waste in order for headspace gas samples to be either representative of gases in the drum or appropriate to use in predicting innermost bag VOC gas phase concentration. The drum age criterion (DAC) defines the age of a drum necessary to reach 90-percent of steady-state concentration within all of the bags and the rigid drum liner. The DAC establishes the time after waste packaging necessary to wait prior to drum headspace sampling to help ensure that the headspace sample analyses are suitable for their intended use.

DACs for two packaging configurations have been determined using transient VOC transport models and indicator VOCs selected on the basis of health risks and magnitude of concentration. The DACs are given in Table I. For drums of Waste Types I and IV, the DAC is 225 days. For drums of Waste Types II and III, the DAC is 142 days.

Table I

Indicator VOCs were selected by using two separate screening techniques and having the screened VOCs comprise the set of indicator VOCs. The screening techniques are consistent with the purposes for determining gas phase VOC concentrations in drums; one of the screening techniques focused on flammability issues related to operational period drum handling and the second focused on human health risk from inhalation for the operational period. To screen for operational period drum handling, the magnitude of VOC concentration in headspace gas samples from 465 INEL and Rocky Flats drums was examined.

The screening resulted in 11 distinct indicator VOCs. For drum handling, n-butanol, methyl ethyl ketone, methyl isobutyl ketone, and methanol were screened for inclusion in the indicator set. For operational period human health risk, 1,1-dichloroethylene, carbon tetrachloride, chlorobenzene, chloroform, methylene chloride, methyl ethyl ketone, 1,1,2,2-tetrachloroethane, trichloroethylene, and toluene were screened and included as indicator VOCs.A computer program was used to calculate the required drum age or vent time for each indicator VOC for three drum categories and two drum packaging configurations. The three drum categories are:

Drum Category 1, Existing Vented Drums: The existing vented drums are those that were stored as unvented drums and subsequently vented, the rigid drum liners punctured, and stored again for a period of time. Drums in this category must have been unvented for a sufficient time such that equilibrium concentrations existed within all confinement layers at the time of venting.

Drum Category 2, Newly Packaged Vented Drums: The newly packaged vented drums consist of drums and associated rigid drum liners that were vented

at the time of packaging. In this category, steady-state concentrations do not exist within all confinement layers at the time of packaging.

Drum Category 3, Existing Unvented Drums: The existing unvented drums are those drums that have been stored as unvented drums for a period of time. The drums are to be vented (i.e., the drum liners punctured and carbon composite filters installed in the drum lids) and headspace gas samples taken from inside the rigid drum liners at the time of venting. Two packaging drum configurations were considered in determining the DACs from the calculated drum ages and vent times. For drums containing sludges (Waste Types I and IV), there are two large bags within the rigid drum liner providing two inner layers of confinement. For solid waste (Waste Types II and III), it is conservatively assumed that five layers are within the rigid drum liner. The drum filter type is assumed to be NFT-020 for both packaging configurations; this assumption is considered conservative since NFT-020 filters are the most restrictive regarding the release of compounds of filter types being used in the DOE complex (Liekhus 1995). The longer of the following, for each configuration and over all indicator VOCs, are taken as the DACs:

Calculated vent time for newly packaged vented drums and

The sum of calculated vent time for existing vented drums and calculated drum age for existing unvented drums

The DACs that result are given in Table I. Potential future packaging configurations (e.g., those using filtered bags) were not considered and will require additional analyses to determine the appropriate DAC.

DISCUSSION OF TRANSPORT MODELING

The INEL test program and its associated transport modeling demonstrates that VOC transport can be modeled based on mathematical simulation of diffusion and permeation processes. This demonstration took place in three stages: transient VOC transport from vented laboratory-scale drums, steady-state VOC transport in laboratory-scale vented waste drums containing simulated waste sludge and the application of the steady-state model to actual waste drums. Based on comparing model results to actual waste drum concentrations, the steady-state VOC transport model demonstrated that innermost bag gas phase VOC concentrations can be predicted from headspace gas concentration data. Because these predictions can be made, sampling and analysis of inner layers of confinement will not be necessary.

The predictions can be made using prediction factors that conservatively approximate the steady-state model predictions. The prediction factors are derived by solving the steady-state model in terms of the ratio between the innermost layer gas phase VOC concentration and the headspace gas VOC concentration. The prediction factors have been computed for 29 VOCs for the two packaging configurations used for the DACs. Use of the prediction factors will be conservative for cases where the actual package has fewer layers than that assumed for the prediction factor. The prediction factors are also based on conservative assumptions of filter diffusivities. The prediction factors are given in Table II and range from 1.1 to 9.5 for Waste Types I and IV and from 1.7 to 39 for Waste Types II and III. To predict the innermost bag gas phase VOC concentration, the headspace gas VOC concentration and associated prediction factor should be multiplied.

Table II

The steady-state VOC transport model and the prediction factors are valid when the DAC has been met. The DAC establishes the time after waste

packaging necessary to wait prior to drum headspace gas sampling to be able to accurately predict the innermost layer gas phase VOC concentration within a drum. The innermost layer gas phase VOC concentration prediction will be the maximum predicted value. The DAC also establishes the waiting time that will ensure that the transport rates between layers of confinement are equal and headspace concentrations can be used in calculations for emissions through the drum filter.

If the drum is unvented (Drum Category 3) and the DAC has been met, modeling shows that the headspace gas within the rigid liner is representative of the drum gases within all layers of polymer confinement, because equilibrium has been reached. If the drum is vented (Drum Categories 1 and 2) and the DAC has been met, the innermost gas concentration can be predicted from the headspace gas concentration using the prediction factors in Table II. The predicted innermost bag concentration is concluded to conservatively represent the maximum concentration within a drum. In addition, if the DAC has been met, the headspace concentration for Drum Categories 1 and 2 can be said to represent headspace concentrations that control the rate of emissions through the drum filter over time.

PROPOSED APPROACH

The proposed approach provides prescriptions for determining drum gas phase VOC concentrations for operational period drum handling and operational period no-migration calculations for each of the three drum categories. The prescriptions are based on the DAC being met prior to headspace sampling and involve methodologies outlined below. Either headspace gas measurements or predictions of innermost bag concentrations, as appropriate, will be used.

Operational Period Drum Handling

Gas phase VOC concentrations in drums will be determined for the purpose of assessing flammability. The maximum gas phase VOC concentration is the concentration of interest for this assessment. The approach is to use predicted concentrations for the maximum in cases where the headspace concentration is not representative of the maximum. The basic steps to the approach are as follows:

1. Determine the drum packaging configuration.
2. Establish and meet the DAC.
3. Sample and analyze headspace gas.
4. Determine Drum Category. Determine drum concentrations as follows:

a. For existing vented drums (Drum Category 1), determine the predicted value using the factors in Table II. The predicted value will be used for the drum concentration.

b. For newly packaged vented drums (Drum Category 2), the predicted value will be used as for existing vented drums (see a.).

c. For existing unvented drums (Drum Category 3), headspace concentrations within the rigid drum liner will be used, because the DAC ensures representativeness.

The selected flammable VOCs concentration values will be summed for each drum.

Operational Period No-Migration Calculations

Gas phase VOC concentrations in drums will be determined for the purpose of calculating VOC emissions through drum filters. Because steady-state conditions are required, the VOC emission rate from a vented drum is a function of the drum headspace VOC concentration and the VOC diffusion

characteristic across the drum filter. VOC concentrations in inner layers of confinement are not used, and thus are not selected for the drum concentration; rather, the approach is to use headspace concentrations. The basic steps to the approach are as follows:

1. Determine the drum packaging configuration.
2. Establish and meet the DAC.
3. Sample and analyze headspace gas.
4. Determine Drum Category. Determine drum concentrations as follows:

a. For existing vented drums (Drum Category 1), the headspace concentration will be used.

b. For newly packaged vented drums (Drum Category 2), the headspace concentration will be used as for existing vented drums.

c. For existing unvented drums (Drum Category 3), headspace concentrations within the rigid drum liner will be used.

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THE APPROACH TO OPTIMIZATION OF HIGH-LEVEL WASTE BOROSILICATE GLASS WASTE FORM

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ABSTRACT

This paper addresses: 1) the selection of borosilicate glass as the reference waste matrix for solidifying high-level radioactive wastes at the Savannah River and West Valley Sites using the vitrification process; 2) controlling the composition of the feed at the melter to ensure

achieving a quality glass product and meeting processing constraints, and 3) the historical development of glass-former compositions at DWPF.

Most countries throughout the world, including the United States, that have produced HLW from the reprocessing of nuclear fuel have selected or are evaluating borosilicate glass as a final waste form. The principal reasons for the international focus on the production of waste glass, particularly on borosilicate waste glasses, are the ease of processing and the relative insensitivity of glass properties to fluctuations in waste composition. The chemical durability of borosilicate glass has also been an important consideration in its choice as a HLW form.

Borosilicate waste glass incorporates the constituents of the high-level waste, which includes Cs-137, Pu-238, and Sr-90, directly into its molecular structure. It can accommodate many different elements and wide variations in waste composition. However, the composition of the melter feed has to be controlled to achieve a satisfactory product and to meet processing constraints. The selection of the glass forming (frit) composition for a given waste composition is made based on desired waste loading, the composition of the high-level radioactive waste, and processing considerations. The goal of tailoring this glass composition is to obtain the optimum balance of all the important glass properties including viscosity of the glass melt, glass liquidus temperature, and glass durability.

Following a "system approach", the Savannah River Site (SRS) developed glass-former compositions that balances glass durability against process reliability. The primary processing properties are the viscosity and waste solubility. In the approach taken by SRS , an optimum frit is defined as one which maximizes waste solubility (by minimizing liquidus temperature) and which yields waste glasses which leach as little material as possible in the standard leach test and which are in the proper range of viscosities.

BOROSILICATE GLASS SELECTION AS FINAL WASTE FORM

Glass has been recognized by a number of countries as a promising medium for the immobilization of high-level radioactive waste (HLW) because of the commonly perceived durability of silicate glasses and the capacity of the glass to incorporate many different elements -- a necessity for radioactive waste immobilization. The projected high glass durability over long periods of time is indicated by the survival of naturally occurring glasses for millions of years and synthetic glasses for thousands of years. The technology of glass fabrication also has a long history. Glass has been made and used since ancient times, and borosilicate glasses have been made and used since the early 1900s (1).

In 1978, an independent panel organized by the American Physical Society (2) concluded that an emplacement of vitrified waste into a geologic repository was likely to provide an adequate solution to the problem of HLW management. Subsequently, DOE's National High Level Technology Program conducted a review of all options for immobilization of HLW along with assessments of the technology required at the various sites (3). Research was conducted on 17 candidate waste forms by national laboratories, universities, industrial laboratories, and DOE facilities. An independent review of this research, conducted by the Alternative Waste Form Peer Review Panel (4), reduced the list of waste form candidates form 15 to 8. Finally, the Panel ranked these 8 candidate waste forms on the basis of a weighted evaluation using leach resistance, waste loading, mechanical strength, radiation stability, and thermal

stability as criteria, with leach resistance from static leach tests receiving the highest weighting factor. In the final ratings, borosilicate glass was ranked first.

INTERNATIONAL FOCUS ON BOROSILICATE WASTE GLASSES

The first production facility to convert HLW into borosilicate glass using the vitrification process was the AVM facility in Marcoule, France in 1978. This facility continues to run today. Currently, the world's most mature vitrification technologies are the French process, which uses a rotary calciner and metallic melter, and several adaptations of the liquid- or slurry-fed ceramic lined melter (LFCM). The French process has been adopted in the Windscale Vitrification Plant facility at Sellafield in Great Britain. Three countries have committed to development of the LFCM technology: Germany, the U.S. and Japan. In 1979, the PAMELA Plant began initial vitrification operation at Mol, Belgium, and went radioactive in 1986. In the U.S., the Savannah River Site Defense Waste Processing Facility (DWPF) and the West Valley Demonstration Project (WVDP) will begin radioactive operation in 1996. Japan's Tokai Vitrification Facility (TVF) will begin operations in 1998. A pilot "hot" facility is operating now. All these processes produce borosilicate glass as a final waste form.

Best Demonstrated Available Technology

In 1990, vitrification of HLW was designated by the U.S. Environmental Protection Agency (EPA) as the Best Demonstrated Available Technology (BDAT) (5) for high-level waste. The Agency provided notice in the proposed rule (54 FR 48492) that DOE was providing, to the Agency, vitrification treatment data and that these data were available during the comment period for notice and public comment. The Agency concluded, after analyzing the data, reviewing comments, and performing a site visit, that vitrification will provide effective immobilization of the inorganic constituents in high-level waste.

GLASS PROCESS AND PRODUCT PROPERTIES

Vitrification of Waste

Vitrification of high-level radioactive waste at the SRS* will be accomplished in slurry-fed joule-heated, ceramic-lined melters. These melters will be fed and poured continuously. A full DWPF melter will hold enough glass to fill 4-5 canisters or about 7500 kg of glass. The average residence time of the molten glass in the DWPF melter will be 65 hours. The glass batch will be melted by Joule heating using two pairs of flat, parallel, Inconel electrodes immersed in the melt. The glass will be poured by a vacuum process. An advantage of a ceramic-lined melter is that ceramic refractories are more temperature and corrosion resistant than typical metal liners thus yielding higher melter life cycle. Development of the slurry feeding concept was an important development for the vitrification of nuclear waste. Slurry feeding significantly reduces capital costs and greatly simplifies the operation by eliminating the need to either dry or calcine the slurry prior to vitrification. Slurry feeding eliminates the capital cost of calcining and yields a more homogeneous feed.

Waste Acceptance Product Specifications

From a production perspective, the main determinant of a satisfactory borosilicate waste glass form is satisfying the requirements of the DOE Office of Environmental Management's (EM) Waste Acceptance Product Specifications for Vitrified High-Level Waste Forms (WAPS) (6). These specifications define the criteria that the waste glass products must

satisfy before they will be accepted by the U.S. DOE Office of Civilian Radioactive Waste Management (OCRWM) for disposal. The WAPS are divided into five sections, dealing with the waste form (borosilicate glass), the canister, the canistered waste form, quality assurance, and documentation. The underlying rationale for most of the specifications of the EM WAPS can be traced to the DOE Office of Civilian Radioactive Waste Management's Waste Acceptance System Requirements Document (WA-SRD) (7). The waste form is specified to be borosilicate waste glass. The chemical composition, crystalline phase projections, radionuclide inventory, a Product Consistency Test result or projection, and other characteristics must be reported for the glass. The canister must be fabricated from austenitic stainless steel to specific dimensions and be identifiable. The canistered waste form must be sealed closed, be free of foreign materials, meet heat generation and dose limits, be capable of surviving a specific drop test, and meet other requirements.

Melter Feed Composition Effects

Just as the expression "high-level waste" does not refer to one waste composition, the expression "borosilicate glass" does not refer to one glass composition, but instead to the family of compositions whose primary glass-forming ingredients are boron and silicon. Borosilicate waste glass is a vitreous material which incorporates the constituents of the high-level waste directly into its molecular structure. It can accommodate many different elements and wide variations in waste composition. However, the composition of the melter feed has to be controlled to achieve an acceptable product and to meet processing constraints.

The chemical composition of the melter feed is the primary determinant of the chemical durability of the waste glass product and many of the important melt processing properties (e.g., viscosity). Thus, chemical composition of the melter feed is the most important variable to be controlled. Since an incorrectly formulated glass cannot easily be remediated once it has been poured into a canister, the feed composition must be correct. The selection of the frit composition for a given waste composition is based on the desired waste loading, the compositions of the high-level radioactive waste, and processing considerations.

The compositions of nuclear waste glasses vary based on the particular balance of physical, chemical, and thermal properties which is desired, the types of equipment being used for processing, and the types of waste being processed. The viscosity of the glass melt must fall within a carefully defined range to assure that the glass will melt properly, will be properly homogenized via viscous flow within the melter, and will pour from the melter. If the viscosity is too low, excessive attack on the refractories in the melter will result. The temperature at which crystals begin to form in the glass if the temperature is maintained below a minimum temperature, termed the glass liquidus temperature, is also determined by the glass composition. Components that tend to make the glass more durable also tend to raise its viscosity and its liquidus temperature. Higher viscosities make processing more difficult while higher melting temperatures required to overcome high liquidus temperatures and lower viscosities can result in increased volatilization. The goal of tailoring the glass composition is to obtain the optimum balance of all the important glass properties.

High-level radioactive waste is added at selected waste loadings to borosilicate glass frit or to components which when melted together will

form the desired borosilicate glass matrix. The primary ingredients of the borosilicate glass frit are silicon dioxide (SiO_2), boron oxide (B_2O_3), and alkali metal oxides (Na_2O, K_2O, and Li_2O). Other ingredients are added to obtain a balance of glass properties. Since the primary chemical constituents of high-level radioactive waste are alumina (Al_2O_3), iron oxide (Fe_2O_3), and manganese dioxide (MnO_2), addition of high-level radioactive waste to borosilicate glass frit enhances the (short-term) durability of the product. When waste loadings of 40% or more are reached, the glass structure is so perturbed that rearrangement of atoms into more stable crystalline structures with accompanying changes to the chemical composition of the remaining glassy phase is favored. For this reason, lower limits on waste loading levels are employed. Effects of waste constituents on important glass processing and product properties are shown in Table I.

EVOLUTION OF WASTE GLASS FORMULATION AT DWPF

The primary objective of the DWPF is to convert the liquid HLW slurries to a stable borosilicate waste glass. By itself, the waste would not form a glass when heated to 1150C. Thus, frit* will be added to the waste in the DWPF in order to achieve a mixture which will form a glass. On an oxide weight basis, DWPF glass will consist of approximately 64% glass frit and 36% waste feed of which 28% is sludge and 8% is a precipitate stream containing much of the Cs-137 radioactive species. This stream is also called precipitate hydrolysis aqueous (PHA). The glass frit and the PHA together make up what is called the glass-former composition.

The glass-former composition* for the DWPF has been developed so that the DWPF waste glasses will have similar properties to simulated waste glasses based on "Frit 165", which is a frit developed for sludge - only processing (early DWPF flowsheets were based on immobilizing only the sludge fraction in the waste tanks). This sludge-only frit was designed to immobilize the full range of sludges contained in the SRS Tank Farm, and be tolerant of variations in waste composition.

In general, the SRS has followed a "system approach" in developing waste glass compositions. This approach balances glass durability against process reliability. Thus, an optimal composition is defined as one which combines desirable product properties with process properties which lead to uninterrupted processing. The primary product property considered is the durability (stability of the glass toward reaction with water). The primary processing properties are the viscosity, liquidus temperature, and waste solubility. Both the viscosity and waste solubility have product implications: glass viscosity affects the ability to fill the canister and waste solubility affects the thermal stability of the glass. In the approach taken at the SRS (8), an optimum frit is defined as one which maximizes waste solubility (by minimizing liquidus temperature), yields minimal leaching based on a standard leach test, and which yields a glass viscosity within the desired range for the DWPF melter (20 to 100 poise at 1150C).

For the frit selected for sludge-only processing, a self-directed optimization procedure was used to reach a "best" composition. An initial series of twelve frits was selected. Since the glass frit must accommodate the entire range of waste compositions, each frit was tested by preparing melts containing the test frit and the extreme waste composition which is most likely to cause the frit to violate one of the criteria above (PUREX waste for durability and waste solubility, Heavy Metal (HM) waste for viscosity). The melting temperature, 1150C, and

waste concentrations were held fixed at values used for development of the Environmental Assessment (EA) glass.

Data was collected on waste glasses made from the first 12 frits and the frits were then ranked by using a desirability coefficient which objectively weighted each of the criteria. The four worst compositions were replaced with four new compositions. The four new compositions were treated in the same way as the previous twelve, and a new ranking was then developed . While the methodology works by examining the worst frits, the end result is reached by examining the best. This process was concluded once it became clear that the region of best composition was not changing significantly. This occurred after the third round. By this time twenty frits had been studied.

The data representing these frits served as the basis for the final step, the proposal of an "optimum" frit. The compositions and properties were examined to determine the effects of each component. In general, higher amounts of silica and zirconia improved durability, but also increased viscosity. The higher viscosity is best compensated by increasing lithia relative to soda since lithia is a more effective flux and has a less deleterious effect on the leach rate. Lanthanum oxide had no clear-cut benefits to recommend its inclusion, and significantly increased the liquidus temperature. Boric oxide, magnesia and titania had relatively smaller effects on the measured properties.

These trends were used to produce five candidates for the "optimum". These frits were tested in the same manner as was done during the statistically-designed portion of the experiments. Three of the five were better than the best of the frits from the statistically-designed tests. These three were so similar in desirability coefficient and in their individual properties that an additional test to choose between them was needed. The test used was a simple side-by-side comparison of their resistance to devitrification.

Each of the final candidate frits were used to prepare waste glasses using a simulated waste composition similar to that in the DWPF Environmental Assessment (9). The EA glass composition was also tested. After melting at 1150C, all of the glasses were cooled in a manner which simulated the behavior in the center of a DWPF canister after filling with glass. Glass in this location sees the slowest cooling rate and spends the longest time between the liquidus temperature and the glass transition temperature. It represents the worst case for crystal formation. The results of this experiment showed that the EA glass contained the most crystalline material. For Frit 165, the best of these compositions, only a few small well-separated crystals could be found. As a result of this test, Frit 165 was selected as the frit composition for sludge-only processing.

This composition (Frit 165) has successfully been used both in tests with actual waste simulating natural environments, and in in-situ experiments in granite, salt, and clay. In all cases, this glass performed better than any of the other high-level waste glasses tested (including some that were melted at higher temperatures). For this reason, when the precipitation process for decontamination of high-level waste salt was adopted at SRS, the glass-former composition was modified to achieve a family of glasses as similar to 165-based glasses as possible. This was called Frit 200. This was achieved by removing some of the boron and alkali from the glass frit composition to take into account additions of these chemicals from the precipitation process. Similarly, when it became

apparent that additional sodium hydroxide and nitrite would be added to the waste as inhibitors to reduce waste tank corrosion, further reductions in the amount of alkali in the frit were made (Frit 202). The evolution in glass frit composition from that used for the EA glass, and that which will be used initially in the DWPF (Frit 202), is shown in Table II.

Table I

Table II

CONCLUSION

Borosilicate glass is not just one glass composition but is instead a family of compositions whose primary glass-forming ingredients are boron and silicon. The design and selection of the optimum glass composition is determined by the desired waste loading, then composition of the high-level radioactive waste, and processing considerations.

Whenever major changes are made to the vitrification process, the composition of the waste stream undergoes a significant change, or some other valid reason; the borosilicate glass composition can be tailored to maintain the optimum balance of all important glass properties including viscosity of the glass melt, glass liquidus temperature, and glass durability.

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CONSTRUCTION OF A VITRIFICATION FACILITY

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ABSTRACT

The Vitrification Facility at the West Valley Demonstration Project (WVDP) is comprised of the buildings, equipment, components, and

utilities that will be used to solidify (vitrify) high-level radioactive waste into stable glass logs contained within stainless steel canisters. Due to the unique nature of the WVDP and the vitrification process, construction of the necessary facilities presented distinct challenges in the areas of new facility construction and existing facility renovation. This paper will focus on construction of the vitrification facilities, which includes a shielded trench and piping to transfer High-level Waste (HLW) to a shielded cell where it will be vitrified and a system to remove nitrous oxide from the waste gas (Ex-cell Off-gas System). These facilities were constructed using unique techniques in order to allow pretreatment of the low-level waste and the vitrification development programs to continue uninterrupted.

Construction of the vitrification facilities was completed in four major phases over six years. The first phase was civil/structural construction, completed in 1990. This construction included the Crane Maintenance Room (CMR) for maintaining the process cranes, a tunnel to transfer glass canisters from the cell to the interim storage area (Transfer Tunnel), a Secondary Filter Room (SFR) housing the secondary filtration system and main exhaust fans, a Diesel Generator Room (DGR), the Cold Chemical (CC) Building to mix radiologically cold chemicals, and portions of the Shielded Vitrification Cell. The second phase was also civil/structural construction and that was completed in 1991. This effort consisted of the HLW Transfer Trench and completion of the 1.2 meter-thick concrete shielding walls for the cell called Wall Modules. In addition, the existing steel building was modified to include more square footage and additional floors. The third phase, completed in 1994, included mechanical construction, instrumentation and controls (I & C), and electrical installation. This work consisted of installation of all equipment, instrumentation, and electrical wiring and cabling for the facility, as well as installation of the final two wall modules. The fourth phase included construction of the Ex-cell Off-gas Facility that was completed June 1995. This construction phase involved converting a portion of an existing building to treat process melter off-gases.

INTRODUCTION

The WVDP is the site of a former commercial nuclear fuel reprocessing facility that operated from 1966 through 1972. In 1980, the West Valley Demonstration Project Act was signed by the president of the United States to become Public Law 96-368. The Act directs the Department of Energy (DOE) to, among other things, solidify the high-level waste stored at the site into a durable, solid form suitable for shipment to a federal repository; to clean and close the facilities used; and to dispose of the low-level and transuranic wastes collected during Project operations. The DOE began managing the WVDP in 1982 with West Valley Nuclear Services (WVNS) Co. Inc. selected as the management and operating contractor. Approximately 2,270 cubic meters of high-level nuclear waste remained in the underground steel tank within a concrete vault at the start of the Project. The waste, the result of nuclear fuel reprocessing operations, was composed of two layers; a bottom sludge and upper liquid layer. The liquid has been processed into cement-filled drums and is stored on site, leaving the sludge to be processed and poured into stainless steel canisters.

The Vitrification Facility was constructed between the Tank Farm, where the waste is stored, and the existing nuclear fuel reprocessing building. Thus incorporating the area being used for development testing called the

Component Test Stand (CTS) in the process. Figure 1 is an artist's rendering of the Vitrification Facility. It was in this area that most of the major construction took place.

Fig. 1

PHASE 1 - CIVIL/STRUCTURAL INSTALLATION CONTRACT

The civil/structural construction contract focused on two areas. The first area was located between the CTS and the existing reprocessing building. This area contained portions of the Vitrification Cell, Transfer Tunnel, Secondary Filter Room, HVAC Control Room, and the Crane Maintenance Room. All of these areas are shielded and built to withstand a seismic event; thus the bulk of the construction work was in the form of heavily reinforced concrete floors, walls, and ceilings. Problems to contend with included: 1) working around contaminated buildings and soil near the existing process building 2) avoiding guy wires from the main exhaust stack 3) working around scheduled transfers of radioactive liquid to the Cement Solidification System (CSS) through the trench located in the area 4) inclement weather and 5) providing continuous access to the test facility.

The conversion of the CTS to what is now known as the Vitrification Cell was accomplished in phases in order to: minimize construction and testing interferences, permit some concurrent development/operations and construction work, and fit the delivery dates of government furnished equipment. Near the end of Phase 1, testing was completed and the CTS was dismantled. The test melter, associated piping, electrical systems, and equipment were removed. This effort required close coordination among two subcontractors and WVNS personnel. Some of the equipment was salvaged to be installed in the new facility.

One of the unique construction features of the Vitrification Cell was the wall modules. These special walls were fabricated off site to facilitate continued testing and were installed between the concrete roof and columns that had been installed before testing started. Soon after the CTS was dismantled, five of the seven wall modules were installed. These large stainless steel modules (averaging 4.7 meters wide by 7.1 meters high by 1.2 meters thick) were prefabricated in a vendor's shop complete with all shielded piping and electrical penetrations, window liners, and manipulator and shield plug ports. The installation included adding reinforcing bar (or rebar), which was tied to the existing columns using threaded rebar screwed into embedded couplings in the columns. The installation of these prefabricated modules was successfully completed as planned and saved a considerable amount of time.

Eight radiation shielding doors were installed as the construction proceeded. The largest of these shield doors, a 105-ton CMR door, required using a 350-ton mobile crane. This was problematic due to the small working area: the crane had to be positioned between the west side of the Vitrification Building, the stack's guy wires, and the Cold Chemical Building excavation.

The second major focus of Phase I construction included areas west of the CTS, where the Cold Chemical Building was built. This building houses tanks, pumps, and grinders used to mix the chemicals and glass formers before they are pumped to the Vitrification Cell to be mixed with the radioactive sludge. It is a structural steel building with a metal skin. The bottom floor is concrete with curbs and sumps to contain potential chemical spills. The second floor is steel with grating around the tops of vessels. Completion of the structural steel portion of this building

was delayed until the dismantling of the temporary cold chemical system, because the existing tanks were to be used for the final system. The tanks were installed as the building was erected.

PHASE 2 - CIVIL/STRUCTURAL MODIFICATION CONTRACT

This civil/structural construction contract modified the CTS Building to bring it to its final configuration and allow for installation of the process equipment. Specifically, it modified the structural steel building, added more floor space, and installed underground fire sprinkler mains and floor drains. This portion of the work was relatively straightforward in that the WVNS Construction Department controlled the area and no operational facilities were adjacent to the CTS Building.

The wall modules, installed by the first contract, had a stainless steel face on the radioactive or "hot" side of the cell wall to be used as the interior form work for placing concrete. The unique portion of this work was placing concrete in these wall modules. See the photograph in Fig. 2.

Fig. 2

WVNS and the DOE consulted with the US Nuclear Regulatory Commission (NRC) on the issues involved in forming the wall modules. All parties were interested in developing a method of concrete placement that would ensure no voids within the concrete interior of the wall modules. It was decided to construct a prototype wall to demonstrate the placement of the specially developed concrete mix. The concrete was placed in the prototype wall, with the procedures and techniques used for placement being carefully watched and critiqued. The wall was then evaluated by destructive examination and this indicated that the concrete did flow around the numerous wall penetrations. The actual placement of concrete in the modules progressed well as a result of the lessons learned from the prototype.

The other major portion of this contract installed the HLW trench that extends from the Waste Tank Farm to the Vitrification Cell. This trench consisted of a 0.31 meter-thick concrete floor, 0.5 meter-thick concrete walls, and 0.61 meter-thick removable concrete covers. The floor and walls were cast-in-place using conventional form work. In addition, there were four concrete valve and jumper pits constructed adjacent to the trench.

The concrete covers were fabricated off site at a precast concrete contractor's shop. There were several different sizes and shapes that were cast in custom steel forms. The quality of these covers was acceptable, with tight dimensional accuracy. The covers fit the cast-in-place concrete trench with only minor grinding.

PHASE 3 - MECHANICAL, I & C, ELECTRICAL INSTALLATION CONTRACT

This mechanical, I & C, and electrical construction contract installed all of the In-cell process equipment including vessels, jumpers, pipes, the Off-gas system, the ventilation system, and the electrical systems. The contract also installed all the controls, electrical, piping for utilities, and equipment located outside the cell to support and operate the In-cell processes. See Fig. 3.

Fig. 3

The layout of In-cell process equipment provided some opportunities for innovative installation methods. One method required the installation of 60 stainless steel pipes and conduits from the hot face of the cell wall to the stub wall. The hot face of the cell wall and the stub wall are perpendicular to each other and the space allocated for this installation was 1.0 meter from the hot face by 3.3 meters from the stub wall by 2.0

meters high. Approximately 120 field welds were made in this area. The sequence of installation was critical: one pipe installed in the wrong sequence could make it difficult to install additional pipes at a later date.

Many areas of the Facility had been modeled three-dimensionally on a computer to check for interferences. It was decided to use this modeling to create the isometric drawings and provide them to the subcontractor. Using this approach saved several weeks in the planning and approval cycles, and allowed the cell equipment to be installed earlier than scheduled.

The HLW transfer pipe was also installed in the HLW trench as a part of this contract. The trench is 152 meters long and from 0.6 to 1.8 meters wide. There are 762 meters of Schedule 40 stainless steel transfer pipe encased by a Schedule 40 stainless steel guard pipe. The pipe was placed in the trench on multiple layers of pipe supports with tight tolerances and critical slopes for drainage. This work required careful planning to ensure that 100 percent of the transfer pipe joints could be radiographed. The installation and field radiography took place in a physically tight area that required confined space entry qualified personnel.

The final seismic evaluation caused some redesign of the transfer piping in the HLW transfer trench. This was done before the subcontractor began shop fabrication of the transfer pipe, thus minimizing the impact to the overall schedule.

After all of the piping was installed and inspected, the trench covers were placed on the trench, the joints caulked, and the covers and upper side of the trench insulated.

PHASE 4 - EX-CELL OFF-GAS FACILITY CONTRACT

The Ex-cell Off-gas Facility construction contract modified and added to the existing 01-14 Building to create a NOx Abatement Facility. The existing building was made of reinforced concrete and included hot cells, ventilation room equipment, controls, and on-going processing of liquid waste. The contract additionally included modification of the Off-gas trench to accommodate the Off-gas pipe. This trench was also used for shielding the liquid transfer line. See Fig. 4.

Fig. 4

Building construction was one of the most challenging parts of the Project. The challenge arose because one half of the building was being used by Operations to process liquid, waste and the Off-gas trench was being used for transferring liquid waste. The utilities and controls were located on both sides of the building and careful scheduling and coordination were necessary to allow production to continue. This construction included the challenge of sequencing the demolition of various systems during operations and relocating operating systems with only short duration shut downs. Also, the building expansion took place in the winter and construction areas had to be winterized with temporary heat and additional shelters erected. Maintaining access was difficult, and it became necessary to install more stairs, walkways, heating systems, and lighting.

Most of the construction in this phase was performed in a radiological buffer area. Other hazards associated with this phase of construction included asbestos abatement of insulation around vessels and roof penetrations, excavations in radiologically contaminated areas, installing fasteners in contaminated concrete, performing 25 radiological

hot taps, and performing the majority of work in confined spaces, as required by DOE Orders, consistent with the Occupational Safety and Health Administration (OSHA) requirements. Furthermore, all crane work had to be sequenced carefully to accommodate limited operating areas and areas which that populated by workers.

Safety

Personnel safety has always been of prime importance at the WVDP. During Construction of the Facility, the Construction Safety Department continually strove to improve the safety of all personnel. In 1994 and 1995, during the peak building period, construction safety at the WVDP had one of the lowest Total Recordable Case Rates (TRC) of any active DOE site; finishing 1994 with a TRC of 3.34 and 1995 with a TRC of 3.44. This is significantly less than the construction industry average of 12.2 and the DOE construction average of 7.5. The overall TRC for all construction on this Project, including subcontractors, is 4.9. This figure represents 11 years and 2.5 million person-hours.

Many factors have led to the outstanding safety record in construction at the WVDP. Listed below are some of the specific factors and actions taken to enhance the safety performance at the WVDP:

All parties on site, from DOE to the smallest subcontractor, are involved in site programs to achieve a total safety culture.

Construction safety has developed a subcontractor safety training checklist that identifies the prerequisite safety training a subcontractor must complete before working on site. Subcontractor safety records are reviewed monthly using this checklist.

Construction and subcontractor safety personnel perform daily walkdowns of construction areas to identify safety hazards. Safety inspections of each subcontractor are performed weekly to a pre-existing checklist to monitor each subcontractor's program.

WVNS Construction Department, WVNS Industrial Hygiene and Safety Department, and subcontractor safety professionals meet monthly to discuss relevant safety issues. These meeting have been very instrumental in identifying and correcting safety issues and concerns expeditiously.

All WVNS construction personnel have been trained in OSHA hazard recognition and instructed to look for hazards during the course of their daily duties. They have been instructed to correct hazards when possible or to report hazards to the appropriate safety professional or WVNS management.

WVNS encourages construction subcontractors to give proper attention to safety. To reinforce this, WVNS requires direct daily communication with the subcontractors' safety persons on safety issues; this includes written records of corrective actions on surveillance items and formal critiques for any event that is defined to be reportable.

CONCLUSION

Construction of all vitrification-related buildings and areas was a unique and sometimes difficult project due to the Project guidelines requiring the conversion of existing facilities, and because of the concurrent vitrification process development, testing, and operations that were occurring during construction. The Construction Department developed unique methods to complete projects and maintain schedules. The significant focus on safety as a culture yielded excellent results, with all parties having a safe work ethic and clearly committed to having a safe work environment.

The actual cost of the vitrification facilities construction closely paralleled the Total Estimated Cost (TEC) for the Project established in 1988. The TEC was $122M and the final cost was $127M; representing only a four percent escalation in cost.
Work continues to tie-in radioactive lines from the Waste Tank Farm to nonradioactive piping in the Vitrification Facility and adjacent structures. Radioactive operations are scheduled to commence in June 1996.

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DEPARTMENT OF ENERGY REMOTE-HANDLED TRANSURANIC WASTE FACILITY CHARACTERIZATION CAPABILITY ASSESSMENT

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ABSTRACT

The Department of Energy (DOE) currently stores remote-handled (RH) transuranic (TRU) wastes at seven sites across the complex. It plans to ship this waste to the Waste Isolation Pilot Plant (WIPP) in Carlsbad, New Mexico for disposal. This requires the waste to be packaged and characterized to meet the WIPP Waste Acceptance Criteria (WIPP-WAC) and applicable transportation requirements. This study was conducted to assess both existing and future capabilities for characterization of RH TRU waste at the seven DOE sites that store these wastes. The characterization methodologies assessed included process knowledge, radioassay, radiography, visual examination, and chemical analysis. The study concluded that the current infrastructure for RH TRU waste characterization will not support certification to the WIPP-WAC. Although the WIPP-WAC for RH TRU waste is not finalized and the associated requirements are not completely known, it is reasonable to assume that the final requirements will be at least as rigorous as those for contact-handled TRU waste. Therefore, considerable progress must be made to improve current capabilities for RH TRU waste characterization, particularly for nondestructive assay and nondestructive examination where there is limited capability.

INTRODUCTION

The Department of Energy (DOE) currently stores remote-handled (RH) transuranic (TRU) wastes at seven sites across the complex. These sites include the Argonne National Laboratory-East (ANL-E), Argonne National Laboratory-West (ANL-W), the Hanford Site (HANF), the Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), and Oak Ridge National Laboratory (ORNL). The current inventory of stored RH TRU waste is approximately 1,600m3, with over 60% of the volume residing at ORNL (1). An additional 3,480m3 of RH TRU waste is also projected to be generated at the above sites over the next several years, primarily from remedial activities. Recently, HANF alone has projected approximately 3,000m3 of RH TRU waste, about 85% of the total forecasted. TRU waste is defined by DOE Order 5820.2A as all radioactive waste that contains greater than 100 nanocuries of alpha-emitting isotopes with atomic numbers greater than 92 and half lives greater than 20 years per gram of waste (100 nCi/g). RH TRU waste is any TRU waste with a radiation dose rate measured at the surface of the waste container greater than 200

mrem/hr. DOE currently plans to ship all RH TRU waste to the Waste Isolation Pilot Plant (WIPP) in Carlsbad, New Mexico for disposal. DOE is required to repackage, characterize, and perhaps treat much of the waste in order to meet WIPP Waste Acceptance Criteria (WIPP-WAC) and applicable transportation requirements.

RH TRU waste exists in solid, liquid, and sludge form. Most of the waste will require repackaging and further characterization before it can be shipped to WIPP for disposal. The RH TRU wastes that require repackaging are either stored in containers not suitable for either transportation or for handling at WIPP (i.e., concrete casks and large storage tanks). They also may require visual inspection and sampling for verification of process knowledge. Some wastes such as sludges are stored in large tanks that may require treatment to meet the WIPP-WAC or other applicable requirements to achieve more stable waste forms.

Due to the high radioactivity and its impacts on operational safety, characterization of RH TRU waste is particularly challenging. Characterization methods used to sample and analyze the waste must be developed to ensure the quality standards are acceptable to the Environmental Protection Agency (EPA). Following evaluation and acceptance of these methods, WIPP documentation including the Quality Assurance Program Plan (QAPP), Sampling and Analysis Guidance Manual, and the Performance Demonstration Program must be revised to incorporate the requirements at each DOE generator/storage facility.

Characterization of RH TRU waste likely will require a substantial portion of the resource commitment by DOE to prepare these wastes for disposal at WIPP. This study was conducted to assess both existing and future capabilities for characterization of RH TRU waste at the seven DOE sites that store these wastes. By comparing this study with the RH TRU waste characterization requirements currently under development, DOE will be able to assess needs and generate plans for system-wide development to support adequate characterization of all stored and projected RH TRU wastes.

CHARACTERIZATION REQUIREMENTS

Adequate waste characterization data for RH TRU wastes must be collected from the DOE sites so WIPP can demonstrate compliance with applicable regulations for the total disposed inventory. Any site that does not provide sufficient waste characterization information will not be eligible to send its waste to WIPP for final disposal. For consistent RH TRU categorization, TRU waste should be correlated to the Waste Matrix Codes established by DOE as acceptable to the WIPP facility. This task, along with other tasks, should be accomplished through implementation of the DOE TRU Waste QAPP. The characterization program should address the data needs associated with regulatory compliance programs listed below.

Performance Assessment - Evaluation of long-term radionuclide containment (40 CFR191).

Land Disposal Restrictions (40 CFR 268.6) - Containment of hazardous constituents.

General Waste Analysis - Verification of waste characterization data [40 CFR 270.14(b)(2) and 270.23(c)].

Transportation of Radioactive Waste - Approval of the Safety Analysis Report for Packaging of the RH TRU shipping cask (10 CFR 71).

Compliance Criteria (40 CFR 194).

CHARACTERIZATION METHODS

RH TRU waste characterization refers to the examination and sampling and analysis of the waste and to the inspection of waste generation records as a means to demonstrate compliance with the requirements of applicable regulations.

Process Knowledge: Process knowledge is the ability of waste generators to use their understanding of an operation to characterize the various components that make up a given waste stream. Process knowledge is the least costly method of waste characterization and does not subject facility personnel to radiological or other operational risks. Because of the particularly high contact radiation levels and inherent handling risks associated with RH TRU waste, process knowledge will play an important role in the overall characterization process.

Radioassay: The Nuclear Regulatory Commission has identified radioassay as an acceptable technique for the identification of radionuclide contents of a RH TRU waste container. Numerous radioassay techniques are available to determine the TRU content of bulk waste. Radioassay methods may include both nondestructive and destructive techniques. Nondestructive radioassay capability needs to be developed for RH TRU waste.

Radiography: Radiography is a nondestructive qualitative and "semi-quantitative" (i.e., used to estimate the volume of free liquids that might be present in within the waste matrix) technique that involves X-ray scanning of waste containers to identify and verify waste container contents. Real-time radiography (RTR) has been developed by DOE specifically to aid in the examination and identification of containerized waste. There is no equivalent or associated method found in EPA sampling and analysis guidance documents.

Visual Characterization: The visual examination of the contents of a waste container is designed to provide data on the type and amount of material in each of the waste containers. Visual examination is an analytical process that does not involve sampling procedures. The technique provides a compliance assessment of the waste contents with respect to the WIPP-WAC. It also verifies process knowledge and physical waste form and determines whether detectable free liquids are present in the waste contents. Also, the results of radiography can be verified through visual examination of a statistically selected number of waste containers from each waste stream.

Chemical Analysis: About 45% of the RH TRU waste volume is "mixed" waste containing both radioactive and hazardous constituents that are regulated under the Resource Conservation and Recovery Act (RCRA). Examples of mixed waste are radionuclide-contaminated spent solvents, discarded materials contaminated with both solvents and radioactive material, scintillation fluids, and discarded contaminated lead shielding. A determination of the analysis requirements for RH TRU waste must identify the appropriate methodologies that are required for sufficient characterization (process knowledge, sample and analysis, etc.).

RH TRU WASTE INVENTORY

RH TRU inventory data in this report were taken from the WIPP Transuranic Waste Baseline Inventory Report (WTWBIR), which establishes a methodology for grouping wastes of similar physical and chemical properties (1). The wastes are grouped into a series of "waste profiles" that can easily be disseminated to regulatory agencies and other stakeholders. The WTWBIR is also used as the basis for radiological source term data for the WIPP Performance Assessment and for hazardous waste source term data used in

the WIPP RCRA Part B Permit and No Migration Petition. Table I summarizes the stored and projected inventory of RH TRU waste at each of the seven sites assessed.

Table I

SITE ASSESSMENTS

Facilities required for the characterization of RH TRU waste should be capable of activities such as removing, sorting, treating, repackaging, visually characterizing, certifying, nondestructive assaying, and examining containers of either solid or liquid/sludge waste streams. This study compiled data provided by the DOE sites that have RH TRU waste in storage and are projecting future generation.

Argonne National Laboratory - East: ANL-E occupies a 1,700 acre tract in DuPage County, Illinois, located southwest of Chicago. ANL-E is a DOE energy research and development laboratory that conducts a broad program of energy related sciences and serves as an important center for the study of nuclear and non-nuclear energy sources (2). RH TRU waste is generated at ANL-E from fuel examination activities at the Alpha-Gamma Hot Cell Facility (AGHCF) and is placed in metal cans, bagged-out and heat-sealed, and packaged in Department of Transportation Specification 7A 30-gallon drums. ANL-E has an RH TRU waste certification plan for shipment to WIPP but a revision to the current WIPP-WAC, Revision 4 is needed (3) . ANL-E has no planned or current facility to load RH TRU waste in canisters (for approved WIPP casks). They, therefore, will continue to ship RH TRU waste to the INEL for interim storage. ANL-E has no plans for facility modifications, new facilities, or new capabilities for characterization of RH TRU waste.

ANL-E does have two facilities that may be used for the characterization of RH TRU waste: the AGHCF and the Building 200 Hot Cell Facility. The 10,000-ft3 AGHCF complex is designed for metallurgical research and consists of a multicurie hot cell, the Electron Beam Laboratory, a small machine shop, and a decontamination/repair area (4). The hot cells contain a total of ten work stations fitted with viewing windows and equipped with master-slave manipulators. Primary auxiliary equipment at the AGHCF could be used in the examination of RH-TRU waste. It was used previously to examine and repackage 30-gal drums of RH-TRU waste. The Building 200 Hot Cell Facility consists of 2 floors of shielded cells, 12 isolated hot laboratories, and a shielded corridor for movement of highly radioactive materials between cells. A cell on the lower level was previously used for the packaging of RH TRU waste and shields against 50,000 Ci of 1-MeV gamma radiation. Two mobile and two bridge-mounted remotely-controlled polar manipulators are also available for use in areas not accessible to the master-slave manipulators.

Argonne National Laboratory - West: ANL-W is situated in the extreme southeastern portion of the INEL and has had a primary mission to develop the Integral Fast Reactor concept (2). ANL-W is an RH TRU waste generator at the INEL and has developed a RH TRU waste certification plan to document the activities necessary to characterize and certify RH TRU waste for disposal at WIPP (5). ANL-W has also completed a preliminary Data and Process Knowledge Assessment to determine the extent of the INEL's RH TRU waste that can be certified for WIPP disposal without additional characterization activities.

Two facilities at ANL-W could be used for the characterization of RH TRU waste: the Hot Fuel Examination Facility (HFEF) and the Fuel Cycle Facility (FCF). HFEF is comprised of two adjacent, shielded hot cells

that include a decontamination cell and a main cell. The main cell uses a high purity argon gas designed for containment of plutonium contamination and is arranged for vertical handling, examination, and cutup of long experiments up to 30 feet in length. The cell is designed for long-term, permanent, remote operations that do not require personnel entry. The decontamination cell is air-filled and used for the handling of irradiated experiments, the decontamination of hot cell equipment, and the loading and shipping of radioactive waste generated at the facility. Much of the in-cell examination equipment for fuel elements is automated or semi-automated. The FCF also has two adjacent cells that include an argon cell and an air cell. The 16-sided FCF argon cell was modified to support the new remote reprocessing and refabrication fuel cycle process for the former DOE Integral Fast Reactor Program. This program has since been discontinued by DOE, effectively ending the current mission of FCF. ANL-W is currently developing nondestructive assay (NDA) techniques for fast and low-cost assay of nuclear materials. These passive and active techniques are based on specific signatures of elemental/isotopic materials contained in certain waste streams and spent fuel. The techniques will be suitable for waste streams exhibiting high radiation, that contain nuclear poisons, (a,n)-sources, and lumped fissionable material in a heterogeneous matrix. The currently designed system is equipped with a 14 MeV, source providing 1011 neutrons per second during the active phase.

Hanford Reservation: HANF is located in the southeast corner of Washington State, approximately 50 miles north of the Oregon border. HANF is located in a structural and topographic depression of the Columbia Plateau called the Columbia Basin. HANF covers an area of about 560 square miles of semiarid land. The mission is to cleanup HANF, to provide scientific and technological excellence to meet global needs, and to be a partner in the economic diversification of the region (6). The retrieval and characterization of the existing RH TRU wastes at HANF will not occur before the year 2000, the earliest expected startup date of the Waste Receiving and Packaging Plant (WRAP) Module 2B. Until that time, the data supplied will be based on available engineering judgement and process knowledge.

HANF does not have existing facilities capable of characterizing the RH TRU waste currently stored at the site. Capabilities for nondestructive examination (NDE) and nondestructive assay (NDA) for contact-handled (CH) TRU waste characterization are available, but they are not readily adaptable for RH TRU. The planned WRAP Module 2B is a major line item construction project at HANF that will enable characterizing and processing RH TRU waste for shipment to WIPP (7). WRAP Module 2B will be a largely remotely-operated hot cell facility with moderate shielding requirements for RH wastes having contact radiation levels of up to 100 R/h. The facility will process the retrievably-stored RH TRU wastes and the projected RH TRU wastes that will be generated by future HANF programs.

The current concept for the WRAP Module 2B consists of a large process hot cell used for the disassembly and size reduction of large items. Adjacent to this cell are several small cells to provide for sorting, treatment, and packaging of the waste into containers. HANF plans an engineering study to determine the size of the main hot cell, the number of processing cells, and other conceptual functional elements associated

with the facility. WRAP Module 2B will have capabilities for visual examination, direct sampling, and NDE and NDA of RH TRU waste.

Idaho National Engineering Laboratory: The INEL covers approximately 890 square miles and is located near Idaho Falls, Idaho. It was established in 1949 to conduct, test, and operate nuclear facilities. Interim storage of RH TRU waste was initiated at the INEL in 1976 with the establishment of the Intermediate-Level Transuranic Storage Facility (ILTSF), located at the Radioactive Waste Management Complex (RWMC). The ILTSF was established for retrievable storage of RH TRU waste with radiation levels greater than 200 mR/hr and less than 4,500 R/hr. The INEL has also developed a test plan for headspace gas sampling of drums in a cooperative venture with ANL-E. Although no gas sampling activities have been initiated, the methodologies have been established and much of the associated operating and design documentation developed.

The INEL does not have facilities that are currently capable of characterizing RH TRU waste. The Stored Waste Examination Pilot Plant which is located at the RWMC, could be adapted for RH waste with the addition of shielding, and remote-handling equipment and with modification to the NDA measurement system and computer algorithm. Hot cell facilities at the INEL are located at the Test Area North and Test Reactor Area. These facilities may have potential application for visual characterization and direct sampling of RH TRU wastes. No descriptive information of these facilities and their current capabilities, however, could be obtained from the INEL.

The INEL is currently developing systems that will enable "at-line" headspace gas sampling of TRU waste drums. Included are a permanently installed monitoring system at the INEL Drum Venting Facility at the RWMC and the development of a portable system that can be transported on-offsite for at-line sampling. The INEL is also preparing a proposal to request funding to support completion of a mobile gas sampling and vent installation station for RH TRU waste drums.

Los Alamos National Laboratory: LANL is located in north-central New Mexico, north-northeast of Albuquerque. Since its inception in 1943, the primary mission of LANL has been the research and development of nuclear weapons. Programs include weapons development, nuclear fission and fusion research, nuclear safeguards and security, and verification and control (8). All of LANL RH TRU waste is generated by the destructive examination of experimental fuel elements from the Experimental Breeder Reactor-2, including segments of fuel discarded after metallurgical examination (9). Over the past year and a half, LANL has been characterizing and packaging RH TRU waste for ultimate disposal at WIPP. To date, 16 WIPP-approved canisters have been characterized, packaged, and placed into retrievable storage. LANL has pioneered much of the radioassay capabilities that are currently in use at many DOE sites, and it continues to research and develop additional technologies. LANL currently has the only system used in the DOE complex that is capable of assaying RH TRU waste. It was designed to specifically measure waste cans having a diameter of 21-cm and a height of 30-cm, with an overall waste volume of approximately 1-gal. The RH assay system chamber size could be increased to accept drum sizes up to 55-gal.

The LANL Chemical and Metallurgy Hot Cell Facility is a complex of 16 cells comprised of 2 banks of 8 hot cells (10). Each bank of eight hot cells has a central corridor and hydraulic-actuated doors to isolate each

cell from the corridor. The corridor has enough space to accommodate 55-gal drums, shielded waste casks, or larger overpacks.

A major LANL research and development project for TRU waste characterization is the Combined Thermal/Epithermal Neutron (CTEN) Interrogation radioassay device. According to LANL researchers, this device is highly suitable for determining the fissile content of cans of RH TRU waste derived from irradiated breeder reactor fuel pins that emit gamma radiation levels of up to 1,000 R/h (11). LANL has made significant progress towards development of a CTEN interrogation unit capable of full-scale operational applications. A fully-operational CTEN instrument that is capable of assaying both 55- and 83-gal drums has been constructed.

Oak Ridge National Laboratory: The 8,771 acre ORNL is located on the Oak Ridge Reservation and is situated almost entirely within the 6.5-square mile White Oak Creek drainage basin. ORNL's mission is to conduct research and development activities for DOE and other government agencies, as well as for private institutions (12). Over 60% of the total DOE RH TRU waste inventory is stored at ORNL.

In the late 1980s, ORNL developed a strategy to process and repackage its stored RH TRU waste that included the proposed TRU Processing Facility (TPF), a major line-item project at ORNL that was originally conceived as the Waste Handling and Packaging Plant (13). To assist in the development of the TPF, ORNL has completed a study of existing documentation to characterize much of the stored RH TRU waste through process knowledge (14). An alternative to TPF is the feasibility of using existing facilities to process TRU wastes. A report describes the existing facilities and technologies considered, the business management methods evaluated for cost effectiveness, processing schedules, life-cycle cost estimates, and final recommendations.

A real-time NDA and NDE system for compliance verification with WIPP-WAC requirements (e.g., fissile material limits and free liquids or particulates) is essential to the TPF. After review of the NDA and NDE techniques applicable to CH TRU wastes, ORNL has concluded that a linear accelerator- (LINAC) based system is likely to succeed in meeting the WIPP-WAC requirements. Development of NDE by LINAC-based RTR is nearly complete, whereas a LINAC system to perform radioassay is in the early stages of development.

Building 3517 (the Fission Product Development Laboratory) is a two-story structure constructed of masonry block, structural steel, and reinforced concrete, with a metal-sided high bay area over the cell bank. The building structure serves as a secondary confinement enclosure. Material receiving and shipping for the cells is performed in a truck bay and airlock at the west end of the building. A 10-ton dolly moves material to where it can be lifted by a 20-ton bridge crane located over the cell bank. The cell roof has removable plugs for access to each cell. The cell bank consists of two rows of cells with the north row of cells divided into nine separate rooms, but all are interconnected by a pipe chase along the full height of the cell. The south row cells are each serviced by one or two windows and one or two sets of master/slave manipulator arms. The interiors of the hot cells are highly contaminated from past operations. Although the north bank of cells is currently undergoing decontamination, the south cell bank is still used for the preparation of radioisotope sources. The soil surrounding this structure is also highly

contaminated with fission products as a result of previous waste-line leaks and spills and requires remediation.

Building 3525 (the High-Radiation-Level Examination Laboratory) is a two-story brick building with a partial basement that houses ventilation equipment. Building 3525 is divided into two functional sections: the front section is a single-story office and laboratory area, and the two-story structure in the rear houses the cell complex, the operating areas, and other support functions. The primary cell structure consists of three straight-line banks that are arranged in the form of a "U" for functional purposes. The cells on this level are served by 15 viewing windows and a pair of master-slave manipulators at each window. The inside surfaces of the cell banks are lined with stainless steel sheet to provide containment of particulate matter. Special penetrations provide for the sealed entry of services, such as instrument lines, water, and gas. Within each cell bank, heavy objects are moved by electro-mechanical manipulators and a companion 3-ton bridge crane. Special penetrations in the cell wall accommodate some basic experimental equipment and shielded and sealed transfer mechanisms. These penetrations include the sleeve ports for the periscopes, the stereo-microscope, and the collimator for the gamma spectrometer. They also include openings for the transfer of samples to shielded transfer stations provided at the rear face of the charging cell (No.7). Items up to 446 ft in size and weighing up to 10 tons may also be transferred through a shielded airlock door system.

Building 7860 (the Hydrofracture Facility) is a single-story structure constructed of reinforced concrete, masonry block, and structural steel. The building is L-shaped with overall dimensions of 5679.5 ft. There is a two-story masonry block portion for an equipment room and a three-story steel-sided portion for the well pipe storage tower. The first floor consists of equipment rooms, an office, a control room, and change rooms. There are three cells in the facility: one cell containing the well head, a second cell containing two well injection pumps, and a third cell containing the mixing equipment. Access into the cells is provided through the cell roofs and a bridge crane in the well pipe storage tower. There is no secondary containment arou`nd the cells. Building 7860 was used to mix and inject liquid waste and grout into highly impermeable shale formations located 700- to 1,000-ft below the surface. The Tennessee Department of Health and Environment will not permit any further use of this disposal technique; therefore, Building 7860 is no longer in use.

Building 7930 (the Thorium-Uranium Recycle Facility) is a three-story structure with a partial basement. It is constructed of structural steel, reinforced concrete, and masonry. The building has an irregular shape, and the structure serves as a secondary confinement enclosure. The first floor provides space for technical personnel offices, operating space for cells and maintenance, a glove-box maintenance room, a receiving area, a fuel storage basin, change rooms, a compressor, CO2 storage, and elevator machinery rooms. The second floor provides space for chemical makeup, sampling of radioactive materials, a development laboratory, a shop for handling slightly contaminated material, a maintenance area, and mechanical and electrical-equipment rooms. Located on the third floor is a high bay that includes the cell roof area and provides facilities for entry of cell services and cell access. The third floor is serviced by a 50-ton bridge crane that has a 5-ton auxiliary hoist. Heavy equipment and

casks can be brought into the facility and transported by the 50-ton crane to various locations over the cell complex.

The principal cell structure is in the shape of a "T" and consists of one straight section that contains 4 cells and another section that runs at a right angle to the first section. In addition, a lightly-shielded equipment storage cell is located adjacent to and below the bulk of the cell structure. A glove maintenance room is connected to the shielded-cell system. The glove-maintenance room is accessed through an equipment airlock. The bulk of the shielding is normal concrete, 5.5-ft thick up to the second floor level around the cells and 4.5-ft thick above the second floor. Decontamination cell shielding is 4-ft thick high-density concrete or equivalent up to the second floor level around the cells and 3-ft thick above the second floor; shielding around the equipment storage cell is 2-ft thick normal concrete. Window and master-slave manipulator openings and cell services are installed at 8-ft modular intervals in the cell face.

CONCLUSIONS

This study provides an assessment of the existing and planned capabilities to characterize RH TRU waste at major DOE sites. Based on the preliminary assessment, there are limited characterization capabilities at the sites that have RH TRU waste. These findings of existing and planned DOE capabilities to characterized RH TRU wastes are summarized below and presented in Table II.

Nondestructive examination capabilities of RH TRU waste using RTR are essentially nonexistent. RTR systems located at HANF, INEL, and ORNL are capable of examining CH TRU wastes only. Application to RH wastes would require the installation of shielding and remote-handling equipment. Some sites posses other systems such as neutron radiography that would require modifications to examine RH TRU waste containers.

Visual characterization capabilities may exist at hot cells located at ANL-E, ANL-W, INEL, LANL, and ORNL. The ANL-E, ANL-W, and LANL hot cells are relatively modern facilities that would be capable of examining RH TRU wastes, but the INEL facilities are outdated and would require upgrades prior to use.

Gas sampling and analysis capabilities exist at ANL-W and INEL for CH TRU wastes. Application of these systems to RH TRU wastes require installation of additional shielding. The INEL may complete a mobile gas sampling and vent station for RH TRU waste drums that has been designed and fabricated. Some preliminary testing been performed on a prototype system.

Table II

Currently available NDA capabilities for characterizing RH TRU waste are limited.The only system capable of neutron assay of RH TRU waste is a small passive-active neutron device located at the LANL. This system is presently capable of successfully assaying only certain waste streams generated at LANL. Application to other offsite RH TRU waste streams would require extensive system modifications. Although there are new neutron interrogation assay system technologies for RH TRU waste under development at LANL and the ORNL, these systems are unproven and are probably years from potential use in an operating environment.

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THE SYSTEMS PRIORITIZATION METHOD (SPM) CD-ROM DEMONSTRATION FOR WASTE MANAGEMENT '96*

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ABSTRACT
In March 1994, the Department of Energy Carlsbad Area Office (DOE/CAO) implemented a performance-based planning method to assist in prioritization within the Waste Isolation Pilot Plant (WIPP). Probabilistic performance calculations were required for the Systems Prioritization Method (SPM) and roughly 46,700 combinations of activities were analyzed, generating a large volume of information to be documented,

analyzed, and communicated. A self-contained information management system consisting of a relational database on a 600-megabyte CD-ROM was built to meet this need. The CD-ROM was used to store performance assessment results, data analysis and visualization tools, information about the activities, electronic copies of 40 CFR 191 and 40 CFR 268, technical reference papers, and the final SPM report. Copies of the CD-ROM were distributed to interested members of the public, WIPP participants, and the Environmental Protection Agency (EPA).

INTRODUCTION

In March 1994, DOE/CAO implemented a performance-based planning method to assist in prioritization within the WIPP project with respect to applicable EPA long-term performance requirements in 40 CFR 191.13(a) and 40 CFR 268.6 (1-5). SPM was designed to define the most cost-effective combinations of scientific investigations, engineered alternatives, and waste acceptance criteria to support the WIPP compliance certification application. Probabilistic performance calculations were required and roughly 46,700 combinations of activities were analyzed in the second phase of SPM (SPM-2). A self-contained information management system consisting of a relational database on a 600-megabyte CD-ROM was built to meet this need.

THE SPM CD-ROM INFORMATION SYSTEM

In order to make this very large volume of data available and meaningful to a wide audience, SPM-2 results were published on a CD-ROM in the form of a self-contained information system. The CD-ROM operates on WINDOWSTM platforms. To minimize incremental costs of its use, all software (except WINDOWSTM) is contained on the CD-ROM in a form that can be reproduced without violating copyright protection. A relational database on a 600-megabyte CD-ROM is used to store performance assessment results, data analysis and visualization tools, information about the activities, electronic copies of 40 CFR 191 and 40 CFR 268, technical reference papers, and the final SPM report (Fig. 1). Information contained on the CD-ROM would, if printed out, create a stack of paper over three hundred feet tall.

Fig. 1

Copies of the CD-ROM were distributed to interested members of the public, WIPP participants, and the EPA. To make the system as user-friendly as possible, the database uses a point-and-click-structure and all software needed to run the database is contained on the CD-ROM. In addition, all files are hyperlinked so that a user can explore regulatory requirementssuch as human intrusionfrom a point of origin in the Federal Regulations through their treatment in technical references. Over fifty online help panels provide introductory material on such topics as the WIPP mission, transuranic waste, performance assessment, and a step-by-step explanation of the SPM calculations. The CD-ROM also contains an online software manual and hard copy reports can be generated for each activity.

One of the objectives of the CD-ROM information system is to promote efficiency (and quality) by providing as much relevant data as possible to support users' understanding of the SPM-2 data, results, and calculations. To that end, a graphical interface is provided to orient the user and help navigate through the relational database system. In addition to providing a contextual background for understanding the meaning of data displayed on any panel, four levels of explanatory information are provided.

The Data Visualizer

The heart of the SPM-2 information system is a data visualizer containing performance measures (cost, duration, and probability of demonstrating compliance (PDC)) for roughly 46,700 programmatic options (activity sets). The data visualizer is implemented using a commercially available off-the-shelf database management system for data storage and retrieval, computations, data display, and graphical user interface.

The database management system provides interactive functions for selecting groups of activity sets for analysis and visualization based on performance measure ranges, presence or absence of particular subsets of activities in the activity sets of interest, or both. The visualization system provides six views of selected activity set groups: two-dimensional scatter charts with three-axis pairs (cost-PDC, duration-PDC, cost-duration); a three dimensional view plotting maximal PDC for activity sets whose cost and duration values fall within user defined ranges; and a sorted list display showing activity composition, PDC, cost, and duration for each selected activity set. The system interactively displays details of activities composing each activity set and the complementary cumulative distribution functions (CCDFs) corresponding to the selected activity set outcome. The database component of the system is built of more than 200 display panels, forms, tables, queries, reports, and macros.

In a database, numbers standing alone without annotation or explanation are without meaning to many, if not most, users. Yet most conventional uses of databases provide nothing more than numbers. This often means that the user may have to expend significant effort on finding documentation for the numbers provided in a database in order to adequately understand their meaning. To the extent that effort expended on finding explanations for the meaning of numbers in a database is effort that could be directed to using the numbers productively, the lack of explanation conveniently at hand while using the database represents an element of inefficiency.

Metadata

Within the visualizer display panel, titles, table column headings, table row labels, and graph legends are supported with metadata: information that defines, explains, and relates each of these data entities to WIPP, to the SPM-2 process, and to the scientific and engineering activities whose consideration produced the SPM-2 results. As a minimum, text provides brief definitions and explanations. These brief annotations are linked to a detailed online overview, which is internally hyper-linked to pop-up definitions, to related topics, and to a glossary often found in the system. Annotations are linked to the online electronic document library where detailed technical references for the WIPP project are provided.

In addition to the passive metadata, items of particular importance are supported by active-interactive panels. For example, one of the three principal display panels in the visualizer, the "Activity Set Composition Panel," is an interactive table. Its rows correspond to individual activity sets. It has two sets of columns. The first set of columns correspond to technical program areas and the second set correspond to the performance measures: PDC, cost, and duration. Clicking on a column heading activates a pop-up panel that displays a "passive" text definition of the technical program area associated with that column; this definition is hyper-linked to an appropriate destination in the

electronic document library. But if the activity set number that labels a row is clicked, an interactive panel is activated. This panel expands the corresponding row of the "Activity Set Composition Panel," displaying detailed information about every activity in the set. This panel computes and displays the PDC for any combination of outcomes for the set's component activities the user selects and upon the user's request will access the necessary data and generate the CCDF chart for the selected outcome combination. This interactive functionality allows the user to exercise one of the most conceptually elusive computations in the SPM-2 analytical process to gain a concrete grasp of the mechanics of the operation, providing an invaluable complement to the detailed explanation of the computation contained in the overview.

Both passive annotations and interactive panels are an integral part of the database structure along with the SPM-2 results. Every row, every column, every cell has an integrated information component which can be used for the appropriate level and detail of definition, explanation, or demonstration.

The Navigator

The navigator (Fig. 2), which acts as the user's portal to the system, provides point-and-click access to all of the other components of the system. The navigator is implemented using an off-the-shelf hyper-text/full text search system. Navigation to pre-defined points in the textual portions of the system is accomplished by HYPERLINKTM. Ad-hoc navigation is supported by a sophisticated full text search system that provides virtually instantaneous shift of focus to locations of topical interest in the text defined by the user. Topical interest can be formulated as words, phrases, and character strings (any of which may include fixed or variable length "wild cards" used to ignore irrelevant variations in textual expression of an idea of interest). These elemental topical recognition cues can be combined with both Boolean (and, or, not) and proximity (e.g., these two words within so many characters or so many words of each other) operators to express complex and precise search specification. The query formulation dialog is designed to be intuitive and easy to use.

Fig. 2

Search time is invariant with respect to query complexity and sub-linear by several orders with respect to the magnitude of text searched. This aspect of system performance is of importance quite beyond the dimension of mechanical efficiency. When pursuing an inquiry regarding the meaning of a particular data set, the more rapidly the system can respond, the more questions can be resolved within a unit of time.

The system provides a complete history of the user's trail through the text (whether by hyperjump or by topical search), displaying a list of all locations visited. The user can click on any item in the history and automatically return to that point in the text. A back-track function allows the user to retrace the trail one step at a time. These functions are an invaluable aide to understanding by allowing the user to leave a particular context to follow explanations and related information and then return with little effort to the original context armed with additional relevant information to augment understanding of the original context.

Online Context-Sensitive HELP

With over eighty hyper-linked topics, online context-sensitive HELP contains explanations of how to use the system, descriptions of each

display control (buttons, check boxes, etc.), and definitions of displayed data items. Brief definitions of SPM-2 terminology are also provided in the software HELP subsystem.

Many of the help panels are graphical and interactive. For example, a multi-level hyperlinked user structure diagram shows the relationships among the user interface panels. Clicking on a block in the structure diagram that represents a particular user interface panel hyperlinks to an image of the panel itself. The panel image, in turn is hyperlinked to explanations of the meaning of data displayed on the panel and instructions on how to use it.

Overview

Definitions of SPM-2 terminology as well as detailed descriptions of WIPP and the SPM-2 project and its objectives, methods, and computational procedures are contained in the SPM-2 Overview. This electronic document is internally hyper-linked, and hyper-linked to the database/visualizer system and to the electronic document library described below. The Overview is also full text content searchable.

Electronic Document Library

The fourth level of explanatory data, the electronic document library (EDL), is also internally hyper-linked and can be content accessed via full text search. The EDL contains the SPM-2 Final Report, the SPM-1 Final Report, the technical position papers which led to the computation of the data contained in the database/visualizer, a set of general references, and the full text for 40 CFR 191 and 40 CFR 268.6. The general references, which consist of SPM-2 charter documents and other key instructions establishing the scope and direction of SPM-2, are included as electronic page images (containing signatures, etc.) and as electronic text documents that can be full text searched.

EVOLUTION AND USES OF THE SPM-2 CD-ROM

The magnitude of the data and results suggested using a database to organize the results in a manageable form. A visualizer was included to enhance the value and utility of the data. The estimated size of the database suggested that a high capacity, high density storage medium would be needed. At about the same time as the storage medium became a consideration, DOE direction to make the SPM-2 data and results available to the public and the consequent need for distribution of many copies introduced the requirement for a low cost storage medium. The CD-ROM was the obvious choice. Release to the public also led to development of the navigator, metadata, and all of the other supporting information and tools that resulted in self-contained form of the final system.

Copies of the CD-ROM have been distributed to interested members of the public, WIPP participants, and the EPA. The system allows project members to understand parts of the project on which they had not participated. The supporting explanatory material in the system may be of interest to a wider audience than the results and data and may continue to be of value well after the original purpose of the system has been served.

CONCLUSIONS

The SPM performance-based decision methodology proved to be useful in structuring a large, complex decision and focusing attention on important portions of the problem.

A self-contained information system to support a specific task is made possible in part by the availability of information in electromagnetic form and by contemporary information system tools; like the personal computer, large capacity data storage media, fast processors, large ram

memory, low cost database management systems, HYPERTEXT and search engines, etc. The CD-ROM is a key element for distribution of self-contained information systems. The SPM-2 self-contained information system could not have been economically feasible as recently as three years ago. Just since the SPM-2 CD-ROM was released, there have been substantial improvements in all of the software and hardware elements used in its creation. Later this year CD-ROMs with a capacity of 6 gigabytes will be commercially available. A technology for manufacture of 100 gigabyte CD-ROMs is nearing maturity and is expected to be brought into manufacture within the next year or two. Multispectral approaches to recording and reading optical data are expected to yield CD-ROM systems with even higher capacities within the next five to ten years. The CD-ROM's cost performance is superior to any paper form or even any microform. When large numbers of copies are required, the per disk cost of the CD-ROM is small. The economies of distribution via CD-ROM are so compelling that there should be a conversion of many publication and record keeping requirements to this medium.

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ASSURING PROPER PERFORMANCE OF THE WASTE ISOLATION PILOT PLANT BY USING ENGINEERED ALTERNATIVES*

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ABSTRACT

The cost and benefit of continuous improvement: using Engineered Alternatives at the Waste Isolation Pilot Plant.

The Waste Isolation Pilot Plant (WIPP) faces many challenges in its bid to open as the nation's first Transuranic (TRU) nuclear waste repository. One of these challenges is to satisfy stakeholder's and regulator's concerns that the repository will perform as predicted, given the uncertainty associated with the long-term predictions of repository performance using a complex system of computer models. The Department of Energy (DOE) is committed to providing a safe and effective means of disposal for these TRU wastes. To provide assurance in the performance of the WIPP facility, the DOE has developed an Engineered Alternatives program.

An Engineered Alternative may take the form of a facility modification, a process modification, or waste form modification so as to address uncertainty which may be associated with important WIPP performance parameters (i.e. transport of radionuclides, increases in the mechanical strength of disposed wastes, the quantity of waste that would be released to the accessible environment in the event of a drilling intrusion). To aid in the evaluation of Engineered Alternatives, the DOE initiated the Engineered Alternatives Cost/Benefit Study in late 1994. The study examined potential alternatives and systematically screened them into a practical and technically feasible set. The study then examined this set of engineered alternatives for their benefits in addressing important WIPP performance-related uncertainty. Examples of the engineered alternatives that were examined include engineered backfills, plasma arc processing of wastes, supercompaction of wastes, repository configuration modifications, and combinations of these technologies. The study addressed the application of engineered alternatives to existing wastes, currently generated wastes, and wastes to be generated in the future. This report details the methodology employed to assess the impacts of each of these alternatives upon the entire DOE complex as well as the impacts on WIPP operations, and long-term repository performance.

In general, the DOE found that risks to workers and the general public are greater for complex treatment options than for simpler technologies such as engineered backfills. Every EA provides performance benefits and associated risks. The DOE must weigh these benefits and risks when making a final decision relative to EAs at the WIPP facility. This study will be used as a means to make this decision.

INTRODUCTION

The Waste Isolation Pilot Plant (WIPP) is a United States Department of Energy (DOE) pilot operation designed to demonstrate the safe disposal of Transuranic (TRU) waste in deep, bedded salt. The WIPP site is located in southeastern New Mexico. By law (U.S. Congress, 1992) the WIPP site has been withdrawn from public use to demonstrate the safe disposal of TRU waste. Also by law, disposal of TRU waste must comply with rules and regulations promulgated by the U.S. Environmental Protection Agency (EPA). The disposal system design consists of multiple barriers, both natural and man-made, located in a geologic salt deposit, 655 meters below ground. These barriers were selected because of their ability to permanently isolate the waste from the accessible environment and, at the same time, comply with performance and assurance requirements in subparts B of Title 40 Code of Federal Regulations Part 191 (40 CFR 191). This

regulation contains; 1) repository radionuclide containment requirements that must be demonstrated using a performance assessment and 2) specific assurance requirements used to provide additional confidence in the containment predictions. Engineered Alternatives (EAs) can be used to provide additional assurance measures beyond those used to meet the existing containment and assurance requirements. This paper uses the term EA to represent engineered barriers that are technically feasible processes, technologies, methods, repository designs, or waste form modifications which make a significant positive impact on the disposal system in terms of reducing uncertainty in performance calculations or improving long-term performance.

The DOE conducted a cost/benefit study to evaluate EAs for potential use as assurance measures. The purpose of this study is to provide the DOE with cost, benefit, and risk information for use in the selection or rejection of EAs. This study includes an assessment of estimated cost, potential risks, benefits, and relative repository performance impacts from the implementation of EAs, and where appropriate, the impact on the entire waste management complex (as a system) was considered. This study is entitled the Engineered Alternatives Cost/Benefit Study (EACBS).

The EACBS evaluated EAs using the following assumptions and guidance.

The present baseline design of the WIPP repository and its predicted performance meet the containment requirements of 40 CFR 191 without additional EAs. The baseline does not include waste processing above that required by the WIPP Waste Acceptance Criteria (WAC).

The results of the EACBS analysis are qualitative. However, both qualitative and quantitative methods were used to generate the output information.

The EA analysis used a multi-factor approach to evaluate the cost; the risk, both incidental and accidental; the benefit and schedule impacts that could be expected from the implementation of each individual EA. The factors were not ranked or weighted. The output of the EACBS compares the results of the EA analysis with the baseline and not to each other.

The approach used in the EACBS was to screen potential EAs compiled from previous studies, proposed regulations, and input elicited from stakeholders. The screening process used a working group composed of technical professionals from various fields to compare the proposed EAs to an EA definition and then to determine if those EAs that meet the definition also meet regulatory and technological feasibility criteria. The output of the screening process was a list of EAs that did not meet the definition and/or screening criteria along with the justification for their rejection from further study, and a list of EAs retained. This list of retained EAs was then optimized to determine which EAs would be analyzed further.

The screening processes evaluated 111 proposed EAs and screened them to a field of 54. The 54 EAs retained were optimized by the DOE using feasibility and effectiveness criteria to provide the final set of 18 EAs used in the EACBS. The 18 EAs remaining for evaluation consisted of nine basic alternatives and nine variations. The 18 final EAs and the baseline are briefly described below. (Numbers refer to their listing in the EACBS; they do not represent rankings or ratings.)

Baseline For EA comparison, the baseline is considered to be the current WIPP disposal system design. Waste meeting the WIPP WAC is emplaced in rooms that are approximately 4 meters high, 10 meters wide, and 91 meters long and access drifts in waste stacks of seven-pack drums (three high)

and Standard Waste Boxes (three high). No backfilling of the waste disposal area is included in the baseline case.

#1Supercompact Organics and Inorganics Solid organic and inorganic wastes are sorted to remove items that cannot be compacted. Sorted waste is pre-compacted in 132.6 liters (35-gallon) drums and then supercompacted. Usually, the contents of four supercompacted drums are placed in a 208-liter (55-gallon) drum. Sludges are not processed.

#6Shred and Compact Organics and Inorganics Solid organics and inorganics are shredded and compacted in 208-liter (55-gallon) drums using a mechanical shredder and a low pressure compactor. Sludges are not processed.

#10Plasma Processing of All Wastes All wastes are processed through a mechanical shredder and the input waste stream is controlled to ensure a suitable metal to non-metal ratio. The waste is processed through a Plasma Arc Centrifugal Treatment System and placed into 208-liter (55-gallon) drums.

#33Sand Plus Clay Backfill A mixture of medium grained sand and granulated clay is used as backfill. The mixture is placed around the waste stack and between the drums filling the void space between drums and unmined host salt in waste emplacement panels. A 50 percent void space is assumed.

#35aSalt Aggregate (Grout) Backfill A salt aggregate grout mixture is used as backfill to fill the void spaces between drums and unmined host salt in waste emplacement panels. This backfill consists of a cementitious-based salt aggregate grout with crushed salt aggregate and is pumped around the waste stack and between the drums filling the void spaces. A 20 percent void space is assumed.

#35bCementitious Grout Backfill A cementitious grout backfill consisting of ordinary Portland cement, sand and fresh water is pumped around the waste stack and between the drums filling the void space. A 20 percent void space is assumed.

#77aSupercompact Organics and Inorganics, Salt Aggregate/Grout Backfill, Monolayer of 2000 drums in a room that is 1.83 meters (6 feet) high, 10.06 meters (33 feet) wide, and 91.44 meters (300 feet) long Alternatives #1 and #35a are combined. The room height is lowered from 3.96 meters to 1.83 meters (13 feet to 6 feet) and only one layer of drums is emplaced in the room.

#77bSupercompact Organics and Inorganics, Clay-Based Backfill, Monolayer of 2000 drums in a room that is 1.83 meters (6 feet) high, 10.06 meters (33 feet) wide, and 91.44 meters (300 feet) long Alternatives #1 and #111 are combined. The room height is lowered from 3.96 meters to 1.83 meters (13 feet to 6 feet) and only one layer of drums is emplaced in the room.

#77cSupercompact Organics and Inorganics, Sand/Clay Backfill, Monolayer of 2000 drums in a room that is 1.83 meters (6 feet) high, 10.06 meters (33 feet) wide, and 91.44 meters (300 feet) long Alternatives #1 and #33 are combined. The room height is lowered from 3.96 meters to 1.83 meters (13 feet to 6 feet) and only one layer of drums is emplaced in the room.

#77dSupercompact Organics and Inorganics, CaO Backfill, Monolayer of 2000 drums in a room that is 1.83 meters (6 feet) high, 10.06 meters (33 feet) wide, and 91.44 meters (300 feet) long Alternatives #1 and #83 are combined. The room height is lowered from 3.96 meters to 1.83 meters (13 feet to 6 feet) and only one layer of drums is emplaced in the room.

#83Salt Backfill with CaO A backfill of commercially available granulated lime (also called quick lime which consists of CaO) and crushed salt are

placed around the waste stacks and between the drums filling the void space. A 50 percent void space is assumed.
#94aEnhanced Cement Sludges, Shred and Add Clay-Based Materials to Organics and Inorganics, No Backfill EA 94a includes two processes to treat the TRU waste. The first is an enhanced cementation process of previously solidified and "as generated" sludge. Existing sludges are fed into a mechanical crusher/shredder. The crushed waste is mixed with an enhanced cement and the product is poured into 208-liter (55-gallon) drums. Newly generated sludges are solidified with the enhanced cement. The second process shreds solid organic and inorganic wastes and adds clay to the shredded waste. This waste product is packaged in 208-liter (55-gallon) drums.
#94bEnhanced Cement Sludges, Shred, and Add Clay-Based Materials to Organics and Inorganics, Sand/Clay Backfill Alternative #94a and #33 are combined.
#94cEnhanced Cement Sludges, Shred and Add Clay-Based Materials to Organics and Inorganics, Cementitious Grout Backfill Alternative #94a and #35b are combined.
#94dEnhanced Cement Sludges, Shred and Add Clay-Based Materials to Organics and Inorganics, Salt Aggregate Grout Backfill Alternative #94a and #35a are combined.
#94eEnhanced Cement Sludges, Shred and Add Clay-Based Materials to Organics and Inorganics, Clay-Based Backfill Alternative #94a and #111 are combined.
#94fEnhanced Cement Sludges, Shred, and Add Clay-Based Materials to Organics and Inorganics, CaO/Salt Backfill Alternative #94a and #83 are combined.
#111Clay-Based Backfill Backfill consisting of commercially available pelletized clay is placed around the waste stack and between the drums, filling the void space. A 50 percent void space is assumed.
EACBS ANALYSIS FACTORS
The 18 EAs were analyzed with respect to the factors listed and described below:
1. The effects of EAs on long-term performance of the disposal system. This factor analyzes the EA's ability to limit water and radionuclide movement to the accessible environment and the potential consequences of human-initiated processes or events.
2. The increased or reduced uncertainty in compliance assessment.
3. The impact on public and worker exposure to radiation (at WIPP and off-site) both during and after the incorporation of an EA.
4. The increased ease or difficulty in future removal of the waste from the WIPP disposal system.
5. The increased or reduced risk (physical accidental, incidental and accidental chemical and radiation exposures) of transporting the waste to the WIPP.
6. The increased or reduced public confidence in the performance of the disposal system.
7. The increased or reduced total DOE waste management system cost and schedule impacts.
8. The impact on other waste disposal programs.
The following discussions outlines the analysis and results for each EA with respect to the eight factors.
Factor 1Effects of EAs on Long-Term Performance of the Disposal System
Factor 1 dealt with the impacts that an EA is predicted to have on the

long-term performance (not specific to the regulatory requirements) of the disposal system. Impacts were predicted using the Design Analysis Model (DAM), a computer model which considered the coupled processes of brine inflow, creep closure, gas generation, and radionuclide migration under undisturbed conditions. The consequences of three human intrusion scenarios were also considered. The three human intrusion scenarios postulated the existence of future boreholes that inadvertently penetrated the waste rooms and panels (waste horizon). These three scenarios were referred to in the EACBS as E1, E2, and E1E2. The E1 scenario involved a borehole that intersects the repository and a hypothetical pressurized brine pocket below the repository. The E2 scenario involved a borehole that only intersects the repository, and the E1E2 scenario involved one E1 and one E2 borehole that intersect the repository at different locations and times. This factor was evaluated by considering the impacts of each EA on the following:

Relative changes in the cumulative 10,000-year release of radionuclides based purely on the quantity of cuttings released to the surface from each of the three human intrusion scenarios

Relative changes in the cumulative 10,000-year release of radionuclides into the overlying Rustler Formation from each of the three human intrusion scenarios.

The impacts of each EA are expressed as changes in the parameters described above relative to the baseline, which is defined as unprocessed waste emplaced in disposal panels.

Although both disturbed and undisturbed conditions were simulated, the greatest consequences of releases are expected to occur as a result of human intrusion. Therefore, the study placed emphasis on the effects of EAs on mitigating releases from the human intrusion scenarios.

Factor 2The Increased or Reduced Uncertainty in Compliance Assessment Factor 2 estimated the EAs ability to treat uncertainty relative to the quantity of radioactive materials that are expected to be transported to the accessible environment as a direct result of human intrusion scenarios. This factor estimated the uncertainties by systematically manipulating the DAM input parameters from the Factor 1 analyses using a Monte Carlo simulation for each EA analyzed. The results of Factor 2 were then used in conjunction with those of Factor 1 to characterize the potential for an EA to provide additional assurance in the performance of the disposal system.

The treatment of uncertainty in compliance assessment can be realized by reducing both the magnitude of radioactive materials released to the accessible environment and characterizing the potential variability in that quantity. Factor 1 addressed the magnitude of this reduction through a Measure of Relative Effectiveness (MRE) for cuttings removal to the surface and groundwater transport to the Culebra Dolomite via the borehole, given scenarios E1, E2, or E1E2 occur. A MRE is a unitless factor that expresses the change in the magnitude of releases with respect to the baseline disposal system design. Factor 2 addressed the ability of the EAs to treat the uncertainty about these estimates of release quantity by treating the uncertainty about predictions of quantities of radioactive material that might be released as a result of the intrusion scenarios.

Factor 3The Impact on Public and Worker Exposure to Radiation Both During and After the Incorporation of an EA This factor characterized the human-health risks (incidental and accidental exposure) associated with the

implementation of an EA, including those impacts realized at the WIPP site and generator or disposal facilities that handle TRU or TRU-mixed waste. Potential impacts include radiation effects (both occupational exposures and the release of material resulting from an off-normal accident scenario), effects from the release of hazardous material, and, in the case of individuals within the facilities, ordinary industrial hazards. Impacts were considered for the following five groups of individuals at the WIPP and at the generator/disposal sites:

Workers directly involved with handling, processing, or storing TRU waste (generally referred to as "workers")

Other workers in the facility who are not directly involved with the TRU waste (referred to as "co-located workers")

The co-located worker who receives the highest exposure to radiation or hazardous material from TRU waste activities

Members of the public who live within 80.5 kilometers (50 miles) of the facility where the TRU waste is being handled, processed, or stored (generally referred to as "public")

The member of the public located off-site who receives the highest exposure from activities associated with TRU handling, processing, or disposal (often called the Maximum Off-Site Individual or MOI).

Factor 4The Increased Ease or Difficulty in Future Removal of the Waste from the WIPP Disposal System For the purpose of this report, waste removal is defined as the activity involving recovery of the waste after repository closure. In assessing the waste removal activities, the waste inventory and physical properties for each EA determine the underground panel geometry that would in turn determine the time required for underground removal (mining of the waste). Underground waste removal considered the compressive strength and density of the waste form as well as the consolidation of the backfill expected to occur after a specified period of time (if applicable). The occupational hazards for industrial accidents include the conventional hazards due to underground mining accidents, hazardous waste exposure, and radioactive waste exposure.

Factor 5The Increased or Reduced Risk of Transporting the Waste to the WIPP The transportation risk factor consisted of the human-health impacts due to radiation- and hazardous-material exposures that could potentially result from transporting CH- or RH-TRU waste. The risk factor was defined in terms of the radiological, chemical, and non-radiological/non-chemical impacts of either normal, incident-free transportation or transportation accidents. Because transportation does not impact all EAs; backfill only alternatives were not analyzed using this factor. The results broke down the total number of shipments from each storage/generator site and present the exposures to the public and workers. Where applicable, reported transportation risks and exposures are in the same units used in Factor 3.

Factor 6The Increased or Reduced Public Confidence in the Performance of the Disposal System This study was conducted in two phases to identify both historic and current public concerns about WIPP's post-closure performance. During Phase 1, existing public commentary was examined to identify concerns about post-closure WIPP. These comments and concerns were further analyzed to determine the relative frequency of the concerns and the persistence of concerns over time. Data sources included:

The WIPP FSEIS (DOE, 1990b)

Response to Comments for Amendments to 40 CFR Part 191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and TRU Radioactive Wastes (EPA, 1993)

Public Hearings on EPA's Proposed Rule 40 CFR Part 194, Criteria for the Certification and Determination of the WIPP's Compliance with Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level, and TRU Radioactive Wastes, March 21-24, 1995 (EPA, 1995)

During Phase 2, comments were collected during a series of focus group discussions and interviews in which participants were invited to share their concerns.

The combined findings from Phase 1 and Phase 2 analyses serve as considerations for selecting engineered alternatives that would address expressed public concerns. A qualitative assessment is made using the comment categories (comments were segregated based on the general nature of the concern) and determining which EAs address the concerns within these categories.

Factor 7The Increased or Reduced Total DOE Waste Management System Cost and Schedule Impacts Factor 7 analyzed increased or reduced cost and schedule impacts from implementation of EAs on the total DOE waste management system. The cost consists of summarized waste processing, transportation, backfill, and emplacement handling for the selected alternatives. The analyzed costs include a comparative analysis of the incremental change in cost of the alternatives relative to the repository baseline. This analysis estimated the level of funding necessary to implement an EA, the estimated manpower for the activities, and a conceptual schedule that provides start and stop dates for each EA analyzed. Cost was analyzed by developing process flow diagrams that segment the alternative into conceptual elements. The costs for the alternatives were developed on the basis of waste quantities and required throughput rates to meet the schedule constraints. The schedule analysis provides a measure of the time required to implement an EA relative to the baseline. The schedule included the incremental change of implementing an alternative on the baseline.

Factor 8The Impact on Other Waste Disposal Programs This factor included an assessment of the impacts that the EAs will have on other DOE waste processing and disposal programs, including programs for LLW and low-level mixed waste (LLMW). Major impacts were assessed based on the additional volumes of waste that were projected to be generated by TRU waste processing with respect to each EA.

Each EA can be classified in three categories, processing, backfill and combination alternatives. Processing alternatives modify the waste into a different chemical or physical matrix, backfill alternatives involve placing engineered materials around the unprocessed waste in the repository, and combination EAs which include combinations of processing, backfill, and repository design modifications.

The analysis considered three processing scenarios which differ based on where the processing was accomplished. Each processing EA was analyzed for regionalized, centralized, and decentralized processing scenarios. Regionalized processing, involved processed all of the waste at five of the major generator sites, centralized processing involved processing all of the waste at WIPP, and the decentralized processing scenario processed waste at the 10 major generator sites with all minor sites shipping their waste to the nearest major site.

OVERALL CONCLUSION OF THE EACBS

After a decision is made concerning the use of EAs at WIPP for additional assurance purposes, any subsequent selection of EAs will be made using total disposal system knowledge of the impacts related to the implementation of an EA. The EACBS report provides comparative information concerning cost, schedule, worker and public radiological/chemical and accidental/incidental risks, disposal system performance impacts, public perception, waste removal impacts, and the impact on other waste disposal systems. The process for the selection or rejection of EAs will use this and other related information to weigh the relative importance and to determine which EAs will be implemented. The information in this report will not be used as the sole bases for the selection/rejection of any individual EA.

The 18 alternatives were evaluated in the eight factor analysis yielding both qualitative and quantitative results. The analysis of each EA resulted in a large amount of data that necessitated the development of a more logical presentation of the results. The analysis results were compiled in a tabular summary and converted into quantifiable performance measures. Some factors were reported with one measure, while other factors could not be adequately expressed with a single measure. Table I summarizes the performance measures and units presented for each factor. A simplistic qualitative representation of the results was made from the performance measures comparing each alternatives performance with respect to the baseline repository design performance. Figure 1 summarizes this output information. As is the case for any analysis, these results are only as good as the models, data, and assumptions used in the analysis. These models, data, and assumptions are based on the best available current information. Technological understanding of many topics considered in this analysis is advancing rapidly, however, and it should be noted that changes in the modeling system or the model input, such as possible changes in our understanding of the future performance of specific EAs, could lead to somewhat different results. General observations are listed below.

The EAs can be separated into three general categories, Waste Processing, Backfill, and Combinations of these alternatives. The following observations were noted from the results of this analysis.

Waste Processing alternatives (EA # 1,6 & 10) were analyzed for the three processing scenarios (centralized, regionalized, and decentralized). Each scenario has inherent benefits and detriments. In general, processing alternatives impact the entire waste disposal system, involving the generator/storage sites, waste transportation, other waste disposal systems, and the WIPP waste handling system. Processing alternatives have higher cost, increased risks, and present increased schedule delays in comparison to baseline or backfill only EAs. In general, processing EAs have a marginal performance impact on the repository except for plasma processing (EA# 10) which shows a significant increase in repository impact, however, at the expense of the highest potential risk for all of the EAs analyzed.

Centralized ProcessingSince the centralized scenario processes all waste at one facility, the construction and operational costs are the lowest of the three waste processing scenarios. Operational and construction incidents and fatalities and public and worker chemical and radiological exposure risks are higher than the baseline. Transportation impacts are similar to the baseline. The centralized scenario has the highest

potential to impact system wide disposal operations. Since one facility processes all waste, this facility becomes a potential choke point for the entire system.
Regionalized ProcessingThe regionalized scenario processes waste at five generator/storage sites. The cost to implement regionalized EA scenarios are significantly higher than the centralized and slightly lower than the decentralized scenarios. In general, the worker and public radiological/chemical exposure risks are slightly higher than the centralized and lower than the decentralized scenarios. Transportation chemical exposure risks are slightly lower than the baseline since the waste is processed into a more inert matrix prior to shipment to WIPP. Accident and radiation risks are similar to the baseline.
Decentralized ProcessingFor this scenario, processing is performed at the ten major generator/storage sites. The scenario has the highest cost of the three processing scenarios (as much as $1 billion difference between the centralized and decentralized for EA# 77a-d). The operation/construction incidents and fatality rates are generally higher than both the centralized and regionalized (baseline included).

Backfill alternatives (EA# 33, 35a, 35b, 83 and 111) have the least impact on the entire waste disposal system. The WIPP waste handling system is impacted; waste transportation, generator/storage sites, and other waste disposal systems are not affected. Cost, schedule radiation and chemical exposure are similar to the baseline estimates. Backfill alternatives improve long-term disposal system performance.

Combination alternatives contain both multiple processing alternatives and/or backfill alternatives. These alternatives (EA# 77a through 77d and 94a through 94f) have benefits and detriments associated with each individual alternative type. The overall cost and schedule impacts are the highest of the EAs. Transportation, worker and public risks (radiological, chemical accidental and incidental) are also the highest of all EAs. The overall impact of combination EAs on long-term disposal system performance are comparable to that associated with the backfill and processing only alternatives.
Table I
Fig. 1

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PROJECTED TRANSURANIC WASTE LOADS REQUIRING TREATMENT, STORAGE, AND DISPOSAL*

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ABSTRACT
Argonne National Laboratory's WASTE_MGMT computational model was used to calculate the volume of transuranic (TRU) waste loads requiring treatment, storage, and disposal at facilities located at various U.S. Department of Energy (DOE) sites. Inventory and generation data were taken from DOE's Waste Isolation Pilot Plant Transuranic Waste Baseline Inventory Report published in February 1995. Results indicate that WIPP's design capacity is sufficient for disposal of the contact-handled (CH) TRU waste located throughout the DOE Complex. Argonne compared the newly estimated waste loads with estimates from a previous study that used inventory and generation data published in 1992 and 1993. The differences

between the old and new estimates, expressed as a percentage of the newly estimated waste loads, range from a few percent to about 60% for treatment of CH TRU waste and from about 10% to 30% for its storage and disposal.

INTRODUCTION

Data on TRU waste loads are important input for validating existing capacities and determining the size and cost of new facilities for TRU waste treatment, storage, and disposal (TSD) for various siting configurations. Such data are used to assess transportation requirements and health risks to workers and the general public. Two of the key parameters for calculating TRU waste loads are the current inventory and estimated generation of TRU waste. A study by Hong et al. (1) projects waste loads estimated on the basis of inventory and generation data published in 1992 and 1993. Recently, updated inventory and generation data have been made available (2,3). This paper compares the waste load information calculated from the updated data with the information presented in the earlier study.

Before TRU can be disposed of at WIPP, it needs to be packaged or treated to meet various requirements. This study considers three levels of TRU waste treatment. The minimum level simply processes and packages TRU waste; it fulfills current WIPP waste acceptance criteria (WAC) requirements. The intermediate level reduces the gas generation rate of TRU waste after it has been disposed of in WIPP; it fulfills the requirements of the performance assessment study in addition to those of WIPP WAC. The most extensive level of treatment further destroys or stabilizes all hazardous constituents in TRU waste; it meets land disposal restrictions (LDRs).

This study considers three siting configurations for TRU waste treatment. In the regional configuration, TRU waste from individual sites would be sent to regional centers for treatment; in the decentralized configuration, it would be treated on the site where it was generated; in the centralized configuration, it would be sent from each of the sites to a centralized center for treatment. All treated TRU waste is temporarily stored in an interim storage facility before being shipped to WIPP for disposal.

UPDATED INVENTORY AND GENERATION DATA

The updated inventory and generation data are from revision 1 of the Waste Isolation Pilot Plant Transuranic Waste Baseline Inventory Report (WIPP-BIR) published in 1995 (2). These WIPP-BIR data are essentially the same as those provided in the Integrated Data Base (IDB) for 1994 (3). In general, these newly published data are more complete, consistent, and accurate than the data found in a previous study, which used data from the IDB for 1992 and from the interim mixed-waste inventory report published in 1993. The updated data were recently collected directly from each TRU waste generation or storage site. The latest data on the inventory volume of contact-handled (CH) TRU waste found throughout the DOE Complex is 73,000 m3; it was previously reported as 65,000 m3. The latest generation volume is 51,000 m3; the previously reported volume was 33,000 m3. Prediction of TRU waste volumes varies at each individual site.

The WIPP-BIR also provides detailed information on the characteristics of the TRU waste in each waste stream. On the basis of this information, waste streams were grouped into categories to facilitate their efficient treatment. The categories are aqueous liquid (1000), organic liquids

(2000), solid process residues (3000), soils (4000), debris (5000), special (6000), inherently hazardous (7000), and unknown (8000). The numbers in parentheses are abbreviated designations used in the accompanying tables and figure. TRU waste in each category would be treated in a specific treatment train that involves a series of treatment technologies including solidification, shredding, incineration, and packaging.

Figure 1 illustrates the treatment trains for five TRU waste stream categories being treated at a high level to meet LDRs. All treatment trains include a pretreatment step that segregates the waste into streams by separating liquids from solids or sorting out solids that have different physical properties. Currently, waste load calculations do not include three waste stream categories (special, inherently hazardous, and unknown). These three constitute less than 10% of total TRU waste volume and are assumed to be set aside to await special processing.

ESTIMATED WASTE LOADS

Argonne used the WASTE_MGMT computational model (4) to calculate TRU waste loads. Input consisted of the updated data from WIPP-BIR for technologies at each treatment site and for each siting configuration. WASTE_MGMT accepts three types of data as input: 1) the volume, mass, and contaminant characteristics of the waste stream inventory, by generating site and waste stream category; 2) TSD unit operating parameters; and 3) site configurations for treatment. Some TSD processes generate secondary output streams that are also tracked through the treatment process. For example, the primary output stream of incineration is ash, but a secondary stream of high-chloride salt waste is generated in the off-gas treatment of combustion gases (Fig. 1).

Fig. 1

Table I shows the annual CH TRU waste loads throughout the DOE Complex requiring representative treatment technologies for three siting configurations and three treatment levels: 1) decentralization of treatment to meet WIPP WAC requirements at each DOE site; 2) regionalization of treatment to reduce gas generation rates at five DOE sites; and 3) centralization of treatment to meet LDRs at one centralized DOE site. It also lists data on waste loads from the previous study for comparison. In general, the waste load for each technology under each siting configuration is higher when calculated with the updated data than when calculated with the previous data. The differences range from a few percent to about 60%.

Table I

Table II shows the total CH TRU waste loads throughout the DOE Complex for storage and disposal. The highest calculated waste loads for the disposal facility at WIPP are as follows: 1) about 140,000 m3 for the decentralization of treatment to meet WIPP WAC, 2) 93,000 m3 for the regionalization of treatment to reduce gas generation, and 3) 63,000 m3 for the centralization of treatment to meet LDRs. The differences between these data and the earlier study's data range from 10% to 30%. The estimated waste loads do not include waste volume that could result from environmental restoration activities. WIPP's design capacity for CH-TRU waste is 170,000 m3.

Table II

SUMMARY

This paper provides information on the volume of TRU waste loads requiring treatment, storage, and disposal at DOE facilities for three

siting configurations. Input consisted of updated inventory and generation data from WIPP-BIR. Results indicate that WIPP's design capacity is sufficient for the CH TRU waste found throughout the DOE Complex. In a comparison of waste loads estimated in this study with those estimated in a previous study, differences, expressed as a percentage of the newly estimated waste loads, ranged from a few percent to about 60% for treatment of CH TRUwaste and from about 10% to 30% for its storage and disposal.

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CLEARANCE OF RADIOACTIVE WASTE: FRENCH AND BELGIAN APPROACHES IN RELATION TO INTERNATIONAL RECOMMENDATIONS

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ABSTRACT

During the last few years, important progress has been made at the international level with respect to the derivation of clearance levels for radioactive material. Although there is general agreement about the necessity to reach a broad consensus among the countries on this issue, it is important to note that these clearance levels must be supplemented by practical and industrial considerations, taking into account the prevailing situation in each country.

This paper focusses on two countries: France and Belgium in comparison with international recommendations.

For many years, France has been developing an important nuclear program to produce electricity. In this framework, radioactive waste management received almost immediate attention. A first low level waste surface disposal started operation in the "Manche" department, near La Hague, in 1969 and a second one in the "Aube" department, 200 km east of Paris four years ago. In 1991, French Parliament passed a law to define the way high level and long lived waste will be managed for the fifteen coming years. In 1994, French authorities decided to start studying, with the producers concerned, the way the very low-level waste, coming for example from the decommissioning of nuclear plants, could be disposed of.

The paper presents the framework of these studies and the first ideas of what could be dedicated disposal facilities.

As for Belgium, the legal role of ONDRAF, the national agency for the management of radioactive waste and enriched fissile materials, regarding clearance of radioactive material is explained. In accordance with the regulations, the objective of ONDRAF is to define practical rules for clearance of radioactive material in compliance with the general safety regulations. In order to reach that objective, a general strategy with well-defined steps is being implemented. To this end ONDRAF participated in advisory groups of the IAEA to recommend unconditional and conditional clearance levels, as well as in an EC working group on the clearance for recycling of metals. In addition, ONDRAF participated in two studies within the scope of a contract with the European Commission. The objective of these studies was to derive clearance levels for landfill disposal: the first study considered the case of an industrial landfill and the second one the case of a dedicated landfill for very low-level waste. The methodology and results of both studies are summarized and commented.

The present situation regarding clearance of radioactive waste takes into account the general Belgian regulations for radiological protection. To conclude, the Belgian and French approaches for clearance of radioactive material will be compared.

MANAGEMENT OF VERY LOW RADIOACTIVE WASTE IN FRANCE

The French approach to the management of very low-level radioactive waste should be placed in the larger context of waste in general, whether industrial, household or radioactive.

Waste in France

France generates approximately 600 million metric tons of waste per year. This is broken down into 400 million metric tons of agricultural waste, 102 million metric tons of inert industrial waste, 30 million metric tons of ordinary industrial waste, 15 million metric tons of special industrial waste, 3 million metric tons of toxic waste, 30 million tons of household waste, and 700,000 metric tons of hospital waste. Only 40,000 metric tons are radioactive waste.

Industrial and Household Waste

Ten years ago, France embarked upon a far-reaching program to improve non nuclear waste management and environmental protection. Regulations to reduce industrial and household waste volumes and to make their disposal sites safer have been set up in the past several years.

Three categories of landfills are distinguished. Class 3 landfills for inert industrial waste are not subject to regulation. Household waste and industrial garbage go to Class 2 landfills, for which a regulatory framework will be established by a ministerial order to be issued in the coming months. Special industrial waste goes to Class 1 landfills or burial sites with advanced safety systems, which are regulated according to the framework defined by ministerial orders dated December 18, 1992.

Nuclear Waste

France also developed a large-scale management program for radioactive waste commensurate with its nuclear power program, which generates 80 % of the country's electricity in 58 powerplants.

With the passage of the Waste Act on December 30, 1991, Andra, the National Radioactive Waste Management Agency, became a public service corporation with State oversight, independent of the CEA. The corporation's principal missions are:

- to design, construct and manage final disposal surface sites;

to register the location and condition of all radioactive waste on French territory; and

to study possible deep disposal facilities through a research program based on underground geological laboratories.

Radioactive waste is managed according to the specific activity and radioactive half-life of the waste. The following table indicates the disposal method to be used for each category of waste, based on the half-lives of the principal radionuclides present in the waste.

Table I

Short-lived low- and medium-level waste is disposed of at the Centre de l'Aube disposal site 200 kilometers east of Paris, near the city of Troyes. This site receives 20,000 m3 of waste per year; the total site capacity is 1,000,000 m3. The site was opened in 1992 and it should be able to receive low- and medium-level waste for the next fifty years.

Concerning high-level waste and long-lived medium-level waste and, in accordance with the Waste Act, Andra is charged with the creation of two underground laboratories designed to certify the suitability of the site to host a repository which provides adequate radioactive confinement and to verify the feasibility of the proposed repository design. This is to be demonstrated by the year 2006. In the event of a favorable outcome at that date, a repository could be operational in 2020. The Waste Act also requires the pursuit of two other programs in parallel: separation and transmutation of long-lived emitters in the waste, and enhanced conditioning and long-term storage.

Long-lived low-level waste will be disposed of in near surface disposal facility currently under design which should be operational by 2001.

Radioactive waste disposal sites, regardless of the activity of the waste (low-, medium- or high-level), are managed according to special regulations for licensed nuclear facilities, including nuclear reactors, reprocessing plants, and radioactive waste disposal facilities. These facilities are subject to special regulations based on a December 11, 1963 decree. The technical requirements are dictated by the Fundamental Safety Rules. The regulatory authority in charge of regulating these licensed nuclear facilities is the Department of Nuclear Facility Safety, known as the DSIN, which reports to the Ministry of Industry and to the Ministry of the Environment.

Very Low-level Waste

Our inventory would not be complete without mentioning very low-level waste. In principle, the June 20, 1966 decree constitutes regulations currently applicable to this type of waste. This decree considers materials whose activity levels are less than 100 Bq/g (or 500 Bq/g for solid natural materials) to be "non radioactive" and thus excludes them from the "radioactive" waste category. Although this exclusion should allow such waste to be disposed of as ordinary waste, this approach is not well received by the public. It is also recognized that the decree on radiation safety was not written to define categories of radioactive waste. For this reason, based on several reports commissioned by the Government in the last five years, restrictive practices have gradually been set up for waste whose activity levels are from 1 to 100 Bq/g (or 500 Bq/g in the case of solid natural materials). At the same time, monitoring gantries were installed at the entrances to disposal sites and scrap recycling sites. These gantries have detected a not insignificant number of radioactive anomalies, thus confirming the need to specify rules for the management of this type of waste.

The DSIN, together with ANDRA and the operators, followed the recommendations of the government reports by declaring that very low-level waste should not be treated as ordinary waste, even for activity levels of below 100 Bq/g, and should not be mixed with other waste. This waste should be disposed of at a special site with the same level of safety as the other nuclear facilities. The disposal site would be turned over to ANDRA, in accordance with the Waste Act, for long-term management and monitoring, i.e., approximately 100 years.

In the view of the DSIN, one fact is clear: government imposition of universal thresholds below which certain waste could be treated as ordinary, or "non-radioactive", waste is precluded for a variety of reasons:

- the public is not ready for the establishment of such universal "below regulatory concern" thresholds, and would perceive such a decision as self-serving and designed to allow the generators to get rid of their waste;
- thresholds could create an incentive to dilute the waste by mixing non radioactive waste with other, very low-level waste to bring its specific activity to below the regulatory threshold; and
- it is not presently feasible to reliably monitor large quantities of waste and on a large-scale when it leaves nuclear facilities or enters a disposal site, and this difficulty will only increase as the first light water reactors are dismantled, generating large volumes of concrete.

A management system should therefore be set up specifically for very low-level waste which distinguishes between the potential sources of such waste, whether the source be the "conventional nuclear sector" of the "miscellaneous nuclear sector". The first sector includes the nuclear power and defense industries; the second sector includes other industries, such as the chemical and metallurgical industries (phosphates, rare earths, etc.), which handle very low-level materials, sometimes without realizing it.

For the "conventional nuclear sector", the French regulatory authority looks for clear-cut, safe and responsible management of very low-level waste based on two principles:

- user responsibility for the fate of the materials used; and
- detailed identification of materials that may be contaminated by radioactive materials, from the initial contact to the moment they leave the site ("traceability").

In practical terms, adhering to these principles requires:

- the division of facilities into areas which are likely to generate radioactive waste and areas with no radioactivity;
- waste research;
- approved and monitored treatment processes;
- dedicated disposal sites managed by ANDRA; and
- more stringent regulatory control.

For the "miscellaneous nuclear sector" the principles should be the same. However, the number of players involved and their lack of knowledge about radioactivity make it necessary to check for radioactivity in their materials with a monitoring gantry at the entrance to disposal sites, steel plants, scrap recycles, or other facilities. If radioactivity is detected that is significantly higher than background radiation, the Regional Divisions of Industry, Research and the Environment could intervene and, based on an analysis of the specific cause, find the

material to be "radioactive waste". ANDRA would then take charge of the materials found to have very low activity levels.

BELGIAN SITUATION AND PROSPECTS FOR CLEARANCE OF RADIOACTIVE WASTE

Radioactive waste in Belgium is managed by ONDRAF-NIRAS, the national agency for radioactive waste and enriched fissile materials created by a law of March 1981 amended in October 1991. The competencies of ONDRAF include transport, conditioning, temporary storage and final disposal of radioactive waste, as well as the various related quality control and quality measures such as the definition of waste acceptance criteria and the qualification of waste treatment facilities.

As for the clearance of radioactive waste, ONDRAF-NIRAS was charged by law with the tasks of establishing, on the basis of general rules proposed to and approved by the competent authorities, detailed specifications and practices regarding clearance of radioactive waste, hereafter called clearance rules. The competent authorities are the Radiological Protection Office of the Federal Ministry of Public Health and Environment and the Office for the technical safety of nuclear installations of the Federal Ministry of Employment and Labor. In a very short future both offices will be replaced by a single federal agency for nuclear safety under the Ministry of Internal affairs.

In order to fulfill its task of establishing acceptance rules and criteria for clearance, ONDRAF has defined a global strategy which is detailed below.

Awaiting the definition of clearance rules and agreement on these rules, the Belgian regulations on radiological protection remain valid. These regulations stipulate that waste generated by licensed facilities must be treated as radioactive waste when its radiation exceeds the natural background level. It is one of the responsibilities of the health physics department of the facility to verify the radiation level. This means in practice that clearance of radioactive waste has been performed until now on a case by case basis and in a non coherent way.

General Strategy for Setting Clearance Rules

In accordance with the regulations, the objective of ONDRAF-NIRAS is to define practical rules for clearance of radioactive waste in compliance with the general safety regulations. These rules will be particularly important in the case of dismantling and decommissioning of nuclear facilities or after large repairs. These operations generate large quantities of low-level or slightly contaminated material which are suitable for processing to reach very low residual activity levels and for clearance. The clearance rules will have an impact on the development of decommissioning sequences and on the evaluation of their costs.

The clearance rules will include clearance levels but also control measures related to these clearance levels. The clearance levels will make it possible to define, for both unconditional and conditional clearance, acceptable levels for surface contamination and for the activity concentrations of groups of isotopes depending on their radiotoxicity. The derivation of unconditional clearance levels must necessarily take into account radiation exposure during all of the reasonably possible uses and movements of the materials intended for clearance. For a given radionuclide, the derived clearance level will be determined by the scenario and exposure pathway which give rise to the highest radiation dose. When a practice that is suitable for clearance is well defined, e.g. landfill disposal, it will usually be possible to take account of the known features of the practice. These considerations may

be expected, in general, to lead to higher clearance levels as compared to the unconditional levels. Furthermore, in case of conditional clearance, the clearance rules must also make it possible to define rules concerning the system and measures to be implemented in order to make sure that the cleared material follow the selected route.

The second aspect of the clearance rules concerns the control of the clearance levels. It is indeed important from a regulatory viewpoint, to be able to verify the applicable clearance levels. This can be done by direct measurement on the material to be cleared, by laboratory measurements on representative samples, by use of properly defined scaling factors or by other means accepted by the competent authorities. Of course, the choice of measurement strategy and appropriate instruments will depend on the type of material and radionuclides present.

ONDRAF-NIRAS has defined the following strategy to set the clearance rules:

The different actions must take into account the international state of the art and progress. It should indeed be kept in mind that materials cleared in one country are not distinguishable from non-radioactive material and may be moved from one country to another. It is therefore necessary to develop an approach and to derive clearance levels that are agreed upon internationally. To this end ONDRAF-NIRAS participated actively in advisory groups of the IAEA (International Atomic Energy Agency, Vienna) set up to derive and recommend unconditional and conditional clearance levels for solid materials containing radionuclides, as well as in a working group of the European Community, set up to derive clearance levels for recycling of metals.

On the basis of conservative nuclear safety rules, in particular for the individual dose criteria, clearance rules will be defined for the following practices:

- recycling or re-use in the nuclear sector;
- recycling or re-use in the non-nuclear sector;
- disposal in an industrial landfill or in a specific landfill intended only for cleared radioactive material (so-called dedicated landfill). The choice between the two types of landfill is first of all an economic choice strongly linked to the quantity of material likely to be cleared. For the derivation of clearance levels for landfill disposal, two specific studies were performed which are detailed hereafter.

In addition to these conditional clearance levels, levels for unconditional clearance will be proposed.

The clearance rules, including practical measures such as controls, will be presented and discussed with the main producers of radioactive waste. It should indeed be noted that one objective of the clearance rules is to ensure coherence of these practical measures between the various waste producers.

After reaching a consensus with the waste producers, the clearance rules will be presented to the competent authorities before becoming effective.

Clearance levels for Landfill Disposal

Within the framework of a contract with the EC (European Commission), ONDRAF participated in two studies to derive clearance levels for landfill disposal: the first concerns the case of an industrial landfill and the second the case of a dedicated landfill for very low radioactive waste.

a. Clearance levels for industrial landfill disposal

Following a conservative approach, the following masses of very low-level radioactive waste were assumed: 100,000 ton averaged over a period of 30 years sent to one class 1 landfill (industrial waste) or ten different class 3 landfills (inert waste).
As a landfill disposal site accepts waste from different origins, it was assumed that the very low-level waste is mixed with ordinary industrial waste.
Different exposures were selected: engine driver, foreman, truck driver, general public.
In addition, the water pathway was studied on the basis of data for a specific class 1 landfill.
The results and a comparison with the (IAEA) unconditional clearance levels are given in Table I.
The table shows that the derived levels vary between several orders of magnitude:

- from 1 to 800,000 Bq/g for the class 1 landfill;
- from 1 to 2,000,000 Bq/g for the class 3 landfill;

They reflect the highly different radiological characteristics of the radionuclides. The lowest values are found for the radionuclides with high gamma energy and for the alpha emitters.
The water pathway was found not to be limiting.
The dose to the truck driver appears to be the limiting scenario for many radionuclides (for this scenario no mixing with other non-radioactive waste is assumed).

b. Radiological impact of a special landfill site for very low radioactive waste

This study was aimed at evaluating the exposure, in terms of doses, resulting from the operation (for workers) of a special landfill (with hypothetical geology and technical characteristics) filled exclusively with very low radioactive waste.
As in the first study, the impact of the site on the public during the post-closure phase was taken into consideration.
The activity of the waste likely to be sent to the facility ranges from 1 to 100 Bq/g.
In the absence of actual data on the isotopic composition of the waste involved (as for the first study: 100,000 tons over 30 years), the radiological impact was based on a specific uniform activity of 100 Bq/g.
For the workers, the highest doses are logically obtained for the radionuclides with high gamma energy and for the alpha emitters, but remain limited to values around 0.01 mSv/year.
For the individuals of the general public, doses around 0.1 mSv/year are obtained for the alpha emitters, and for some radionuclides such as Nb-94, Ag-108 m and Ra-226 doses above 1 mSv/year are found.
The water pathway generates the highest dose for I-129 (around 0.1 mSv/year); the other doses remain below 0.02 mSv/year.

CONCLUSION

Both in France and in Belgium, management of very low-level waste fits within the scope of the general policy on radioactive waste management pursued by the respective national organizations, i.e. ANDRA for France and ONDRAF for Belgium. The management strategy outlined by these organizations shows nevertheless different levels of progress in the two countries.
In France, a strategy was outlined in consultation with the safety authorities for very low-level waste, i.e. waste with an activity below

100 Bq/g produced by the "conventional nuclear" sector. This strategy aims at disposing of the waste in a "dedicated" disposal site after having identified the areas of installations that are likely to generate radioactive waste.
In Belgium, the strategy outlined by ONDRAF aims at defining exemption levels and rules for different practices according to the nature of the waste and its destination. One of the practices envisaged corresponds with that foreseen in France. However, this solution of "dedicated" disposal can only be envisaged, both technically and economically, for waste quantities. It is also not excluded that, for some type of very low active waste, a so called unconditional clearance, based on internationally agreed levels, will be possible in Belgium after agreement of the competent authorities.
The different progress in the implementation of a strategy for very low-level waste in the two countries can be explained, among others, by the different order of priorities due to the existing quantities of waste suitable for disposal.
It should be noted that both organizations agree on the fact that one of the main difficulties associated with the management of this category of waste is caused by measurement problems (levels, representativity of samples) regarding low activity levels.

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HIGH ACTIVITY WASTE DISPOSAL
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ABSTRACT
OHM Remediation Services Corp., Nuclear Services Division (OHM) has developed a container that is capable of holding high activity waste and can be shipped as a DOT Type A shipment. By making the container special form the amount of activity that can be transported in a Type A shipment is greatly enhanced. This leads to a great savings in the transportation costs associated with cask rental, handling and shipping. Normal form packages can allow access to the radioactive material without destruction of the package. Special form material presents an extra degree of protection to the environment by requiring the package to be destroyed to get access to the radioactive material. The package must undergo specific testing requirements to meet these criteria (10 CFR 20 and 49 CFR 173). Typically, sealed sources do not meet the requirements for normal form. For example, Radium 226 is normally limited to 50 milli-curies per container for transportation as DOT Type A. With the special form container up to 10 curies of Ra 226 can be transported in a single package.
DESCRIPTION OF WORK
OHM was performing a superfund remediation job for the EPA in Queens, New York which involved the removal of approximately 120 curies of radium needles, plaques and sources, plus various amounts of highly contaminated material. The problem arose as to how to dispose of the sources in an efficient manner to minimize radiation exposure, without escalating

costs. The following items were taken into consideration that lead to the development of the special form disposal container:

Prevent having to handle each of the approximately 10,000 sources,
Limit whole body and extremity exposure,
Not having to determine if they leaked or not, and
Prevent an excessive number of shipments of small activity,

As mentioned before, with radium the packaging efficiencies greatly increase with the special form container. Other radionuclides where this container offers a distinct advantage are given in Table I.

Table I

Through review of all of the restricting criteria from the disposal site and the Department of Transportation, a plan of action was made to develop several operational containers capable of meeting the above needs. Criteria considered included:

A. Exposure limits on the package - 1000 mR/hr.
B. Exposure limits on the surface of the vehicle -200 mR/hr.
C. Exposure limit of 10 mR/hr at 2 meters from the vehicle.
D. No leaking sources are acceptable for disposal (over 10,000 sources had to be determined to be either leaking or non leaking).
E. Sources required to be inside a DOT 2-R container.
F. The disposal site can have no more than 25 curies above ground at one time.
G. Weight constraints on the 55 gallon drum (1400 lbs).

Immediately, the concern turned to protection of the workers loading and handling the container. Through calculations and shielding analysis, the exposure rates for various amounts of radium were approximated on the outside of the container. Appropriate shielding was then designed to protect the workers loading the container, welding and inspecting the container, and then handling the container for shipment. Calculations showed unshielded radiation levels on the outside of the container to be approximately 15 R/hr on contact on the side, 21 R/hr on contact on the bottom, and 800 mR/hr at one meter from the side of the container.

The use of lead shielding was obviously necessary to allow the workers to stay within the administrative dose guidelines of OHM Nuclear to meet its ALARA philosophy. Calculations were then done to determine the maximum amount of lead that could be placed in a container and still meet the weight constraints of our 55 gallon drum and the testing criteria of 49 CFR 173.

INNER CONTAINER

The size of the inner, special form, container was determined by considering what type of material would be placed inside. Radium sources can be used for pharyngeal applications that require needles up to eight inches in length, with the diameter being about one eighth of an inch. Other longer needles were assumed to be possible, but the probability was determined to be low and it was decided they could be remotely handled and made to fit the container. The potential for extremely high dose rates that are associated with 10 curies of radium 226 (or 30 curies of cesium 137) warranted special shielding considerations. With the burial site criteria of radium being required to be inside a DOT 2-R container the diameter of the encapsulating container and additional lead shielding was limited to 12" by the DOT specifications for the 2-R container. Therefore, the next consideration was a compromise between the physical sizes of the inner, special form, container and the lead shielding that would reduce the exposure rates the most for this application. Because of

the constraints of welding and testing the weld on the special form container extra attention was given to the exposure levels at the top of the inner container. The dilemma here arose from the fact that the top had to be welded, and then the weld inspected, both tasks requiring close hands-on work in a high radiation field. See Fig. 1. Steel was used as shielding material on the sides of the container, while lead was used at the top. To meet special form criteria the outer sides, top, bottom, and lifting lugs were all stainless steel. All welds were inspected non-intrusively prior to final release.

Fig. 1

ENCAPSULATED CONTAINER

The special form container was required to be placed inside a DOT 2-R container, which limited the physical size to a 12 inch diameter. Additional shielding was required to maximize the activity for disposal in the inner container. With the evaluation of the weight limitations, the size dimensions, the shielding needs, and possible materials, it was determined that one inch of lead was the maximum that could be placed along the sides of the encapsulating container. The bottom would require more shielding because of its close proximity to the sources and could have one and one half inches of lead shielding.

Fig. 2

It was determined that shielding inside the special form container was required to meet the needed constraints, mainly due to ALARA considerations. Testing of the special form container proved difficult because of the weight involved with a container of such size. The walls of the container were made from one quarter inch stainless steel. Therefore, the welding to seal the bolt of the container had to be a Tungsten-Inert Gas (TIG) weld. This type of weld is difficult and was made even harder by being in a difficult place to perform. As can be seen it would be inside 55 gallon drum, inside a shielded 2-R container.

In the development of the container several designs were proposed and tested. Physical constraints that were imposed by the mere numbers of sources to be handled and the uncertainty of the dimensions of the items to be disposed required shielding analysis of all proposed containers as part of the design phase. Changes that were the result of actions at the superfund site required an additional container be constructed with a wide mouth opening. This container was capable of holding items too large to fit inside the small opening of the first container. The larger container was also designed to completely enclose the smaller container if the need arose.

CONCLUSION

The containers designed were capable of meeting the design criteria as imposed by DOT and the burial site and were used extensively throughout this project and at other projects. The dose received by individuals handling the large amounts of radium fell well within the ALARA concept. It was shown that high activity waste could be handled safely and efficiently in large quantities without exposing the worker to undue risk. In one instance over two curies of radium was packaged and shipped for disposal with the exposure for all workers involved totaling 0.49mSv (49 mrem).

GETTING STARTED AND MAKING PROGRESS

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ABSTRACT

This paper reports on the current efforts of the Office of Spent Fuel Management (EM-67) to meet the major challenge facing the U.S. Department of Energy-owned spent fuel Quality Assurance community: the implementation of an effective, efficient, and integrated Quality Assurance program that meets all regulatory requirements of the Office of Civilian Radioactive Waste Management's Quality Assurance Requirements and Description (DOE/RW-0333P). The paper presents the goals and status of the implementation of the Department's Spent Fuel Quality Assurance Program. It summarizes progress made in assessing and implementing Quality Assurance requirements for Department-owned spent fuel, the methodology for verifying compliance, where appropriate, with DOE/RW-0333P, and implementing processes and procedures to qualify Quality Assurance programs and verify quality affecting data. Lastly, it discusses the vision forward initiatives for the Spent Fuel Quality Assurance Program.

INTRODUCTION

The Department of Energy (DOE) confronts the difficult task of managing and disposing of the inventory of DOE-owned spent nuclear fuel (SNF) generated by past production and ongoing test/research reactor operations. In 1992, the Secretary of Energy directed the Assistant Secretary for the Office of Environmental Management (EM) to develop an integrated, long-term SNF management program to consolidate, under EM management, all DOE SNF and associated facilities not addressed by the Office of Civilian Radioactive Waste Management (RW), which manages the program for the safe disposal of SNF produced by civilian nuclear power reactors. Currently, EM is responsible for the DOE-owned SNF Program's policy, while overall coordination is assigned to the Office of Spent Fuel Management (EM-67). EM-67 is therefore responsible for mitigating vulnerabilities and implementing the safe and cost effective near-term interim storage of DOE-owned SNF prior to the final disposition of the fuel at an RW-managed storage facility or repository.

An integral component of this responsibility is the implementation of quality assurance (QA) programs for the management of DOE-owned SNF (1). QA includes all those planned and systematic activities necessary to provide confidence that a structure, system, or component will perform satisfactorily in service. Commonly required actions in a QA Program include well-defined responsibilities and authority, controlled and documented personnel qualifications, and documented evaluation activities. QA assists in ensuring that the SNF Program is managed in a coordinated manner consistent with relevant technical requirements. A systems engineering process was introduced to the Program to establish a consistent complex-wide standard of excellence which fosters continuous improvement and enhancement of SNF activities. This process was intended to facilitate the application of appropriate QA practices and applicable standards integrated into the SNF management system of planning program,

establishing requirements, integrating activities, and providing resources throughout the life cycle of the SNF Program, from assuring existing storage to preparing for final disposition.

A key element of the SNF QA Program is its linkage with the RW QA Program. A primary element of RW's mission is to site, construct, and operate a geologic repository for the disposition of commercial nuclear power plant spent fuel and high-level waste (HLW). Since the Department of Energy has determined that the path forward for final disposition of its spent fuel is in the geologic repository, EM-67 has adopted RW's Quality Assurance Requirements and Description (DOE/RW-0333P, hereinafter referred to as the QARD) as the baseline for all quality affecting activities associated with preparation for disposal (2).

Because DOE-owned SNF may eventually be turned over to RW, it must be characterized and processed so as to meet the long-term Nuclear Regulatory Commission (NRC) requirements for long-term disposal and thereby be acceptable to RW at the time of turnover (3,4). All producers of HLW and owners of SNF are required to comply with these QA requirements to assure the acceptability of their HLW and SNF by RW for disposal in a geologic repository. The QARD, the principal QA document for RW, establishes the requirements for the HLW/SNF QA program. It is a compliance-based document designed to meet the NRC licensing requirements. Since the QARD has been reviewed and approved by the NRC, its adoption will facilitate the movement of the SNF Program toward NRC-licensed (or licensable) interim storage facilities. Additionally, EM adoption of the QARD will align the SNF QA programs with the RW QA program and facilitate ultimate disposal. Implementation of the QARD involves EM-30, EM-60, DOE Operations/Area Offices, and its management and operating (M&O) contractors (5).

The major challenge facing the DOE-owned SNF QA community is the implementation of an effective, integrated, and efficient QA program that meets all regulatory requirements of the QARD. The goal of the DOE-owned SNF QA Program is to obtain RW acceptance of the National SNF QA Program and begin qualification of SNF sites to support the SNF Program's path forward for final disposition. The systems engineering breakdown provides the road map to implement the QARD and other applicable standards.

BACKGROUND

The SNF Program is currently incorporating the RW QA requirements to SNF quality-affecting activities. Since EM-30 initially established a QA program for HLW vitrification activities that addressed all RW QARD requirements, the Deputy Assistant Secretary for Waste Management (DASWM) concluded that the HLW QA program should be extended to include SNF. Because the HLW QA Program was accepted by RW on August 13, 1994, the Office of Program Integration (EM-33) and the Office of Spent Fuel Management (EM-37) signed an agreement in principle (AIP) and a proposed implementation process to designate the HLW/SNF QAPM (EM-33) as the single point of contact between EM-30 and RW for QA matters, and to develop one set of SPPs for SNF and HLW for implementation of the requirements in the QARD (6). In August 1994, a working group of EM-33 and EM-37 HQ and site personnel was established to integrate applicable SNF QA activities into the HLW SPPs. The SPPs describe internal and external interfaces, organizational structures, requirements and responsibilities for the HLW and SNF quality affecting work at Headquarters. Compliance with the SPPs is mandatory for all HQ

organizations (including HQ support offices) involved in the HLW/SNF QA Program.

The HLW/SNF SPPs were issued in February 1995. In addition, the HLW/SNF Quality Management Plan (QMP) was developed as an appendix to the HLW/SNF SPPs. The QMP describes the HLW/SNF QA Program and the SPP process for controlling HQ and HQ support offices. The QMP provides the overall basis for an effective QA program to address both the HLW and SNF functions and corresponds to the criteria of the QARD. It identifies the applicability and responsibilities for the appropriate programmatic requirements for HLW/SNF QA activities, including a general description of the QA program hierarchy, identifying those elements applicable to HQ and guidance information at the Operations Offices and M&O contractor levels. The QMP is based on the requirements of EM-30, "Quality Assurance Program Description" (QAPD), and the QARD. Also under this AIP, EM is developing for submission to RW an SNF requirements matrix that identifies the SPPs to implement the QARD requirements.

In December 1995, the Office of Spent Fuel Management (EM-37) was realigned under the Office of Nuclear Material and Facility Stabilization (EM-60). This realignment/reorganization shifted program management functions within the following EM Offices:

The Office of Spent Fuel Management (formerly EM-37) has become EM-67, interfacing with other EM-60 offices (EM-63 and EM-65) on SNF site issues;

HLW/SNF QAPM (formerly within EM-33) is now in EM-37 (Office of Technical Services) and is still charged with overall management of HLW/SNF QA Program;

QAPM is still the lead point-of-contact to date between RW and EM for QA activities related to both HLW and SNF; and

QAPM is responsible for submitting QA documents to RW for acceptance, i.e., organizational descriptions and QARD Requirements Matrices, as required by RW QARD).

Figure 1 displays the document hierarchy chart for the SNF Program.

Fig 1

RW ACCEPTANCE OF THE HEADQUARTERS SNF QA PROGRAM

The QA Program being applied to the DOE-owned SNF activities is described in the HLW/SNF and the SPPs. The SNF Program has established the QARD as the standard for the following applications:

Characterization or data collection for input or use for interim storage or ultimate disposition that could affect the acceptance of SNF in a Nuclear Regulatory Commission (NRC) licensed storage facility or repository.

Conditioning for interim storage or into final form for disposal that could affect the acceptance of SNF in an NRC licensed storage facility or repository.

Handling and packaging for interim storage or disposal that could affect the acceptance of SNF in an NRC licensed storage facility or repository.

The implementation of the QARD for these activities will facilitate the interfaces between EM, RW, and NRC and assure the smooth transition of the SNF from existing conditions through interim storage to final disposition.

The appropriate criteria from other quality standards will be utilized as they apply to the remainder of SNF activities as defined in the individual site quality program. If the decision is made to privatize the

operations of new interim storage facilities or packaging and transportation of SNF, the NRC regulations will apply through the direct application of Title 10 of the Code of Federal Regulations (CFR), Part 72 or Part 71, respectively, assuming privatized storage facilities will be licensed by and appropriate fees paid to the NRC.

The Headquarters SNF QA Program is taking a number of actions to obtain RW acceptance by March 29, 1996. Figure 2 displays the implementation of the SNF QA Program. Actions or steps taken or to be taken include:

Coordinating with RW in revising the QARD to include an SNF QA Appendix.

Revising the QMP, which defines the scope of application to those SNF Program management activities performed within EM-60 organizations involved with the SNF Program. The QMP identifies the specific authority and responsibility for SNF quality-related activities, and also defines roles, responsibilities, and activities of the EM-30 organization assigned QA responsibilities for the SNF Program.

Revising the SPPs to show compliance with recent revision of the QARD. Issuing the SPPs to SNF personnel. Providing SPP training to all SNF personnel who implement SPPs.

Performing internal reviews of implementation

Submitting RW requirements matrix and organization procedure to RW for acceptance

Performing evaluation and assessment activities of implementation both at HQ and in the field

Obtaining RW-3 acceptance of the SNF QA Program

Fig 2

SNF QA PROGRAMS AT THE SITES

Documents from a variety of sources are used to develop the quality program, quality assurance plans, and procedures to control SNF work activities. However, the contents of the source documents do not necessarily apply to all program activities. The applicability of quality program regulations, DOE Orders and directives, national standards, and guidance documents to the SNF program needs to be evaluated on a site-by-site basis in accordance with the following guidance. For each specific site, the quality assurance programs for SNF management are still under development. Exceptions and/or changes to each site's quality assurance program may be needed to demonstrate implementation of the QARD.

In response to DOE Order 5700.6C, Quality Assurance, and 10 CFR 830.120, Quality Assurance Requirements, each site is developing or has developed a QAPD to cover their site operations. These QAPDs establish the foundation for operation of each site's quality assurance program. The QAPD applies to organizations, individuals, vendors, and other entities working for the site. 10 CFR 830.120 provides requirements for M&O contractors in the design, construction, and operation of nuclear facilities, while nonnuclear facilities are governed by DOE Order 5700.6C. The 5700.6C requirements are applicable to site activities not related to SNF activities.The SNF Program is undertaking actions to qualify the SNF QA Programs at Idaho, Richland, and Savannah River by FY 1996 and FY1997. (See Fig. 3). Actions or steps include:

Directing (via the EM Program Execution Guidance for FY 1996) RW QARD implementation for ongoing SNF activities, managing SNF data and records.

Initiating a consolidated QA activities report which provides status of QA programs at HQ and at applicable SNF Sites.

Establishing lines of communication and identifying SNF points-of-contact.

Providing systems engineering integration guidelines.

Conducting QA and data reviews at the sites. Providing guidance based on review results.

Establishing SNF data proposed path forward based on reviews.

Conducting qualification audit and activities of the Operations Offices and its M&O contractors to verify QA program implementation in accordance with the QARD.

Fig 3

Fig 4

DATA QUALIFICATION PROCESS

QA records are completed documents that furnish evidence of the quality of items and activities. SNF QA records are generated when SNF is handled, moved, characterized, or conditioned. QA records also are generated during the design, specification, and qualification of facilities or equipment involved with SNF disposal. SNF QA records must be controlled to ensure that information is accurate and available for use in support of the eventual disposition of SNF in a disposal facility. Accordingly, each SNF site will implement a records management system specifically for the DOE SNF Program.

Records generated prior to development of the DOE SNF Program also may be needed to support interim storage and ultimate disposition of SNF. Until RW identifies all SNF information needs, sites will ensure that previously generated records are protected from damage, loss, or destruction. When these records are used to support interim storage or ultimate disposition of SNF, they will be validated as QA records and managed under each site's QA records program.

The Integrated Spent Nuclear Fuel Database System (ISNFDS) was developed as a management tool specifically for EM-67 as the single source of data for management planning during disposition of DOE's SNF. The ISNFDS now contains information on most DOE-owned SNF. Data in the ISNFDS needed to support acceptance of SNF in a geologic repository will be qualified to requirements of the QARD by the SNF sites. The ISNFDS will be updated with qualified data which will then be made available to the Unified Database under development by RW.

VISION FORWARD: SNF QA INITIATIVES

The QMP will formulate national Program QA management and oversight strategies and define the site-specific QA implementation strategies necessary to accomplishing the SNF Program mission. All site-specific SNF activities will be categorized using the SNF systems engineering process to identify the applicable QA implementation standards. The SNF Program has selected the RW QARD for SNF activities affecting interim storage or final disposition of SNF, from assuring existing storage through achieving interim storage to preparing for final disposition. QA controls will be applied in a graded approach consistent with, and as defined by, the applicable implementation standards.

Implementation of the RW QARD must be performed at the Sites to cover ongoing quality activities, such as the TRIGA cask loading at DOE Richland, the lab studies at DOE Savannah River, and the canning and conditioning at INEL as fuels are moved from wet to dry storage.

The SNF QA Program has projected a number of activities and initiatives for Fiscal Years 1996 and 1997. However, these plans may require

significant modification and updating in light of the uncertainties associated with various reductions in both EM and RW budgets.

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APPLICATION OF QUALITY ASSURANCE CONTROLS TO TBM TUNNELING ON THE YUCCA MOUNTAIN PROJECT

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ABSTRACT

As part of the Yucca Mountain Project (YMP), a 7.62-meter diameter tunnel is being constructed using a Tunnel Boring Machine (TBM). This tunnel, which may form a portion of a permanent high-level nuclear waste repository, is being constructed under the auspices of a nuclear quality assurance (QA) program. The YMP nuclear QA program applies to items and activities determined to be important to radiological safety, waste isolation, and potential interactions with the environment. The items and activities determined to be important have been assigned a quality assurance classification. This paper focuses on the items (rockbolts, steel sets, and shotcrete) and quality affecting activities involved in providing ground support and excavating the tunnel. Typical activities that have been assigned QA classifications include TBM maintenance, control of water used in the tunnel during construction, and control of diesel emissions in the tunnel. The paper concludes that the key to the successful implementation of nuclear QA requirements for tunneling at Yucca Mountain was the assignment of personnel with the appropriate mix of tunneling and nuclear experience.

OVERVIEW OF THE CONSTRUCTION PROJECT

The U. S. Department of Energy's Yucca Mountain Project (YMP) is being conducted to determine the suitability of Yucca Mountain, located northwest of Las Vegas, Nevada on the Nevada Test Site, as a high-level nuclear waste repository. As part of this project, an Exploratory Studies Facility (ESF) is being constructed to characterize the site and to form a portion of the permanent repository should the site be found acceptable. Construction of a 7.62-meter diameter tunnel and associated alcoves approximately 7.8 km in length is a major portion of the ESF. Kiewit/PB is responsible for the construction of the tunnel using a

design provided by the Management & Operating (M&O) Contractor for the Yucca Mountain Project, TRW Environmental Safety Systems Inc.

The tunnel is being excavated using a Tunnel Boring Machine (TBM). Ground support, in the form of rockbolts or steel sets (structural steel members), is installed as the TBM advances. Steel reinforced concrete invert segments and rail track are also placed as the TBM advances and provide the surface and track which support the TBM trailing gear. As can be seen from Fig. 1, the TBM trailing gear includes a mapping gantry used by scientists assigned to the project to conduct detailed geologic mapping of the ESF. Use of this mapping gantry allows the geological mapping to be conducted at essentially the same time as the tunnel is bored and prior to the installation of support utilities such as air and water lines, electrical cable, and the muck conveyor.

Fig. 1

As of December 1, 1995, close to 3.2 km of the approximately 7.8 km in the total tunnel loop had been constructed. It is planned that the tunnel be finished in calendar year 1996 or early 1997. The possibility exists that additional tunneling beyond that of the main loop may eventually be constructed.

QUALITY ASSURANCE REQUIREMENTS

ESF construction work is conducted by Kiewit/PB in accordance with specifications and drawings provided by the M&O Architect Engineer (A/E) organization. These specifications and drawings identify items and activities which have received a QA classification referred to as a 'Q' classification and fall under the auspices of a QA program required to be implemented by Kiewit/PB. 'Q' classifications include items and activities important to radiological safety, important to waste isolation, and important to potential interaction with the environment. Importance to radiological safety involves ensuring that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. Because the ESF tunnel may become part of the permanent repository, this classification is applied to certain items providing ground support. Importance to waste isolation involves ensuring that barriers to prevent the migration of radionuclides remain intact during storage. Use of organic material which could degrade such barriers over time is limited and strictly controlled as a result of the application of this classification. Items whose failure could impair the ability of another item to perform its intended radiological safety or waste isolation function are classified important to potential interaction and also fall under the auspices of the QA program.

The overall QA program that is required to be applied to 'Q' classified items and activities for work at Yucca Mountain is defined in the U.S. Department of Energy Office of Civilian Radioactive Waste Management Quality Assurance Requirements and Description (QARD). This document is a compilation of the various regulatory and commitment documents regarding QA that are applicable to the YMP but is based primarily upon the following documents:

10CFR60, Disposal of High-Level Radioactive Wastes in Geologic Repositories, Subpart G - Quality Assurance

10CFR50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

ANSI/ASME NQA-1(1989), Quality Assurance Program Requirements for Nuclear Facilities

NRC Review Plan (Rev. 2), U.S. Nuclear Regulatory Commission Review Plan for High-Level Waste Repository Quality Assurance Program Descriptions

Kiewit/PB has established Management Control Procedures which address and implement the QA requirements contained in the QARD. In addition, Technical Control Procedures have been written that define how quality-related work will be conducted by the construction forces and engineering support organizations. Quality Control Procedures have also been written that define how quality control inspections will be conducted and documented.

The overall Kiewit/PB organization includes a QA organization that reports on a day-to-day coordination basis to the Kiewit/PB Project Manager and to a corporate Quality Services Director on QA issues. The procedural and organizational structures of Kiewit/PB as they relate to QA requirements are similar as would be found for most organizations implementing nuclear QA requirements. Of greater interest as reported herein is how the nuclear QA requirements are applied to the construction of a tunnel and this is discussed in some detail in the following section.

APPLICATION OF QUALITY ASSURANCE REQUIREMENTS

The only hardware items that have received a 'Q' classification are those items providing ground support for the tunnel. These items are rockbolts, steel sets, and shotcrete. The use of shotcrete is minimized because of its impact on the geological mapping and evaluations being conducted by the scientific community. No quality-related shotcrete has yet been placed in the main portions of the tunnel and alcoves. A description of the application of quality assurance requirements to the procurement and installation of rockbolts and steel sets follows.

Rockbolts

Two types of permanent function rockbolts are currently being utilized for 'Q' applications. The first is Super Swellex which are manufactured by Atlas Copco and the second is a hollow core continuously threaded rockbolt provide by Williams Form Engineering Corporation. These two types of rockbolts are shown in Fig. 2 along with the required installation pattern for both.

Fig. 2

The Super Swellex bolts, three meters in length, are circular steel tubes, folded to reduce their diameter. They are placed in drilled holes and expanded using high-pressure water. The bolts expand and may compact weaker material surrounding the hole and adapt their shape to any irregularities in the hole. The resulting frictional and mechanical interlocking reinforce and increase the stability of the rock surrounding the drilled hole. It should be noted that rolled steel channels and wire mesh may be installed along with these rockbolts as necessary for personnel safety.Because the manufacturer of Super Swellex bolts does not have a nuclear QA program meeting QARD requirements, the rockbolts are purchased as 'commercial grade' off-the-shelf items. Upon receipt, the rockbolts are 'dedicated' for use in a QA-classified application by the conduct of special tests and inspections. These special tests and inspections are conducted by the Kiewit/PB Quality Control group upon receipt of the rockbolts at the site and must be successfully completed before the rockbolts are released by the Quality Control group for use. Specifically, Quality Control verifies that the rockbolts have not been damaged or contaminated and that the dimensions specified by the

manufacturer are correct. One bolt out of every 100 received is destructively pull tested to verify that it meets the manufacturer's published minimum breaking strength of 200 kN.

During installation, Quality Control Inspectors monitor that the holes drilled for the rockbolts are clean and free of loose debris and are located in accordance with design pattern requirements. In addition, Quality Control Inspectors monitor the installation of the rockbolts to ensure that they are being properly pressurized to a pressure between 290 and 310 bar using calibrated pressure gauges. This ensures that the rockbolts are being properly expanded during the installation process. After installation, five of every 100 rockbolts are nondestructively proof load-tested to verify that a specified load can be placed without exceeding a specified displacement of the rockbolt. Although the tests are performed by the construction force, the tests are witnessed and the results verified by Quality Control Inspectors.

The Williams hollow core continuously threaded rockbolts are initially anchored in place by the mechanical anchor head at the end of the bolt placed into the drilled hole. As a nut on the outside end of the bolt is tightened against a plate on the tunnel surface, the mechanical anchor expands against the sides of the hole and the bolt is set in place. The bolt is then grouted in place by pumping grout through the hole in the center of the bolt until the space in the hole surrounding the bolt is completely filled. As with the Super Swellex bolts, the supplier does not have a nuclear quality assurance program and they are purchased 'commercial grade' and dedicated by the Kiewit/PB Quality Control group by conducting special tests and inspections upon receipt at the site. The bolts are inspected and one rockbolt out of every 100 is destructively pull-tested in a manner similar to Super Swellex rockbolts.

During installation of the Williams rockbolts, Quality Control Inspectors monitor to ensure that the bolts are being installed in accordance with design pattern requirements, proper hole depths for the rockbolts are achieved, there is sufficient protrusion of the rockbolts from the holes to ensure proper nut engagement, and that the proper setting of the rockbolts is occurring by checking the setting torque using calibrated torque wrenches. Steel channels and wire mesh are installed along with these rockbolts as necessary for personnel safety. The Inspectors also verify that the grout is properly mixed, at the correct temperature, and pumped until there is a continuous grout return visible at the rockbolt bearing plate. In addition, Quality Control takes sample grout cubes during each shift of grouting operations, cures the sample cubes in accordance with specification requirements, and conducts break tests in a site laboratory operated by Kiewit/PB Quality Control to verify that the required compressive strength of 20.69 MPa is being achieved.After the grout has set for a minimum of 72 hours, nondestructive pull testing of five of every 100 rockbolts is conducted and the test results verified by Kiewit/PB Quality Control.

A recent specification change, currently under evaluation for impact, has reclassified steel channel, wire mesh, and short rockbolts, as 'Q.' These items, previously unclassified and only installed by Kiewit/PB as necessary for personnel safety, will now be required to be receipt inspected and dedicated, installed in accordance with work procedures, and have the installation monitored for adequacy by the Kiewit/PB Quality Control group.

Steel Sets

Steel sets are used as tunnel ground support in areas where the ground conditions are inadequate to allow the use of rockbolts. Steel sets are structural steel rings, made from W8 steel shapes bent or rolled to the proper diameter, set on the concrete invert segments, and expanded into place on the inside diameter of the tunnel surface as shown in Fig. 3.
Fig. 3
The rings are normally placed four feet apart (two-foot spacing is allowed) and lagging, in the form of steel channel or wire mesh, is placed between the rings as necessary to support the ground. As shown in Fig. 4, the rings consist of three main segments, bolted together, and four shorter segments used in conjunction with shims during the installation expansion process. Hydraulic jacks are used to expand the rings against the tunnel surface, shims added as necessary, and the expansion segments bolted together.
Fig. 4
Although the initial steel sets installed in the tunnel were manufactured by a supplier working under the Kiewit/PB QA program, most steel sets have been procured from fully qualified suppliers having QA programs meeting nuclear requirements. The Kiewit/PB QA organization conducts surveillance of these suppliers during the manufacturing process to ensure specification requirements are being met. Upon receipt of the steel sets at site, they are inspected by Kiewit/PB Quality Control for damage and spot-checked for correct dimensions, configuration, and adequate welding. Documentation supplied by the manufacturer, including material test reports, welder qualifications, nondestructive test results, inspection results, and nonconformances, is also reviewed by Kiewit/PB Quality Control when the steel sets are received.
During installation, Quality Control Inspectors ensure that steel set components are not damaged; correct components such as nuts, bolts, and washers are used; the jacking force used to expand the rings does not exceed the maximum specified value of 242 kN; the spacing between the steel set rings is correct and that the rings are not placed closer than four inches from the front or rear edges of the concrete invert segment; and verify that the bolts connecting the steel set segment pieces are properly tightened. The Inspectors also record the unique identification numbers stamped on each of the ring segment pieces. In addition, the installation of the lagging is monitored to ensure that it is properly installed and is not damaged.
Other construction activities, including shotcreting, drill and blast excavation of alcoves, and fabrication and installation of concrete invert segments, require Quality Control monitoring and inspection. Specifically, the Quality Control group performs civil testing related to these activities including aggregate gradation, moisture, slump, and concrete cylinder and shotcrete core strength tests as well as other specific installation inspections.
QUALITY ASSURANCE CONTROLS RELATED TO OTHER SITE ACTIVITIES
Numerous other QA Controls are detailed in the specifications that govern the overall construction activities being performed by Kiewit/PB. These activities do not require direct inspection or monitoring by the Quality Control group but must be performed by trained personnel working to approved procedures. In most cases, quality assurance records are generated as a result of these controls. The function of the Kiewit/PB QA organization, as it applies to these QA Controls, is to ensure that the required work procedures are developed, the procedures adequately address

the applicable specification requirements, and that the requirements of the procedures are being properly implemented by the applicable Kiewit/PB organization responsible for conducting the work. This is accomplished by conducting reviews of all procedures that address QA Controls and by conducting surveillance of the implementation of these procedures. In addition, the overall Kiewit/PB QA program is periodically audited by the Department of Energy to verify that this program is being effectively implemented. Examples of typical QA Controls follow.

The specifications contain a QA Control requiring that a procedure be developed to monitor tunnel alignment to ensure that the tolerances of the excavated opening are within one-half tunnel diameter (3.81 meters) of the line and grade shown on the A/E drawings. This procedure has been developed and implemented by the Kiewit/PB Construction Department Survey Group. The Kiewit/PB QA organization conducts periodic surveillances to ensure that the requirements of this procedure are being correctly implemented.

The TBM specifications state "Perform periodic maintenance based on manufacturer's recommended maintenance/surveillance frequencies, including, for example: cutter inspection, leak inspection, oil sampling, lubrication point inspection, belt tensioning, belt inspection, scrubber cleaning, parts replacement, belt placement, etc." The purpose of this QA Control is, by requiring periodic maintenance, to eliminate or mitigate leaks or other failures of the TBM that could occur as the result of equipment failures that would result in tunnel contamination by organic materials, i.e. hydraulic fluid. In response to this control, Kiewit/PB has developed a Technical Control Procedure titled "TBM Operating and Maintenance Procedure." The procedure requires daily, weekly, and 500-hour maintenance to be conducted using maintenance checklists. These completed checklists, along with other maintenance test documentation, become QA records. Maintenance is performed by the Maintenance group in the Construction Department. The Kiewit/PB QA organization conducts periodic surveillance to ensure the procedure is being followed and properly completed records are being accumulated.

Another example of QA Control is related to water usage in the tunnel. The specifications state in part "The amount of construction water lost in the 7.62-meter diameter Topopah Spring Loop excavation shall not exceed 7.4 m3 per linear meter of tunnel excavated. Verification that this limit has not been exceeded shall be performed for each working shift by dividing the total cubic meters of water lost during the shift (as calculated from the water balance) by the linear meters of tunnel advance for the shift and comparing this value to the above limit." This requirement is addressed in another Technical Control Procedure titled "Water Use and Control - Subsurface." The procedure is implemented by Construction personnel and Kiewit/PB conducts periodic surveillance to ensure that the required meter readings are being recorded, the calculations correctly performed, and the results properly documented as QA records.

A final example of a QA Control procedure involves control of diesel emissions in the tunnel. This control states:

"Engines for all diesel powered equipment used in the North Ramp and associated alcoves shall be maintained in accordance with the manufacturer's recommendations, including adjustment of the fuel system, proper maintenance of the engine air intake system and exhaust

conditioning system. Exhaust emissions shall be checked/ monitored prior to first use in the North Ramp and thereafter as follows:

a. When excessive emissions are visually observed (excluding warm-up periods or hard acceleration when smoking is normal)

b. Prior to taking the equipment underground following maintenance of any emission related engine system

c. At not more than 500-hour intervals (+ 5%) of engine operation."

The Control further states that exhaust emission constituents to be monitored include diesel particulate matter (DPM), oxides of nitrogen (NOx), sulphur dioxide (SO2), carbon monoxide (CO) and carbon dioxide (CO2). Again, these controls are addressed in a Technical Control Procedure titled "Diesel Exhaust Emission Testing, Monitoring, and Control Procedure." Results of emission tests are conducted and the results documented by Construction Maintenance personnel. Kiewit/PB QA conducts periodic surveillance of this testing to ensure that it is being properly conducted and documented.

In total, the specifications applicable to the Kiewit/PB scope of work contain over 600 QA Controls which must be addressed in QA, work, and inspection procedures.

CONCLUSIONS

TBM tunneling operations have now been underway for over a year at Yucca Mountain. During this time, a QA program based upon nuclear standards has been successfully applied to these construction activities. This program has been under constant scrutiny in the form of QA audits and surveillances by outside organizations. Few, if any, serious programmatic problems have been identified. Because personnel experienced with tunnel design and construction typically have little or no experience with nuclear quality assurance requirements and, because personnel experienced with nuclear QA requirements have little or no experience with tunnel design and construction, implementation of the QA program has not always been without pain. Overall, however, implementation has gone well by providing the proper mix of personnel with tunneling experience and personnel with nuclear experience. Providing the appropriate mix of experienced personnel at the earliest possible point in the design and construction process is the key to success on a project such as this. In addition, assigned personnel must be made aware of the necessity of relying on each others experience, whether nuclear or tunneling, to achieve project success.

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THERMAL ANALYSIS OF REUSABLE SHIELDED TRANSPORT CONTAINERS FOR ILW

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ABSTRACT
UK Nirex Ltd (Nirex) is developing a deep repository for the disposal of intermediate level and some low level radioactive waste. Nirex is also responsible for producing standard designs of transport containers, including a range of reusable shielded transport containers (RSTC) which are being designed to meet IAEA Type B requirements. The contents of the RSTCs will normally be cemented ILW in either four 500 litre drums or a 3m3 box or 3m3 drum of similar outside dimensions to the four 500 litre drums in their transport frame.
This paper presents an analysis to demonstrate the performance of the RSTC and its contents during the fire test specified in the IAEA Transport Regulations. Temperatures, activity release and internal pressure have been calculated.
The thermal analysis was carried out using OASYS LS-DYNA3D, a finite element code for three-dimensional structural and thermal analysis. Since the IAEA Transport Regulations require an impact test to precede the fire test, the thermal models include the deformation resulting from a 9m drop on to a rigid target. Thermal insulation is provided on the surface of the container to limit temperature rises in a fire, particularly in the region of the elastomer seals. The effect of partial loss of this insulation in the impact has also been analyzed.
Temperature rises throughout the RSTC and four 500 litre drums of cemented sludge wasteform have been calculated. Activity release into the internal cavity of the RTC has been calculated from the waste temperatures and the results of small-scale tests on active waste samples, taking no account of the integrity of the welded stainless steel drums. Steam will be generated from the waste during the fire test, so the steam pressure has also been determined.
The analysis has demonstrated that temperature rises are acceptable, and that activity is only released from a small fraction of the waste volume that reaches elevated temperatures. The total release is well below the limit of 1A2 in one week. Steam pressure inside the RSTC is well within design limits and there is no significant increase in the leakage rate due to reduction in seal compression. These conclusions apply even with some loss of the thermal insulation.

INTRODUCTION

UK Nirex Ltd (Nirex) has been established by the UK nuclear industry, with the agreement of the UK Government, to develop and operate a deep repository in the UK for the disposal of solid intermediate and some low level radioactive wastes (ILW and LLW). Nirex is also responsible for developing an integrated transport system for the movement of waste from waste producing sites to the repository.
Reusable shielded transport containers (RSTC) will be required to transport some types of ILW and will comply with the requirements of the IAEA Transport Regulations (IAEA, 1990) for Type B packages. These include meeting containment and shielding requirements following an impact on to an unyielding surface from 9m and a subsequent 30 minutes, 8000C fully engulfing hydrocarbon fire.
The RSTC will transport ILW in either four 500 litre drums in a transport frame (stillage), or a 3m3 box or 3m3 drum. The RSTCs will be produced in a range of shielding thicknesses from 70mm to 285mm, to suit the requirements of the different waste streams. The RSTC development program has now reached a stage where a single preferred concept has been

adopted; Fig. 1 illustrates the 70mm and 285mm thick versions (RSTC-70 and RSTC-285).

Fig. 1

This concept has been designed so that the sealing function does not depend on the lid. Sealing is provided through the use of a stainless steel lid seal member (LSM) which is clamped to the body of the container and is independent of the lid. The lid is held in place by 24 radial chocks around the periphery. A shock absorber is attached to the underside of the LSM to protect it from damage by the RSTC contents in inverted impacts, and this also reduces the impact force transmitted by the contents to the lid.

A notable feature of the design is the solid metal flow shock absorbers on the four corners of the body. These limit the accelerations and impact experienced by the RSTC for any inverted orientation by deforming in plastic manner. A stainless-steel clad balsa shock absorber is attached to the top of the lid for lid-down impact attitudes.

To provide thermal insulation for the fire accident scenario the design considered in this paper includes an external intumescent coating, although alternative thermal protection methods are currently being considered. This coating does not significantly affect the thermal performance of the RSTC under normal conditions. However, under fire accident conditions, the thermal resistance of the intumescent coating increases as the material foams up. This limits the amount of heat absorbed by the RSTC and also the maximum internal temperatures.

Various aspects of the design have been described in detail elsewhere (Sievwright et at 1991; Smith et al 1992; McKirdy et al 1994). This paper presents the analysis carried out to demonstrate the performance of the RSTC during the 30 minute, 8000C fire test specified in the IAEA Transport Regulations (IAEA, 1990).

SCOPE OF THE ANALYSES

The RSTC thermal analysis carried out had the following objectives:

To determine the steady-state temperature conditions in the RSTC under IAEA specified ambient conditions for a Type B(U) package (+380C and specified solar radiation input).

To determine the transient temperatures in the package during and after the fire test, up to the point where steady-state conditions were reached once again. The steady-state conditions after the fire test were different from the steady-state conditions before the fire, because the intumescent coating layer has a higher thermal resistance after the fire.

To quantify the increase in temperature of the waste and the seal face due to loss of intumescent coating in an impact before the fire.

The calculated temperatures were used to determine the following information:

Quantity of steam released from the cement grout used to encapsulate the waste, and hence the pressure-time transient within the RSTC cavity.

Activity released from the waste packages into the cavity of the RSTC.

Thermal distortion of the RSTC.

Performance of LSM.

ANALYSIS METHODOLOGY

The IAEA Regulations specify that the thermal test must follow the impact tests; hence the thermal analysis must take account of impact deformation and possible loss of intumescent coating. An impact analysis was therefore carried out to obtain the deformed geometry of the RSTC prior to the start of the fire. One scenario examined involved an impact in a

lid corner attitude. This results in a large 'knock-back', and a loss of intumescent coating from the shock absorber close to the seal face, which would cause an increase in seal face temperatures. The amount of the intumescent coating lost was determined from separate tests. To quantify the effect of the loss of insulation, an impact damaged model with insulation intact was also analyzed.

The requirement that the thermal distortion of the RSTC and the performance of the LSM had to be evaluated implied that a thermo-mechanical finite element (FE) analysis of the RSTC was necessary. The finite element code chosen was OASYS LS-DYNA3D (Oasys Ltd, 1994). This code is renowned for its capabilities in analyzing non-linear dynamic problems. Recently its capabilities were extended to include thermal analysis by incorporating the heat transfer code TOPAZ3D (Shapiro, 1985). It is now possible to use LS-DYNA3D to perform fully-coupled thermo-mechanical analysis.

Prior to the commencement of the work it was decided that the most efficient way to carry out the analyses would be to use the same FE models for both the thermal and mechanical analyses. Figure 2 shows the finite element mesh for the RSTC-285 as an example. In general 8-noded solid elements were used. On some components where there was only a single layer of elements (eg the parts of LSM, and the stillage) a layer of shell elements was also used to ensure that the correct elastic stiffness was maintained. The bolts in the clamps holding the LSM to the body were modelled using spring elements. A plane of symmetry was taken through the damaged shock absorber, this reduced the size of the FE models by half. The total number of elements used in a half model was 57,044 for the RSTC-285 and 44,708 for the RSTC-70.

Fig. 2

All features of LS-DYNA3D that were used in the analyses were tested by running simple models and comparing the results with closed-form solutions. More complex models of actual physical tests were developed and results compared with those from tests, for example tests on 500 litre drums subjected to heating in a furnace. In this way it was possible to demonstrate with confidence that realistic results could be obtained from the full models.

Representative material properties were assumed, and other assumptions were based on the IAEA test requirements. The thermal properties of the wasteform were based upon those measured by Nirex for a cemented sludge wasteform (Bush et al 1990). The drums were assumed to generate a maximum of 50 watts each, which is the maximum permitted in the Nirex Waste Package Specification. The evaporation of steam from the wasteform was modelled by including a 'spike' in the specific heat of the wasteform at 1000C.

RESULTS

Normal Conditions of Transport

The steady-state analyses of an undamaged RSTC-285 and RSTC-70 for normal conditions of transport gave predicted maximum waste temperatures of 660C for the RSTC-285 and 650C for the RSTC-70. These maximum temperatures were located towards the top of the drums and are primarily related to the insolation, which is highest on the top surface of the RSTC (IAEA, 1990). It was predicted that steady-state conditions would be achieved after 35 days. Temperatures are highest near the center of the RSTC while the lowest temperatures are at the bottom corner of the drum. This indicates a temperature gradient which causes heat flow out of the drums

and towards the walls and base of the RSTC. The waste temperatures are acceptable because the maximum is well below 1000C, the temperature at which steam would begin to be driven of the cement matrix which immobilizes the waste.

Fire Test Conditions

The model used for the fire test analysis used as its initial conditions the calculated steady-state temperature distribution, together with the deformed geometry obtained from an analysis of a 9m lid corner drop onto an unyielding target. The effects of both undamaged and damaged intumescent coating were also taken into account. The stillage and drums were positioned as close as possible to the damaged corner so as to maximize the heat input to the waste.

Figures 3 and 4 show the predicted temperature distribution for the RSTC-285 and RSTC-70 with damaged intumescent coating at 5 hours after the commencement of the 30 minute fire. The corresponding temperatures for the undamaged intumescent coating are shown in Figs. 5 and 6. This was the time at which the temperature peaked. After the end of the fire, the heat absorbed by the RSTC flows into the waste as well as to the surrounding atmosphere, thus causing the peak waste temperatures to occur some time after the end of the fire. In these figures only the body of the container and the waste are shown for clarity.

Fig. 3

Fig. 4

Fig. 5

Tables I and II summarize the principal results. It can be concluded from these results that the only significant effect of the partial loss of intumescent coating on the shock absorber is to increase the peak temperature on the seal face of the RSTC-70 by 810C. The maximum temperatures of the waste for the RSTC-285 and RSTC-70 were 1200C and 1070C respectively and occurred after 5 hours. Table III summarizes the strains in the major RSTC components resulting from the thermal transient for the undamaged intumescent coating scenario.

Tables I and II show that the predicted activity releases into the RSTC cavity are three orders of magnitude below the limits for release from the package as a whole. Therefore the performance of the seals is not critical in this scenario.

Table I

Table II

The RSTC-70 with damaged intumescent coating had the highest seal face temperature of 1820C. This is due to having a smaller wall thickness and hence a shorter heat path to the seal face. The temperature-time history of the seal face is shown in Fig. 7. In tests (McKirdy et al 1994) the candidate seal was subjected to 5 hours at 2000C with satisfactory performance. Hence the seals will still be effective in providing containment to the RSTCs after the fire.

Fig. 6

Fig. 7

The analysis showed that the current designs of RSTC-285 and RSTC-70 with damaged intumescent coating are capable of satisfying the IAEA fire test requirements (IAEA 1990) following an impact from 9m on to an unyielding target.

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7-30

A VALIDATED IMPACT ANALYSIS MODEL FOR ILW TRANSPORT CONTAINERS

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ABSTRACT

UK Nirex Ltd (Nirex) is developing a deep repository for the disposal of intermediate level and some low level radioactive waste. Nirex is also responsible for producing standard designs of transport containers, including a range of reusable shielded transport containers (RSTCs), which are being designed to meet IAEA Type B requirements. The contents of the RSTC will normally be cemented ILW in either four 500 litre drums, or a 3m3 box or a 3m3 drum of similar outside dimensions to the four 500 litre drums in their transport frame.

This paper describes the development of a validated impact analysis model for the RSTCs, based on predicted and measured results for one-third-scale models with two different shielding thicknesses.

Decelerations, strains and local impact deformations have been both measured and predicted, for a series of drop tests from 9m which comply with the IAEA Transport Regulations. A finite element model has been developed using the DYNA3D code, which can model the large plastic strains and transient loading within acceptable computing times.

The model has been validated in the following respects. It can accurately predict the extent and shape of knock-back displacements in the impact zone. The model can predict deceleration transients reasonably well, with good agreement when both the measurements and the predictions have been processed by low-pass filtering. Strains are predicted with varying levels of agreement, partly because the predictions are for the centroids of elements, and not for the surfaces where the strains were measured.

INTRODUCTION

UK Nirex Ltd (Nirex) is responsible for developing and operating a deep repository in the UK for the disposal of intermediate and some low level radioactive wastes (ILW and LLW). Nirex is also responsible for producing standard designs of waste packages and transport containers, as well as the development of an integrated transport system for the movement of waste from waste producing sites to the repository.

For the transport of some types of ILW, reusable shielded transport containers (RSTCs) will be provided. The RSTCs are designed to comply with Type B requirements of the IAEA Transport Regulations (1). Nirex is currently developing RSTC designs in a range of shielding thicknesses, ranging from 70mm to 285mm for this purpose, depending on the waste stream. The possible contents for the RSTCs are four 500 litre drums in a transport frame (stillage), a 3m3 drum or a 3m3 box.

The RSTC development program has now reached a stage where a single preferred concept has been adopted. Figure 1 shows the 285mm and 70mm versions (RSTC-285 and RSTC-70). The key features of this design are:

The lid is retained by 24 cam-operated chock mechanisms arranged at 300 to the horizontal, and can be lifted and handled using the pintle. A stainless steel clad balsa wood shock absorber is located on the top side to limit impact forces.

The lid has no sealing function in this design. Sealing is provided by the lid seal member (LSM), which is a thin plate containing O-ring seals in a thicker outer rim, which is clamped to the body. An internal shock absorber is attached on the underside of the LSM to protect it from damage by the RSTC contents during impact.

Primary protection for lid-down impact orientations is provided by the solid metal shock absorbers which are located at the corners of the container. In impact these deform plastically and hence act to reduce the decelerations on the RSTC.

Fig. 1

Various aspects of the design have been described elsewhere (2,3,4). This paper describes the development of a validated impact analysis model for the RSTC, based on measured results from a series of IAEA regulatory drop tests on one-third-scale models of the RSTC-70 and RSTC-285. The purpose of this was to develop and validate finite element (FE) models that could be used for predicting the effects of any subsequent design changes.

DROP TESTING

The IAEA Transport Regulations (1) require a 9m drop test onto a flat, rigid target in the least favorable impact attitudes. Compliance may be demonstrated by a combination of analytical modelling and practical drop tests. A series of fully-instrumented drop tests were there fore carried out on one-third-scale models of the RSTC-70 and RSTC-285, each carrying four one-third scale 500 litre drums in their transport frame.

The impact attitudes used in the drop tests were selected from the results of earlier analytical modelling, which showed that the potentially most damaging attitudes would be with the center of gravity over a lid corner or over a lid edge, or a flat impact on one side.

The drop test tower incorporated vertical guide rails, and runners were attached to the RSTC scale models to ensure that it did not rotate out of the desired impact attitude while falling. The last two metres of the drop test were in free-fall, but high speed photography showed that the correct impact attitude was achieved within a fraction of a degree.

The instrumentation included accelerometers and strain gauges connected to high-speed data recorders by a free-falling umbilical cable; for example Fig. 3 shows the locations of some of the accelerometers and strain gauges for the lid corner drop of the RSTC-70 model. Two high-speed cameras filmed each drop from orthogonal viewpoints and all image frames were accurately timed. Each drop was also recorded using a video camera. Several reference points were scribed on the container body and lid to enable any distortions resulting from the drop to be established by physical metrology.

The instrumentation provided data that could be compared directly against the predictions of the finite element (FE) models described below. The results of the drop testing are presented as part of the comparisons with the model predictions.

DEVELOPMENT OF FINITE ELEMENT MODELS

FE models were generated for the two containers of different nominal wall thickness for each of the three impact attitudes. The models were analyzed using DYNA3D (5).

The FE analyses were based on the same basic model which is an eighth slice of the RSTC. This section was then rotated and reflected to generate the required model. It was assumed that all impact damage would be symmetric and hence it was only necessary to model one half of the RSTC. It was recognized that the RSTC could bounce and impact again in a non-symmetric manner but at this late stage (typically after 0.1 seconds) the majority of the available impact energy would have been absorbed.

To simply the analysis it was assumed that the drums and frame would act as a solid mass, and that the drums could not move relative to each other. For lid corner and lid edge impact, the mass of the frame and drums acted continuously on the internal shock absorber as a single mass. Therefore it was not possible to measure drum movements. However, visual examination after the tests showed that in most cases the drums remained tightly fixed in their frame.

The containment integrity was determined after each drop test by a gas leak test. To attempt to model the actual behavior of the LSM in the FE analysis would require a very detailed model and it was assumed that the LSM rim remains in position and does not move relative to the RSTC, and so the clamping mechanism was not explicitly modelled.

The wooden shock absorber on the lid was also omitted because it does not affect the structural response of the RSTCs in the impact attitudes modelled. The steel and concrete target used in the drop tests was modelled explicitly, since it cannot be assumed that any practical impact target is 'rigid'.

Appropriate material properties were used throughout the FE models. The upper parts of the external shock absorber 'ears' (Figs. 1 and 2) were modelled with strain-rate enhanced yield stresses, and the internal shock absorber was modelled with a constant crush stress.

Fig. 2

Fig. 3

Measured coefficients of friction between the RSTC and the target are not available. Analysis of the results of the first drop test indicated that a value of 0.15 gave good agreement, and the same value was used for all other tests. Elsewhere in the model, the same value was used for other coefficients of friction, with two exceptions. Instead of modelling the lid-retaining chock mechanisms, the coefficient of friction around the chock locations was increased to 0.75 to give an adequate representation

of the chock performance. Friction was excluded from the contacts between the transport frame and the RSTC cavity walls, ensuring that the full weight of the RSTC contents could act on the internal shock absorber. For both the RSTC-70 and RSTC-285, the following impact attitudes were considered, in each case for a 9m free drop producing an impact velocity of 13.3m s-1.

1. Lid corner - The plane of symmetry cuts diagonally through from the impact point to the uppermost base corner (Fig. 4). The contents have already slid down to rest against the interior shock absorber. The model consisted of 11297 elements.
2. Lid edge - The plane of symmetry cuts through the center of the RSTC from the impact edge to the uppermost base edge and both corner shock absorbers contact simultaneously. The contents have already slid down to rest against the interior shock absorber. The model consisted of 13340 elements.
3. Flat side - The contents are resting on the impact side. The model consisted of 16017 elements.

The accuracy of a finite-element model is dependent on the mesh size, mesh density, geometrical accuracy and assumptions regarding elastic and plastic material properties, friction, and strain rate dependency. Sources of experimental inaccuracy include; asymmetry of impact; the positions of strain gauges and accelerometers, and how rigidly they were attached; timing the moment of first contact; the settings and performances of trace filtering hardware; damage to the impact area during secondary impacts following the main rebound; and the cumulative deformation of scale models that had been dropped more than once. These were taken into account in developing the FE models, as far as possible.

Fig. 4

VALIDATION OF MODEL

The key parameters for validating the FE model are knockback (ie permanent deformation of the impact area), accelerations and strains. Owing to the volume of data generated for the two wall thicknesses, each dropped in three attitudes, only a summary is given here.

A major approximation in the validation was that although the same scale models were used for more than one drop test, the FE models were assumed to be undistorted. Post-test metrology supported this assumption because permanent distortions outside of the impact area were minimal, and the RSTCs were always dropped onto undamaged areas.

The duration of the primary impact for all three attitudes was in the region of 10msec. The measured deformations in general agreed very well with the predictions, both in shape and in the extent of knockback from the unperformed profile. For the 9m lid corner and lid edge attitudes a comparison of predicted and measured knockback of the metal shock absorber is provided in Table I. Agreement in the deformation results was within 5mm. Variations between the results are consistent with the RSTC-285 being heavier than the RSTC-70, and the fact that in the lid edge attitude two shock absorber are activated as opposed to one for the lid corner. In a flat side impact, energy is absorbed by the deformation of the ribs on the side of the RSTC; good agreement was again achieved.

Table I

All accelerometer traces were low-pass filtered with a cut-off frequency of 400Hz because filtered traces can be compared more readily with filtered acceleration data from the FE model. A representative selection of plots comparing predicted and measured accelerations are shown in

Figs. 5 and 6, for the two RSTCs and the three impact attitudes. In general good agreement was obtained both in the timing and magnitude of the peak accelerations.

Fig. 5

Fig. 6

The maximum accelerations for the RSTC-285 in the lid corner and lid edge attitudes were 300g and 400g respectively. The correlation for the RSTC-285 flat side impact does not show good agreement because the RSTC in the test did not land quite squarely on all the ribs together. This could not be represented in the FE analysis, as symmetry boundary conditions were applied. The maximum acceleration was 1400g.

The maximum accelerations for the RSTC-70 were 520g in lid corner attitude, 620g lid edge attitude and 1200g in the flat side attitude. Measured strains and deformation contours detected small asymmetries in the way the containers landed on most occasions. In the FE model the areas of the container that remain elastic are more subject to numerical 'noise' in the strain data than those in areas of higher deformation, but the measured traces for the gauges in the elastic regions generally changed from compressive totensile near the times predicted by the model. Measured strain rates up to 100 s-1 were recorded; no gauges could be mounted at the points of impact but predicted strain rates in these areas were of the order of 4000 s-1 for the 70mm container and 1600 s-1 for the RSTC-285.

There was reasonable agreement in strain gauge data. Known sources of inaccuracy from the model are the use of straight-sided elements to represent curved surfaces such as the corner shock absorbers, and the fact that the model calculates strains at the centroid of the 3-D 'brick' elements whereas all measurements were made at the surface. This could be overcome by coating the surface with thin 'shell' elements. Because of the large variations in strains encountered at a single point (eg -1000me for one gauge) and the high strain rates (up to 4000 s-1) the use of finite elements closer in size to the strain gauges themselves would give more accurate results.

CONCLUSIONS

Both FE models have been validated in the following respects:

1. The extent and shape of knockback displacements in the impact zone can be accurately predicted, generally to better than 5mm.
2. Deceleration transients in terms of duration and magnitude showed good agreement between measurements and predictions.
3. Strains are predicted with varying levels of agreement, partly because predictions are for centroid of elements and not for the surfaces where strains were measured.

This work has demonstrated that the FE method can be used successfully and with confidence to predict the acceleration, knockback and extent of local plasticity for an RSTC.

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LHS MODELING OF PACKAGING RELEASE CHARACTERISTICS FOR RADTRAN 4 TRUCK ACCIDENT-RISK ANALYSIS

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ABSTRACT

Previously, we have investigated the sensitivity of RADTRAN accident-risk results to the required input parameters and studied key variables using Latin Hypercube Sampling (LHS). A general conclusion of this work is that the total risk uncertainty does not become excessive when rather conservative distributions are used to describe input parameters such as atmospheric turbulence conditions, highway accident rates, population densities, and package release fractions. However, packaging release fractions, which correspond to the spectrum of accident severities, typically range over a few orders of magnitude, and modeling their values with the simplest distribution functions was unsatisfactory in some respects. Because efforts to calculate release fractions are greatly complicated by the large varieties of packaging characteristics and detailed accident conditions that may affect the outcome, explicit inclusion of uncertainty by use of LHS and modeling of physical conditions through improved choice of distribution shape appeared to be a productive course.

We describe tests of more complex distribution functions, such as Lognormal and Bounded Normal, in fitting the point estimates of release fraction values. These distributions and their descriptive parameters were used as input parameters to Sandia National Laboratories' Latin Hypercube Sampling code to generate 100 sets of RADTRAN 4 input parameters used together with point estimates of other necessary inputs to calculate 100 observations of estimated accident dose risk. This procedure was applied to several distinct transportation situations to reveal the statistical uncertainty of accident-risk estimates calculated by RADTRAN 4. These results are in turn compared to previous results, obtained with simpler Uniform and Loguniform distributions, to assess potential improvements in statistical behavior and absolute risk estimates.

Difficulties encountered and examples of cases that yielded improvements are presented for Type A and Type B packagings employed in representative truck transport situation.

INTRODUCTION

One of the choices that must be made in using Latin Hypercube Sampling (LHS) (1) with the RADTRAN Code for transportation risk assessment (2) is the type of distribution function to use in describing the uncertainty of a particular variable. Four options available with LHS are Uniform, Loguniform, Normal and Lognormal distributions; these adequately describe the uncertainty of most of the RADTRAN variable types selected for LHS analysis (3). These include the Pasquill probability fractions which have been described by Uniform distributions of values between 0 and 1. The uncertainty in two other variable types, link population densities (LPOPD) and link accident rates (LARAT), have been described by Normal distributions since, in general, they are averages with standard deviations obtained from the census and other surveys.

In previous work (3,4), release fraction (RFRAC) parameter values (the source terms for subsequent dispersion), were modeled by Uniform or Loguniform distributions spanning one or more orders of magnitude. Those studies demonstrated that Loguniform distributions yield smaller average risk estimates than Uniform distributions, in agreement with expectation, but the variances were large, especially if the loguniform distribution spanned more than one decade. The release fractions employed in a RADTRAN calculation are critical in determining the resultant risk but experimental measurement of releases from particular packagings under various accident conditions is very costly in time and money. Therefore, improved, conservative modeling of releases that will not generate inflated risk estimates or variances, and that can be derived by extrapolation of existing data is quite desirable.

The results of modeling release behavior of Type A and Type B packagings by use of available distribution shapes and widths are presented here. In addition, we discuss investigations of the effect of aggregating uncertainty by reducing the number of severity levels and modeling the change of release fraction versus severity level with a selected distribution shape. The goal in all cases was to maintain reasonable conservatism without introducing excessive uncertainty in the risk estimates.

SHIPMENTS OF TYPE A PACKAGES

The spectrum of severity levels and corresponding release fractions for this type of package may be divided into eight levels with the lowest level having no release (5). The increase in release fraction over the remaining seven levels is then defined according to the specific nature of the shipment being analyzed. For the examples presented here (multiple Type A packages), the highest five severity levels result in 100% release of the package contents; this suggested simplifying the analysis by reducing the spectrum of severities to three levels and modeling the corresponding release fractions by various distribution functions. The results of modeling the spectrum of release fractions with Uniform and Loguniform distributions over the indicated ranges are shown in Table I. The characteristic difference between these two types of distributions is evident in the shifted mean risk values and increased standard deviations for the Lognormal cases. Results obtained with Normal and Lognormal distributions in the three-level scheme are shown in Table II.

Table I

Table II

SHIPMENTS OF TYPE B PACKAGES

Calculations of risk estimates for a standard test case have been performed using various specifications of all four distribution functions

examined. This case is particularly instructive because it describes shipments of spent nuclear fuel (SNF) and CRUD (scale containing cobalt 60 which is deposited on the exterior of the fuel cladding during pool storage). The magnitudes of accident risk estimates calculated for such shipments are determined almost entirely by CRUD release. We again investigated potential simplification through aggregation of uncertainty in a reduction from six severity levels (6) to four. Tables IIIa and IIIb illustrate how the severity fractions of levels 4, 5 and 6 were summed to yield one level while levels 1, 2 and 3 include nearly 100% of accidents. In the case of CRUD, level 3 (in addition to 4, 5 & 6) results in release because it is external to the full cladding; past practice has been to assign these levels a release fraction (1.2E-2) that corresponds to total release of all suspended material inside the cask. This has been recognized as an extremely conservative model for the vast majority of accidents. For the current calculations, this value was taken as the maximum and the increase from 0 to 1.2E-2, over the range of severity levels, was modeled by a variety of distribution (and sampled by LHS). The results of calculations with the various distribution functions and defining parameters are given in Table IV.

Table IIIa

Table IIIb

Table IV

CONCLUSIONS

For Type A packagings, comparison of the results in Table I indicates that neither Uniform nor Loguniform distributions in a three-level scheme provide a fully satisfactory model in that average risk values tend to increase slightly and the standard deviations increase significantly, especially in the case of the Loguniform distribution. Use of Normal distributions yields results (Table II) which are similar to those for the Uniform and Loguniform distributions without the large standard deviation of the Loguniform case. Finally, use of a Lognormal distribution gives values which are very similar to the 8-level results with a modest increase in standard deviation.

For Type B packagings, the values shown in Table IV clearly show that use of the Loguniform distribution over a parameter range of many decades has two effects: it lowers the average significantly but the fractional standard deviation (%) is in turn increased significantly. The large standard deviation is exacerbated by the fact that the risk is dominated by one distribution: CRUD release. It seems that a more satisfactory means of reflecting the statistical prevalence of lower RFRAC values is to employ a Bounded Normal distribution with the mean at the lower end of the desired RFRAC range. Use of the Bounded Lognormal distribution provides even further improvement in that the averages are comparable to the 6-level values and the relative standard deviations are less than those for the other distributions.

In general, it is concluded that distribution functions available in the Latin Hypercube Sampling code can be used to model the increase of release fraction with increasing accident severity. Inexactness in knowledge of the scaling with discrete severity levels can be addressed by aggregating them into combined severity levels. Most importantly, in many cases the least specific distributions (Uniform and Loguniform) can be employed without altering average risk estimates significantly or producing such large relative standard deviations as to diminish the

value of the analysis, while at the same time permitting the analysis to be simplified.

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INVESTIGATION OF RADTRAN STOP MODEL INPUT PARAMETERS FOR TRUCK STOPS*

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ABSTRACT

RADTRAN is a computer code for estimating the risks and consequences associated with the transport of radioactive materials (RAM). RADTRAN was developed and is maintained by Sandia National Laboratories for the U.S. Department of Energy (DOE). For incident-free (i.e., no transportation accidents or off-normal occurrences) transportation, the dose to persons exposed while the shipment is stopped is frequently a major percentage of the overall dose. This dose is referred to as Stop Dose and is calculated by the Stop Model. Because stop dose is a significant portion of the overall dose associated with RAM transport, the values used as input for the Stop Model are important. Therefore, an investigation of typical values for RADTRAN Stop Parameters for truck stops was performed. The resulting data from these investigations were analyzed to provide mean values, standard deviations, and histograms. Hence, the mean values can be used when an analyst does not have a basis for selecting other input values for the Stop Model. In addition, the histograms and their characteristics can be used to guide statistical sampling techniques to measure sensitivity of the RADTRAN calculated Stop Dose to the uncertainties in the stop model input parameters. This paper discusses the details and presents the results of the investigation of stop model input parameters at truck stops. The current investigation was limited to

refueling stops, but the methodology developed can be applied to stops with other purposes.

INTRODUCTION

The objective of this investigation was to determine statistically meaningful estimates of the average values of the parameters which affect the calculation of stop dose at truck stops. Stop parameters observed for trucks in general are considered to be representative of RAM transportation trucks specifically because whether or not a truck actually carries RAM should not affect its time to refuel. This investigation has not attempted to characterize road-side stops or stops at any other location than truck stops.

To begin this investigation, it was necessary to identify which parameters affect the stop dose. This was accomplished by studying the documentation supporting the RADTRAN Stop Model (1) and an earlier investigation "Truck Transportation of Radioactive Materials" (2). The parameters identified as most directly affecting stop dose are: the duration of the stop, which is referred to as Stop Time; the number of persons exposed to the truck while it is stopped, which is referred to as Number of Persons Exposed (NPE); and the distance from the truck to the exposed persons, which is referred to as Exposure Distance. The NPE can be divided into people outside at the truck stop (i.e., without radiation shielding), and people inside buildings at the truck stop (i.e., with radiation shielding). Thus, the NPE should be observed as two parameters: NPE-outside, and NPE-inside buildings. In addition, the building materials should be noted as they could affect shielding values.

The prototype RADTRAN 5.0 Stop Model allows two options for specifying the NPE at a stop: 1) User specified number of people at a fixed exposure distance from the RAM shipment; and 2) a user specified number of people within an annular region with user specified minimum and maximum exposure distances (3). In addition, RADTRAN 5.0 allows for up to 20 distinct stops to be modeled in a single RADTRAN run. Hence, a refueling stop can be modeled with separate input values than a weigh station stop or a repair stop. A stop during peak time (with respect to congestion at the truck stop) can be modeled separately from a stop during off-hours.

To determine the range of values possible for the Stop Model input parameters, factors expected to affect these input values were identified. Factors possibly affecting Stop Time, NPE, and Exposure Distance include: truck stop location (i.e., rural, urban, or suburban, proximity to other commercial establishments, etc.); time of day; and purpose of the stop (i.e., refueling, service, etc.). Truck stop location is likely to affect the amount of business at that stop and, hence, the number of persons potentially exposed as well as the amount of time needed for service and potential for delays. Time of day was expected to have a similar effect. Truck stop location could also affect the configuration of the truck stop and, hence, the exposure distances. For example, land availability may affect the square footage occupied by metropolitan and rural truck stops, which could in turn, alter the distances between refueling bays, service bays, and restaurants, etc. The purpose of the stop is expected to affect the stop time. For example, a simple refueling stop is likely to be of shorter duration than a stop requiring repair service.

The stop times measured in this investigation were limited to refueling stops because all transports of sufficient distance must refuel, whereas a service stop may be unnecessary. Also, a refueling stop is most likely

of shorter duration than a service stop. Therefore, a refueling stop is a reasonable representation of a routine minimum stop time. Because the RADTRAN 5.0 Stop Model allows for up to 20 distinct stops to be modeled in a single RADTRAN run, the RADTRAN analyst can select a larger stop time when modeling a service stop. The observations of exposure distance and NPE should be applicable to all stops that occur at truck stops.

The actual investigation included traveling to truck stops and measuring stop time, number of persons exposed, and exposure distance. Truck stops were selected in several locations and observations were made at various times of the day and on various days of the week to yield a representative sample of truck stop parameters.

TRUCK STOP OBSERVATION PROTOCOL

A truck stop observation protocol was necessary to help ensure consistency in observation techniques between different truck stops, and to help ensure statistical soundness of the resulting data. We observed each truck stop anonymously so that our presence would not alter the data (i.e., the fuel pump attendant doesn't alter his or her pace, etc.). To avoid any unintentional preferential treatment in selecting which trucks to observe, we decided to measure stop time for every truck that refueled during each observation period. Two persons were involved in performing these observations and were present at each observation. To help ensure uniformity, each person performed the same duties at each observation.

Preliminary truck-stop observations were performed at two truck stops in Albuquerque, New Mexico, and at one truck stop in rural Arizona (Holbrook). The purpose of these preliminary observations was to familiarize ourselves with the conditions at truck stops so that we could develop an efficient and uniform plan for observing data at various truck stops. Based on these preliminary observations, an observation ledger form was developed to help ensure consistency between observations at each stop. The observation form included columns for observing the following variables: truck stop location; observation date; day of week; time of day; stop time; NPE-outside; outside exposure distance; NPE-inside buildings at the truck stop; building material(s); number of trucks parked in the first row (with respect to the refueling bays); distance from the refueling bays to the first row of parked trucks; paving material; and a section for comments. A free-hand sketch of the configuration of each truck stop was also prepared for each truck stop. A video camera was used to record the truck stop observations and a stop watch was used to measure stop time. The video tapes and ledgers are part of the quality assurance file for this project. The observation data are stored in a spread-sheet file (Microsoft Excel) so that data can be added as additional observations are performed.

The number of parked trucks was noted because parked trucks may have one or more persons sleeping inside. These persons could be considered as a distinct subset of NPE-inside. All parked trucks were not counted because persons sleeping in the 2nd or further row of parked trucks would be shielded by the first row of parked trucks.

The exposure zone was defined as the area enclosed by major shielding objects such as building walls and the first row of parked trucks. Since most of the persons travelled from the parking areas to the buildings, the exposure zone was modelled as a circular region with the diameter being the distance between the buildings and the first row of parked trucks.

Preliminary observations indicated that tracking the exposure distance for each individual entering the exposure zone would be nearly impossible because most persons walked through the zone rather than occupying it in a stationary manner. The preliminary observations also indicated that the same individuals did not occupy the exposure zone during the entire stop time for any given truck: people entered and exited the exposure zone at various times during any given truck's refueling period. Therefore, the protocol for observing NPE was defined as follows: the number of persons in any portion of the exposure zone was observed, along with the approximate time that each person occupied the zone (which is referred to as residence time). Summing the residence time for each person observed in the exposure zone yields a time-integrated NPE, with units of NPE-time. Dividing the time-integrated NPE by the total observation time yields an equivalent number of persons occupying the exposure zone for the entire observation period, as if they were not travelling into and out of the exposure zone. With this method of calculating NPE, one person occupying the exposure zone for two minutes is counted the same as two persons occupying the exposure zone for only one minute. This is consistent with RADTRAN methodology because RADTRAN calculates population dose rather than individual dose. Also, this definition of NPE is compatible with the NPE Input Option 2 in the RADTRAN 5.0 Stop Model, in which the NPE is expressed as the number of persons in an annular exposure zone with a user specified minimum and maximum radius. The two NPE Input Options available in RADTRAN 5.0 are discussed in the Introduction section of this report.

In practice, it proved impossible to observe NPE separately for every truck in the refueling area, because there were frequently numerous trucks refueling at the same time or in over-lapping time periods. Therefore, the protocol was adjusted to track the NPE for an observation time period during which many trucks were observed individually for stop time. Each observation period was approximately 60 minutes in duration. Therefore, the data presented in the Results section will show more observations of Stop Time than NPE.

RESULTS

Upon completion of the Truck Stop Observation Protocol, the actual investigation was performed. The actual investigation included traveling to truck stops and measuring stop time, number of persons exposed, and exposure distance. Two metropolitan truck stops in Albuquerque, New Mexico (Fina and Union 76) and one rural truck stop in Tucumcari, New Mexico (Shell) were observed.

The Fina observations were performed on July 24 (Monday), July 27 (Thursday), July 28 (Friday), August 1 (Tuesday), and August 2 (Wednesday) 1995. To provide a sampling of times-of-day, these observations were performed in the mornings and afternoons. The Union 76 observations were performed in the mornings of August 29 (Tuesday), September 6 (Wednesday), and September 7 (Thursday) 1995. The Shell observations were performed in the afternoon of October 23 (Monday) 1995. The average values, standard deviations, and number of observations for stop time, NPE-inside, and NPE-outside for each truck stop individually and for all three trucks stops as a composite, are listed in Table I. The number of trucks parked in the first row (with respect to the refueling bays) and the exposure distance for each truck stop individually and for the composite are also listed in Table I. The number of trucks parked in the first row is not presented as a statistical observation with numerous

observations because the value appeared to be constant for each truck stop; the first row was always observed to be full and, therefore, limited by the number of parking spaces in that row. The exposure distance for each truck stop is also not presented as a statistical observation because that value is constant for each truck stop, but depends on site-specific factors.

Histograms of the stop times for each individual truck stop and for the composite are shown in Figs. 1-4. The histograms and their characteristics can be used to guide statistical sampling techniques to measure sensitivity of the RADTRAN calculated Stop Dose to the uncertainties in the stop model input parameters.

Table I

Fig. 1

Fig. 2

Fig. 3

Fig. 4

Discussion

The results listed in Table I indicate that the two metropolitan truck stops (Fina and Union 76) have nearly identical average stop times, average NPE-outside, and exposure distances, whereas the rural truck stop (Shell) has a lower average stop time, a lower average NPE-outside, and a larger exposure distance.

The exposure distances at both Albuquerque truck stops seem to be practical minimums because they represent the minimum distance needed to accommodate the number of refueling bays and a minimum lane width. The exposure distance at the Tucumcari truck stop is, indeed, larger than that at the Albuquerque truck stops.

Qualitative observations are useful for helping the analyst to design and perform analyses of truck stops. The first qualitative observations concern the refueling process. A general observation is that the truck stop employees work efficiently to refuel all of the trucks quickly. Dual fuel tanks were usually filled simultaneously; a separate fuel pump was used to fill each tank. Also, most of the truck drivers removed their trucks from the refueling bays promptly after the refueling was completed. All of these factors combine to minimize waiting time at the refueling bays. This is consistent with industry practice, as most truck drivers are concerned with travel time and strive to minimize their own delays and those of other truck drivers.

A second qualitative observation is that truck stops, in general, are quite busy. People concentrate around the restaurant, fueling area, mechanical shops, pay stations, and gift/supply stores. Numerous automobiles were observed parked at the restaurants at both Albuquerque truck stops, the Tucumcari truck stop, and the Holbrook truck stop. This indicates that people not associated with truck transport (i.e., local residents, automobile travellers, etc.) patronize the restaurants and other truck stop amenities. In addition, at the Albuquerque truck stops, people were observed walking through the exposure zone in route from one neighboring business to another; these people were not using truck stop services.

A third qualitative observation concerns trucks parked at the truck stop. The majority, approximately 90%, were equipped with "sleeper" cabins located aft of the crew cabin. This supports the concern that each parked truck could potentially contain at least one sleeping person who could be

exposed to a RAM transport. However, these persons would be significantly shielded by the tractor cab and engine compartment.

A last qualitative observation concerns building materials because they may affect shielding. Most of the buildings were of concrete cinder-block construction. However, some associated out-buildings such as the weigh-station booth, fuel payment-booths, etc., might be constructed of different materials providing less shielding.

SUMMARY

This paper discusses an investigation of stop model input parameters at truck stops. The parameters of stop time, number of persons exposed (inside building), and the number of persons exposed outside are presented as average values with standard deviations. Histograms of the stop data are presented as an aid for the analyst who may want to use statistical sampling techniques. The results presented in this report indicate that a metropolitan truck stop has a longer average stop time, larger average NPE, and closer exposure distances than the rural truck stop. However, fewer observations of the rural truck stop were performed than for the metropolitan truck stops. This data is stored in a spread-sheet file (Microsoft Excel). This format will allow new data to be added to the database as additional observations are performed.

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DEVELOPMENT OF VITRIFIED WASTE STORAGE SYSTEM

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ABSTRACT

Ishikawajima-Harima Heavy Industries (IHI) has developed its own radioactive waste vitrification and storage technology. Concerning the vitrified waste storage system, IHI has developed and designed two type of storage systems. One is a forced convection air cooling system, and the other is a natural convection air cooling system. IHI has experimented and analyzed heat transfer, earthquake proofing, vitrified waste drop and radiation shielding, etc. In consideration of these understandings, IHI designed and constructed the first natural air cooling vitrified waste storage facility in Japan.

INTRODUCTION

The method of treating the high-level radioactive waste is that, to vitrified glass solidum to safe forms, and store the vitrified waste for 30 to 50 years for intermediate storage. Japan has two option program for vitrification, one is the domestic reprocessing plant, the other is the overseas reprocessing contract with COGEMA in France, and BNFL in England.

Ishikawajima-Harima Heavy Industries (IHI) has developed intermediate storage system, two of which are currently available. One is a forced convection air cooling system for domestic vitrification, and the other is a natural convection air cooling system.

CONCEPT OF STORAGE SYSTEMS

Regarding the design of the vitrified waste systems, some type of management systems can be provided. Table I shows the design concept of Vitrified Waste Storage systems (1). This classification takes notice of decay heat cooling function. To think about another functions following, enclose, shield, long time adaptation, economical adaptation, and so on. In these cases, vault air cooling systems have been validated by the following:

water cooling system is needed for the water circulation and the filter system.

the silo and the cask cooling system needs the large site area, and thus is not good for space efficiency. And the air flow is more complex than the inner flow in the annular tube.

Regarding the vitrified waste storage system is development, we have completed the design of two types of storage systems: one is a forced convection air cooling system (Fig. 1.), and the other is a natural air convection system (Fig. 2.) (1).

Table I

Fig. 1

Fig. 2

The forced air convection system is as follows: The vitrified waste is positioned in the tube directly the cooling air. The cooling air is induced by the ventilation system, which has blowers, filters, etc. and needs the electric power supply units. This system can perform without electric power, but needs emergency electric power. Because this system must be able to operate with the loss of the electric power supply. Therefore, the forced air convection air cooling system, adapts a small scale storage facility, because the ventilation system can be used in common with the storage area and other utilities, for example the vitrification cell. The natural air convection system is as follows: The vitrified waste is positioned in the thimble tube and the cooling air only contacts the outer surface of the thimble tube and not to contact the vitrified waste, so that the cooling air does not become contaminated. The decay heat is removed with the air flow induced by the draft force, which depends on the heat generation rate. The natural air cooling maintains a wide margin on acceptable vitrified waste for all site specific environmental conditions. And IHI has developed cooling properties, shielding, potential dropping and others for the natural air convection system.

The world experience of the interim storage facilities are: the forced air cooling system adopted by the A.V.M. (Atelier de Vitrification Marcoule) and the T.V.F.(Tokai Vitrification Facility), and the natural air cooling system adopted by the W.V.P.(Windscale Vitrification Plant) (1). And peculiarity IHI natural air cooling system is as follows: Each thimble tube is connected to the inner gas pumped down system, and the cavity at thimble tube is kept to negative pressure. The system can be designed to protect leakage from the vitrified waste containment boundary.

THE NATURAL AIR COOLING SYSTEM

The storage vault is shown in section in Fig. 3. The cooling air enters though a hood, protecting the wind and snow, and flows though a labyrinth and a louver which prevents radiation streaming and enters the storage area. The air flow of the storage area is as follows: The air in the annular gap between the thimble tubes and the ventilation pipes is heated

by the decay heat of the vitrified waste. Because of the density difference between the heated air and the open air, the draft force, due to the buoyancy, is generated in the annular gaps. And the cooling air takes decay heat from the vitrified waste, though the exit louver and labyrinth, up the concrete shaft and out to the atmosphere via a wind and snow hood. The flow rate of the cooling air increases as the heat generation rate of the vitrified waste becomes higher, and as the height of the annular gap and the air cooling outlet shaft increases.

Fig. 3

Estimating of the performance of natural convection air cooling system may be achieved by:

- thermo-hydraulic experimental testing about 1/3 scale models (2,3).
- development of thermo-hydraulic numerical analysis method (2).
- wind tunnel test to confirm the positive cooling effect of all directional wind conditions (3).

The thermo-hydraulic experimental apparatus is shown in Fig. 4. This basic model simulates three thimble tubes, and its scale is about 1/3. The heat generation rate of each thimble tube is controlled. The inlet flow velocity and the outlet air temperature of each annular tube is measured. And the experimental and the numerical results are shown in Fig. 5. Figure 5 shows the accuracy of the numerical method. This numerical method will applied to the facility design.

Fig. 4

Fig. 5

A low speed wind tunnel of 1/150 scale model simulates the storage facility and other buildings. The experimental parameter is in the shapes of the outlet shaft. The experimental result shows that the pressure difference of all the types is positive. These outlet shaft shapes increases the air flow rate.

Flow modeling of the vault module calculates both the general cooling air flow rate and particularly its distribution within the thimble tube. And the increase of the temperature inside the thimble tube has been carried out using the TAC-2D (4) code to calculate. The increase model of temperature inside the thimble tubes is shown in section is Fig. 6. And the calculation result shows that the decreasement of vitrified waste temperature by the storage years is Fig. 7 (5).

Fig. 6

Fig. 7

THE RADIATION SHIELDING

Most of the storage area are enclosed with concrete walls and floor, thus the radiation from vitrified waste is shielded in concrete. The natural air convection system have an inlet shaft and an outlet shaft for the cooling air. These shafts must be opened to the atmosphere, therefore, the both direct radiation and skyshine radiation, must be shielded to protect the environment. As the result of the shielding design, each shafts have louver and labyrinth. (See Fig. 3.) In the development of the vitrified waste storage technology, IHI compared the experimental data and calculation data (6). Figure 8 shows the experimental set up. An experiment was carried out to study the radiation distribution in a single bent duct and shielding effect of the louver. The single bent duct applied to the concrete duct for the ventilation system, and the louver applied to the inlet and outlet shaft for natural air convection system.

Fig. 8

THE POTENTIAL DROPPED ACCIDENT

The safety submission claimed that the single failure proof grab and hoist mechanism. The vitrified waste handling system is of sufficient integrity to render the probability of dropping the vitrified waste into the vault during loading operations, at nearly nil. To evaluate the potential consequences of dropping and possibly rupturing the vitrified waste; tests were run to an examination these consequences. It is assumed that the vitrified waste drops on the floor from the height of 9 meters, and drop on the previously stored vitrified waste in the thimble tube (3). The dropping velocity is an important parameter, and the calculated free dropping velocity from the height of 9 meters is about 13 m/sec, but the dropping velocity in the thimble tube is lower than that of a free drop because the narrow space between the dropping vitrified waste and thimble tube cause an air cushion effect.

The finite element grid is shown Fig. 9 and the numerical results are shown in Fig. 10 and 11.

Fig. 9

Fig. 10

Fig. 11

Figure 10 shows that the shoulder of the vitrified waste is deformed and Fig. 11 shows that the equivalent - plastic strain dose not exceed 20%. The numerical analysis result shows that the strain of canister does not exceed the tensile strain.

OTHERS

For design of the vitrified waste storage facility, following items are required.

The integrity during earthquake

Design criteria of the vitrified waste storage facility was established considering the special circumstances of Japan, such as limited site and strong earthquakes.

The concepts of operation

The vitrified waste is handled by remote control from a central control room. The central control room is equipped with three operational console and a watch foreman's console. Each operators console have a CRT units which is operated by a touch panel operation, and a ITV monitor.

DESIGN AND CONSTRUCTION

IHI designed and constructed the vitrified waste storage facility that is the first application of the natural air cooling system technology in Japan. The major specification of the JNFL's Facility are shown in Table II (5,7).

These appreciation for the Vitrified waste storage system will be applied to the design of the spent fuel storage facility. Development of the vitrified waste storage systems has been carried out IHI in order to obtain the design data and method of the spent fuel storage facility (8).

Table II

CONCLUSIONS

1) IHI has been designed and constructed two type of vault air cooling system for the vitrified waste storage facility. One is forced air cooling system, and the other is natural air cooling system.

2) Technology that carried out from the vitrified waste storage system development will be applied to the spent fuel storage system. And the concept design of vault air cooling system have been researched and developed.

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7-35

STUDIES ON PLUTONIUM ADSORPTION BY INORGANIC ADSORBENT

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ABSTRACT

Japan Atomic Energy Research Institute (JAERI) and Unitika Ltd. have jointly developed an inorganic adsorbent, which can remove plutonium (Pu) in reprocessing liquid waste and reduce a volume of the waste containing TRU by the incineration.

Previously We reported that this inorganic adsorbent had an excellent characteristics to adsorb Pu in the presence of 1 M nitric acid(1). We had done the flow-through column tests to determine practical conditions of the treatment process. From these results, this inorganic adsorbent was found to have good adsorption properties under the following conditions.

- adsorbent length / adsorbent diameter 3
- space velocity 1.6 h-1

The cold and hot incineration tests showed that the optimum conditions for the incineration of this fibrous adsorbent were that the temperature was 550C and the air flow rate was 30ml/min. We also found that the volume reduction factor was approximately 20 and no flying of adsorbed Pu was observed during the incineration process of the adsorbent.
This inorganic adsorbent has good characteristics to remove Pu directly without a pre-treatment of the liquid waste containing Pu, and reduce the volume of used adsorbent by the incineration. Therefore this process is very effective for the treatment of liquid waste.

INTRODUCTION

The radioactive liquid waste from reprocessing plants contains the fission products such as Cs, Sr and the TRU such as Pu and U. It is necessary to separate these products from the waste and to reduce the volume of the waste. The Coagulation-Sedimentation or the Ion-Exchange method had been conventionally used to treat the waste. However, these methods encountered the treatment of the secondary waste such as a sludge and the used resin itself, which shows only a small reduction of the volume by the incineration because of the flame retardant. The development of more effective adsorbent and treatment processes have been studied in order to overcome this shortcoming.
JAERI and Unitika had jointly developed an inorganic adsorbent, which is molded in the cartridge forms to be easily applied in the industrial fields. We made inorganic adsorbent by 80 wt% of fibrous adsorbent, that is, fibrous activated carbon which has the specific surface area of approximately 2,000 m2/g and 20 wt% of inorganic binder. The fibrous adsorbent was improved so as to be more hydrophilic by the oxidation without decrease of the specific surface area. As the fibrous adsorbent is consisted of more than 90% of carbon, it is physically and chemically stable, and easily converted to CO2 and H2O by the incineration. It is possible to expect useful applications form at the standpoint of the volume reduction of the waste.
In this paper, we report the Pu adsorption properties of the inorganic adsorbent and the incineration properties of this adsorbent.

EXPERIMENT

Adsorption Test

In the flow-through column test and equilibrium adsorption test, we used radioactive liquid waste as the test solution. The waste was spent-fuel generated from reprocessing at JRR-3 which is the first reactor in Japan. The properties of the liquid waste and test adsorbent are shown in Table I and II.
Table I
Table II

Equilibrium Adsorption Test

The immersion method was applied to the equilibrium adsorption test. The adsorbent was immersed in the test solution for 24 hours, and then the test solution was filtrated. Pu adsorption capacity of the adsorbent is obtained from the Pu concentration in the filtrate measured by a spectrometry.
The quantity of Pu adsorption is calculated as follows.
Eq. 1
Where :
QPu : Quantity of Pu adsorption (mg/g)
V : Quantity of solution (ml)
M : Quantity of adsorbent used (g)

C0 : Pu concentration of the test solution before adsorption (mg/ml)
C : Pu concentration of the test solution after adsorption (mg/ml)
The conditions of equilibrium adsorption test are the following:
Solution volume : 50 ml
Adsorbent weight : 0.1, 0.2, 0.5 g
Preservation time : 24 h
Flow-through Column Test
Figure 1 shows a diagram of the flow-through column test. We measured Pu concentration before and after the flow through of the adsorbent by a spectrometry.
Pu concentration ratio is calculated as follows:
Eq. 2
Where:
CPu :Pu concentration ratio
C0 :Pu concentration of the test solution before flow-through the adsorbent (mg/ml)
C :Pu concentration of the test solution after flow-through the adsorbent (mg/ml)
The conditions of flow-through column test are the following :
Adsorbent length / Adsorbent diameter : 6
Space velocity : 1.6 h-1
Adsorbent weight : 2.5 g
Fig.1
Fig. 2
Cold incineration Test of Inorganic Adsorption
Figure 2 shows a diagram of the incinerator used in this test. We used an inorganic adsorbent immersed in 1 M nitric acid in the cold incineration test.
The conditions of cold incineration test are the following:
Adsorbent weight : 1, 2, 3 g
Incineration temperature : 550, 600, 700C
Air flow rate : 5, 15, 30, 45 ml/min
Hot incineration Test of Inorganic Adsorption
We used an inorganic adsorbent immersed in test solution shown in Table I. We used an experimental apparatus similar to one shown in Fig. 2 in the hot incineration test of the inorganic adsorbent. We measured the radioactivity of the inlet, and the outlet glass-filter, at each point as numbered in Fig. 2 and the condensed water for the evaluation of the radioactive distribution.
The conditions of the hot incineration test are the following:
Adsorbent weight : 1.8 g
Incineration temperature : 550C
Air flow rate : 5, 15, 30 ml/min
RESULT
Equilibrium Adsorption Test
Figure 3 shows the adsorption isotherms of Pu at 20C. Table III shows the adsorption characteristics of b and g nuclide. The inorganic adsorbent molded in the cartridge forms was slightly less than the fibrous adsorbent in the quantity of Pu adsorption. The adsorption of b and g nuclide were not observed and it means that Pu was adsorbed selectively.
Table III
Fig. 3
Flow-through Column Test

Figure 4 shows a breakthrough curve obtained by the flow-through column test. The result shows that, the both adsorbents maintained a removing efficiency of approximately 97 % even when the volume of the flow-through liquid waste was 40 times that of the column.
Fig. 4
Cold incineration Test of Inorganic Adsorption
Figure 5 shows the effect of temperature on the incineration at the adsorbent weight of 2 g and the air flow rate of 30 ml/min. From this result, it was found that the incineration temperature did not affect on the rate of weight reduction of the inorganic adsorbent at the air flow rate of 30 ml/min, and also found that incineration was quite possible even at 550C (The ignition point of fibrous adsorbent is 480C).
Figure 6 shows the effect of the air flow rate at 550C. As the air flow rate increased, the rate of weight reduction became faster; however, over 30 ml/min, the rate of weight reduction became constant.
Figure 7 shows the effect of the adsorbent weight at the air flow of 30 ml/min and the temperature of 550C. The result shows that, when the adsorbent weight of 2 g and 3 g were used, the rate of weight reduction during the time elapsed of 1-5 hours coincided each other.
It was observed that the volume reduction factor was approximately 20 and no flying of particles and soot was observed with the incineration of this inorganic adsorbent.
Fig. 5
Fig. 6
Fig. 7
Hot incineration Test of Inorganic Adsorption
We evaluated the radioactivity distribution inside of a combustion pipe after the hot incineration test.
Table IV shows the radioactivity of glass-filters, Table V shows the radioactivity at each point as numbered in Fig. 2, and Table VI shows radioactivity of the condensed water which was generated at the incineration. From these results, we detected the radioactivity of Pu only in the residual ash but not inside of the pipe. This fact means that no flying of the nuclides was generated at the incineration.
Therefore, it is possible to incinerate the inorganic adsorbent at the air flow rate of 30 ml/min and temperature of 550C.
Table IV
Table V
Table VI
CONCLUSION
From the equilibrium adsorption test and the flow-through column test, we found that the inorganic adsorbent had an efficient adsorption properties on Pu. The incineration test of the inorganic adsorbent showed that volume reduction factor reaches approximately 20 and no flying of adsorbed Pu was observed.
These results suggest that the process using the inorganic adsorbent for the treatment of the radioactive waste is very effective from the standpoint of the volume reduction of the waste.
In the near future, this process will be applied for the treatment of stored liquid waste in JAERI.
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7-36

EVOLUTION OF THE LR56 TYPE B(U) CASK SYSTEM FOR THE TRANSPORTATION OF RADIOACTIVE LIQUID WASTE

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ABSTRACT

The French Atomic Energy Commission (CEA) initially developed the LR56 cask system for off-site radioactive liquid transportation. Two models of this IAEA, type B(U) certified cask have been in day-to-day use in France and other European countries since 1988. Recently, the U.S. DOE decided to purchase two LR56 systems, one each for use at the Hanford and Oak Ridge sites. Modifications have been made to the original design (such as the addition of an integrated spray rinsing system for removal of particulates) to comply with the needs of the DOE sites.

INTRODUCTION

Since the establishment of the French Atomic Energy Commission (CEA) Saclay nuclear center in the 1950's, liquid radioactive wastes from different facilities within the boundaries of the center have been transported to the site's liquid processing plants using specially designed tanker vehicles. As part of the development of nuclear activities in France, the shipment of radioactive liquids between different sites throughout the country became necessary. Based upon the experience acquired at the CEA nuclear centers, four models of tanker trailers, compliant with the French and European regulations for off-site transportation of radioactive liquids have been developed by the CEA. The main trailers, which are frequently used for transportation in France and also in other European countries, are the following:

The LR54 trailer is equipped with an unshielded tank with a usable capacity of 19.5 cubic meters. It is an industrial package designed for the transportation of low activity liquids (less than 10-5 x A2 per gram of specific activity, and 100 x A2 for the total activity, Ref. 1). Three units have been in service in France since 1985.

Note: A2 is an activity limit defined in Ref. 1 and 2 which depends upon the nature of each of the radionuclides (e.g. A2 limit is equal to 0.5 TBq for Csl37).

The LR56 system is a trailer equipped with a 50 mm equivalent lead shielded cask with a 4 cubic meter usable capacity. It is a type B (U) package (1, 2) designed for the transportation of medium to high level activity liquid. The cask model has obtained the certification as a type B(U) container by the French safety authorities. One unit has been in operation in France since 1988, a second unit since 1993.

The LR44 is a trailer equipped with a 150 mm equivalent lead shielded cask with 1.3 cubic meters usable capacity. It is a type B (U) package (1,2) designed to allow transportation of high level activity liquids using more shielding than for the LR56. The cask model has obtained the certification as a type B(U) container by the French safety authorities. One unit has been in operation in France since 1978. The LR40 is a trailer equipped with a 30 mm equivalent lead shielded cask with 2.8

cubic meters usable capacity. It is a type B (U) package (1) designed for the transportation of medium activity liquid. The cask model has obtained the certification as a type B(U) container by the French safety authorities. One unit has been in operation in France since 1980.

The present paper focuses on the LR56 liquid transportation system, which was recently selected by the Hanford and the Oak Ridge U.S. Department of Energy (DOE) sites for transportation of radioactive liquids. The paper presents the new applications and provides technical details regarding the modifications that have been made to the system in order to comply with the type of waste to be transferred.

THE BASIC LR56 SYSTEM

The LR56 assembly (see general outline of the LR56 system presented in Fig. 1) is composed of the liquid transportation cask itself mounted on a road trailer with additional peripheral equipment. Liquid transfer operations can be performed and monitored without external assistance using trailer mounted ancillary systems such as: a hydraulic arm to handle the cask plugs, a nitrogen supply station to purge the tank from eventual hydrogen build up due to radiolysis, an air filtration and vacuum/compressed air production cabin and utilities, and a control and monitoring cabin. Vacuum/compressed air and pressurized oil to operate the crane are supplied through pumps actuated by an on-board combustion engine. Only the cask itself is type B(U) certified; it is sealed and disconnected from all the peripheral equipment in the transport configuration.

Fig. 1

Cask

The cask consists of a horizontal cylindrical confinement tank surrounded by several layers of containment, mechanical, biological and thermal protections. The final external dimensions of the cask are 3.7 meters long by 2.15 meters in diameter. The currently approved maximum weight is 23,100 kg.

Confinement Vessel

The confinement vessel seals the waste from the environment. It is made of stainless steel type Z1 NCDU 25.20(or B 625 UNS 08-404 under ASTM code), highly polyvalent steel for use with aggressive chemicals, in particular phosphoric, sulfuric and hydrochloric acids. The vessel is cylindrical with plate thickness of 8 to 10 mm and rounded heads. The interior of the vessel has two baffle plates and supports for level indicators, transfer pipes and sampling pipes. Operating equipment such as valves, level measurement sensors and samplers enter through three wells on the upper longitudinal axis of the vessel. The wells are closed with bolted lids equipped with double O-ring seals which ensure the continuity of the first confinement envelope. The pressure between the two seals is monitored to check for tightness before and after transportation.

The following equipment is flanged to the well's base plates and is removable for maintenance:

Rear Well: Two special self-sealing stainless steel connectors are used to connect the loading or the unloading hoses to the tank. These connectors are derived from a standard design used for commercial airline refueling and provide a high degree of safety with minimum retention of contaminated liquids. Two separate mechanical devices ensure the latching of the connector on its base and the opening of the internal sealing valves. The two actions are mechanically interlocked. The connector bases

are mounted on hand valves to isolate the tank content during transportation. A liquid sampling connection is connected to a liquid sampling tube ending near the bottom of the tank.

Central Well: The central well is equipped with the level measurement sensors: a float driven very high level switch, a low level float driven switch and a float driven continuous level measurement system activating a resistive line.

Front well: The front well (with respect to the direction of motion of the trailer) is equipped with the cask venting connection, the hydrogen gas sampling connection and a redundant very high level float driven sensor

Containment Vessel

The containment vessel is made of a 6 mm 316L stainless steel, offset 20 mm from the confinement vessel by spacer tubes. It protects the other components from any liquid spills. A drain tube is placed in the intervessel space to check for liquid presence and drain it.

Biological Shield

35 mm of soft lead is placed around the containment vessel. The continuity of the biological shield around the access wells is ensured by two means. First, 35 mm plates of lead are cast in stainless steel crows welded inside the confinement vessel around the well base plates. Second, removable lead bricks are bolted on the top of the base plates and placed around the penetrations. The mechanical protection envelope (described below) provides additional biological shielding leading to a total of 50 mm lead equivalent shielding.

Mechanical Protection

The lead shield is surrounded by a thick carbon steel vessel (30 mm A42FP) providing the mechanical protection of the cask in the drop test conditions. Continuity of the mechanical envelope at the access well openings is provided by three heavy plugs. These plugs also ensure the closure of the secondary containment vessel.

Thermal Protection

A non-centered shell of 150 mm to 550 mm balsa and redwood provides thermal protection of the tank against external fires. It also provides for mechanical shock absorption in case of external impacts or drop. The continuity of the wood envelope is also ensured inside the removable plugs.

External Envelope

A 6 mm painted carbon steel layer provides final weather protection of the assembly.

Cask Securement

The cask lies horizontally on cradles which are welded to the trailer frame. The cask is immobilized by shackles bolted onto the cradles and lengthwise by stops built into the front of the trailer and rear of the rear cradle.

Trailer

The total length of the French version of the trailer is 9.37 meters (31 feet), width is 2.45 meters and height 4 meters. The fully loaded gross weight is 32 metric tons without the tractor

IAEA Certification Tests

Tests have been performed on a 1/3 scale mock-up demonstrating the tightness of the cask after the regulatory 9 meters drop test and 1 meter penetration drop test on a 15 cm rigid rod. Behavior under the fire,

immersion and other regulatory conditions has been demonstrated by calculations.

Main Characteristics of the Content

The reference cask model has been certified for the transportation of liquids with the following general specifications:

Activity limits:

The IAEA regulatory limit (1) imposes a maximum of 0.1 mSv/hour radiation field at 1 meter from the cask, which leads to the following activity limits:

< 10.7 x 1011 becquerels/cubic meter (29 Ci/m3) for gamma emitters at less than 0.8 MeV

< 8.14 x 1010 becquerels/cubic meter (2.2 Ci/m3) for gamma emitters at less than 1.3 MeV

If the total activity is larger than 3 x 103 A2 or 1.11 x 103 TBq (3 x 104 Ci), special notice must be made to the French authorities.

Maximum concentrations:

nitric acid: 30%, 5.5N

sulfuric acid: 50/O, 7N

CL- anions: 1 g/l

F- anions: 0.1 g/l

Maximum plutonium content: 200 g

Thermal power

10 W (derived from the fissile content)

Maximum operating effective pressure:

3 bar (3 x 105 Pa), the confinement vessel is pressure tested at 4.5 bar and helium leak tested.

HANFORD SITE APPLICATION

The US DOE Hanford site has operated since 1943 with the primary purpose of plutonium production (3). The liquid waste from the plutonium chemical separation process has been stored in large underground tanks (the "tank farms") for future retrieval, treatment and final disposal. The forms of waste include various mixtures of sludge, liquid, salt cake, or slurry. The waste is alkaline; principal radionuclides include uranium, uranium fission products and fission decay products. Approximately 46 single-shell tanks at the Hanford site contain interstitial and supernatant liquids which may leak from the tanks, due to the age of the tanks. Typically, pumping via pipeline to other tanks is utilized whenever such leakage is detected. However, various conditions may preclude such transfers taking place, thus the need for a flexible mobile above-ground auxiliary transfer system arose. Such a system must meet DOE approved on-site packaging standards for type B quantities of high-level radioactive materials.

At the present time, no domestic DOE or NRC type B packaging with the appropriate level of shielding is available in the United States for transport of radioactive liquids in bulk volumes (i.e. greater than 1 liter volume). It was decided that the best way to meet the tank farms' needs was to select an existing and proven type B packaging, the LR 56 cask system used in France.

This system is currently being purchased by Westinghouse Hanford Company (WHC) from Numatec, Inc. (a Cogema Inc./SGN company) under license from the CEA. The system was delivered to the site in late 1995.

OAK RIDGE SITE APPLICATION

The US DOE Oak Ridge National Laboratory (ORNL) is in the process of upgrading the liquid low-level radioactive waste-generator collection

facilities, transfer systems, treatment facilities, and interim storage facilities on site to meet regulatory requirements. Existing single-wall tanks are either being replaced with doubly-contained tanks or removed from service. Single-wall piping systems are either being replaced with doubly-contained piping systems or are being removed from service and the radioactive waste is being transported to the central waste collection system.

The current transportation of liquid radioactive waste at ORNL uses a tanker truck which is not shielded and does not meet the requirements of the US Department of Transportation (DOT). In addition to transporting waste from the waste generators on the ORNL site, a DOT Type A package transporter will be capable of transporting liquid radioactive waste via public highways from other local US DOE sites not contiguous with the ORNL.

In 1992, the operator of the US DOE Oak Ridge National Laboratory, Martin Marietta Energy Systems (MMES), initiated an evaluation of potential suppliers of a liquid low-level radioactive waste transporter with a capacity of approximately 1000 gallons. MMES action based upon this initial evaluation was deferred due to funding priorities. In early 1994, funding was made available and the evaluation was re-initiated. Independent of the WHC assessment of available liquid radioactive waste packages, MMES determined that the procurement of an existing Type B package (the LR56) was advantageous because of the time and expense involved in initial design and testing of a DOT Type A container. Subsequently, MMES entered into a teaming agreement with WHC (which had by this time initiated procurement activities with Numatec, Inc) for WHC to perform the procurement activities for a second unit to be supplied for the Oak Ridge site. Under such a teaming arrangement, the two US DOE site operators were able to share resources in reviewing design modifications, performing witness points, and contract administration. This second system purchased by WHC will be delivered to the Oak Ridge site in early 1996.

EVOLUTION OF THE LR56 BASIC DESIGN

In order to meet the Hanford and Oak Ridge sites' requirements, some modifications of the basic LR56 system have been made. The main modification is the addition of a rinsing system inside the compartments of the confinement vessel to handle the presence of sludge and insoluble particles in the liquids to be transferred. The modifications are described hereafter; the Oak Ridge version contains only the ones marked by an asterisk.

Rinsing/Spray-wash (*)

Each compartment of the confinement vessel is equipped with one or two rotary spray wash nozzles to facilitate the wash down of solids towards the sump. Fixed nozzles are also installed to improve the cleaning of critical areas such as the level measurement floats. The water supply pressure is 60 to 90 psig. To allow for replacement of the spray nozzles, the base plates of the wells have been made removable with the penetrations welded on the flange.

This modification does not affect the confinement barrier (closed at the lid level) and provides additional flexibility to the system. Moreover, the removable lead brick protections initially placed in the wells have been replaced by permanent shielding plates cast in additional welded enclosures placed inside the confinement vessel. This clears the space around the penetrations and simplifies the maintenance operations.

Temperature Sensors

In order to provide for the ability to monitor the temperature inside the liquid and the sludge at various levels, five thermocouples have been installed in the tank. One thermocouple is placed in a guide tube in the intervessel space between the confinement and the containment vessel to measure the bottom tank temperature. Four additional sensors are placed in a tube fixed on a well base plate to measure temperature at various levels. The upper level sensor positioning corresponds to the maximum high level of liquid in the tank.

Redundant Level Measurement

A capacitive continuous level measurement probe is installed in the central well. It is not directly used in the control logic but provides redundant information using a different technology than the primary system, using a float driven technology.

Annulus Leak Detection

In order to permanently check for the presence of liquid in the intervessel space, a resistive probe is placed in a guide tube down to the bottom sump. The information is sent back to the monitoring station and processed in the safety logic of the automatic controller.

Neutron Measurement

A 15 mm inner diameter tube is placed in the lower protective wood layer, in contact with the carbon steel mechanical protection envelope. It is intended to be used as a guide for the insertion of a neutron detection foil in order to detect the presence of any sedimented fissile material.

ASME Filtering (*)

The cask vented air filtering station has been redesigned to comply with the ASME 509 and 510 requirements. The HEPA and active coal filter efficiencies are now testable in situ. The new design also allows in-situ replacement of the filters under bag transfer without dismounting the filter housing.

Electrical Motors (*)

The on-board combustion engine (e.g., used for the vacuum/pressure pump and the pressurized oil for the crane) has been replaced by electric motors supplied by an external source of 460 V AC current. The system is less autonomous than the French version but gains in flexibility, decreased pollution for indoor operations and noise.

PLC Control System (*)

The control system, initially relay driven, has been replaced by a Programmable Logic Controller (PLC). New modes of operation have been added to take into account the rinsing operations and the various loading and unloading configurations which are specific to the U.S. applications. The PLC allows the exchange of monitoring and permissive signals with the external loading/unloading stations in order to interlock the transfer operations. A remote control console is provided with the Hanford version allowing it to operate the system from a shielded area. The remote console is identical to the trailer console and is linked to the on-board PLC via a standard bus line.

US DOT Trailer (*)

The cask and ancillary equipment are mounted on a new trailer conforming to U.S. Department of Transportation (DOT) requirements. The total length of the trailer is 41 feet, which is significantly larger than the French version, due to differences between the European and U.S. road and highway transportation regulations.

Type B(U) Certifiability (*)

The Type B performance requirements of the French regulations are virtually equivalent to those of the United States, as they are ultimately based on the International Atomic Energy Agency (IAEA) Safety Series 6 (1) requirements used by both countries. However, an on-site SARP is still necessary because the French Certificate of Compliance for the LR56 is not recognized for domestic use in the United States. For off-site uses as a type B package, a licensing process will be pursued through the DOE or NRC, but such a license is not necessary for on-site use. The modifications made on the cask and the ancillaries have been designed in order not to affect the ability of the system to meet Type B(U) performance criteria. Analyses and calculations have been made by SGN in order to support this demonstration.

DOT 7A Compatibility (*)

To address the potential use on public access roads and highways between the ORNL site and nearby DOE facilities, an analysis of the LR56 system's compliance with the U.S. DOT 7A criteria has been conducted. As the system is already Type B certified in France, it easily meets the type A criteria. This has been confirmed by a systematic analysis. The volume of liquid waste to be transported under this certification will be operationally limited to meet the US DOT A2 quantity limits.

CONCLUSIONS

The LR56 system is a unique, well proven, and flexible Type B(U) liquid waste transportation cask. The equipment benefits from all the experience gained in the design and operation of liquid transportation casks in the French nuclear community, since the early 1950's (LR56 stands for "56's design of Radioactive Liquid Cask"). Two casks are presently in day-to-day use by the CEA. Since they were put into service, the LR56 systems have transported about 80 TBq (2100 CD of active liquids over a total distance of 35,000 km without any significant incident involving the hazardous material transported.

The two new systems that will soon be brought into operation on the U.S. DOE Hanford and Oak Ridge sites present new features which are broadening the spectrum of the LR56's applications.

ACKNOWLEDGMENTS

This paper presents work done by Numatec, Inc. under contract from Westinghouse Hanford Company. Numatec, Inc. (a Cogema/SGN U.S. company) has subcontracted to SGN, the nuclear engineering company in France. SGN is leading the engineering design, fabrication and commissioning of the system, working with SDMS, the main subcontractor already involved in the previous casks fabricated for the CEA. The authors also acknowledge Magali Ranchoux, Jean Michel Troude and Jean Louis Saulze of CEA liquid transportation departments, Guy Cordero of STMI and Didier Mercier of SDMS for their support in the development of the U.S. versions of the LR56 cask system.

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DEMONSTRATION OF A DRUM VENTING SYSTEM (DVS) FOR TRU WASTE DRUMS

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ABSTRACT

Transuranic (TRU) wastes generated in Department of Energy (DOE) operations have, since 1970, been packaged to a large extent in unvented 55-gallon steel drums and have been stored with the intention of future retrieval. Today, there are safety concerns regarding these drums because of the potential presence of hydrogen and methane generated by radiolysis of hydrogenous wastes, and/or of traces of combustible Volatile Organic Compounds (VOCs) contained in the original waste materials. Any future handling and transportation of these waste packages must address these concerns. To provide a method for penetrating the drum and drum liner lids, sampling and analyzing the headspace gases, purging of undesirable gas mixtures, and installing an approved High Efficiency Particulate Air (HEPA) filter vent, a Drum Venting System (DVS) has been designed, built, and performance tested. The DVS has been shown to be capable of safely and effectively meeting all performance requirements in the venting of previously filled drums of TRU waste.

INTRODUCTION

Transuranic (TRU) wastes generated in U.S. Department of Energy (DOE) operations have, since 1970, been packaged to a large extent in unvented 55-gallon steel drums and have been stored with the intention of future retrieval. These wastes are intended to be disposed of permanently in the DOE Waste Isolation Pilot Plant (WIPP) facility. Today, there are safety concerns regarding these stored drums because of the potential presence of combustible headspace gases. Such gases can include hydrogen and methane resulting from the radiolytic decomposition of hydrogenous waste materials (e.g., paper, plastics, moisture) and/or from the presence of generally small amounts of combustible Volatile Organic Compounds (VOCs) that are cocontaminants of the TRU wastes. Any future handling and transportation of these waste packages must address this concern. Additionally, the WIPP Waste Acceptance Criteria require all packages to be vented, and that all such packages shipped to WIPP in the DOE TRUPACT-II Type B container must be both vented and shown to meet combustible gas concentration limits. To provide a safe method for penetrating the drum lid and, if present, inner plastic liner lid, sampling and analyzing the headspace gases,and installing an approved filter vent, NFT Incorporated (NFT) has developed a Drum Venting System (DVS) for the remote performance of these functions. This paper discusses the design, operation, and performance testing of the DVS.

SYSTEM DESIGN

The NFT DVS, as depicted in Fig. 1 and 2, is a stand-alone, skid-mounted system, requiring only external electrical power. The DVS is capable of properly installing an approved filter vent in a previously filled drum of TRU waste. In this process, a sample of headspace gas is collected and analyzed. In the event of a flammable headspace gas mixture being detected, the system can purge the gases present. The DVS is comprised of

several components and subassemblies, including the powerhead subassembly, glovebox, drum cabinet, gas analysis system, air filtration system, and remote controller assembly. Each of these is described below.
Powerhead Subassembly. The electrically powered powerhead subassembly is the key mechanical subassembly within the DVS. It contains a structural frame that supports and stabilizes a linear drive and a nutrunner.
Linear Drive. A stepper motor-driven, screw type linear drive controls the vertical movements of the powerhead assembly. The DVS employs a linear drive controller which precisely controls the movements of the powerhead assembly. The controller can store up to 100 unique and predetermined linear motion sequences or programs.
Nutrunner. The DC-motor driven nutrunner is mounted to the powerhead frame and provides the rotary motion to a socket attachment that holds the head of the filter assembly during the piercing and insertion operations. This nutrunner is a variable-speed unit capable of operating at higher speeds when piercing the lid and slower speeds when installing the filter. The nutrunner functions as the torque source which enables the filter threads to cut into the drum lid such that the filter seats securely in the drum lid. The nutrunner controller is a microprocessor-based, programmable, rotary motion controller. The controller can store up to 64 unique and predetermined rotary motion sequences or programs.
Glovebox. The glovebox that contains the Powerhead Assembly was designed and fabricated in accordance with the American Glovebox Society's "Guidelines for Gloveboxes", AGS-G001-1994. An inlet filter housing contains one 8x8 inch gasket seal High Efficiency Particulate Air (HEPA) filter element. A small airlock (approximately 10x10x10 inch) is provided to permit entry of drum filters and other small components, as needed.
All electrical components within the DVS glovebox comply with National Electrical Code standards in that they are intrinsically safe (nonsparking), or are of a "purge" design which prevents ignitable gases from entering their interior spaces where electrical discharges may be present. For example, the linear drive motor has a constant flow of air to prohibit any contact with ignitable gases which could possibly be present in the glovebox. The nutrunner, on the other hand, is nonsparking in design.
Fig. 1
Fig. 2
Drum Cabinet. The drum cabinet is a rectangular heavy steel box structure designed to enclose a 55-gallon waste drum, or an 83-gallon overpack drum, and of sufficient size as to dissipate the pressure from a worst case headspace gas ignition (drum pressurization to 136 psig) to <15 psig. The glovebox is attached to the top of the cabinet, with a common mounting plate used to seal the top of the cabinet and bottom of the glovebox. Following fabrication, the cabinet was successfully pressure tested to 22.5 psig.
The cabinet design includes a 24x24 inch fluid seal HEPA filter housing for inlet air. A backflow prevention device is present in the cabinet to minimize the probability of filter failure in the case of overpressurization. Two doors are present. One is a small access door used primarily to inspect the top surface of the drum for contaminants prior to opening the large, drum access door. Both doors are gasketed to provide a tight seal during both normal operating conditions and abnormal events. Both doors have sensors to tell the operator of their open or closed status.

The drum lift and weighing subassembly is contained within the cabinet. The major components include a pneumatically - operated air-lift, (air stroke actuator) and force transducers. Two force transducers (weight load cells) are used with the lifting mechanism to provide downward force information. Following placement of the drum in the cabinet, the lifting device is raised slightly to permit weighing of the total mass lifted, from which drum weight is determined. The drum then is lifted to seal against the drum lid seal, thus isolating the cabinet from the above glovebox. By adding a predetermined sealing force to the total weight, the total upward lift force required for this operation is determined.

Gas Analysis System. The headspace gas analysis system utilizes a Varian Model 3400 CX gas chromatograph (GC) tuned and calibrated to quantify hydrogen, methane, and a selection of VOCs. The dual column system uses a Flame Ionization Detector (FID) to detect the VOCs, and a Thermal Conductivity Detector (TCD) to detect hydrogen and methane. The system collects a real-time sample of drum headspace gas, and within approximately 5 minutes, prints out the gas concentration results. A personal computer (PC) software package fully automates the analysis and quantitation of the headspace gas results. The system alarms if any gas concentration exceeds the preset levels.

Air Filtration System. The air filtration system is responsible for ensuring that any radioactive contaminants released from a waste drum by any mechanism within the DVS are fully contained within the system. During operation, constant negative differential pressure is maintained within the drum containment cabinet and glovebox. The exhaust fan is an induced draft type, which is rated at 1,000 CFM maximum. The fan is equipped with a variable speed controller to permit the adjustment air velocity, and hence the pressure differential within the cabinet and glovebox. Parallel ductwork connects the outlets of the glovebox and cabinet to the common filter train. In the outlet from the glovebox, an adjustable damper is present to allow adjustment of the airflow. Adjacent to the damper is a backflow device that prevents overpressure within the cabinet from backing up into the glovebox. The filter train consists of one roughing filter and two HEPA grade filters, all in series.

Remote Controller Assembly. An Industrial Process Controller (IPC) is the key interface between the machine and the operating personnel. The IPC is connected to a special touch-screen interface that allows the operator to control the course of action during the drum-venting process. For example, the linear drive controller and the nutrunner controller are electronically connected to the IPC, which issues discrete logic signals that cause the execution of predetermined motion sequences or programs. The IPC software prompts the operator for input via the touch-screen user interface. The resident software provides a flowchart-oriented operational basis for operation of the system (an operation is completed before the next operation in the sequence is begun). The entire control and data station is enclosed in an electrical cabinet affording both personnel and equipment protection during the operation.

DRUM FILTER ASSEMBLY

The filter used normally in the drum-venting operation is pictured in Fig. 3. A short-stem version, used in the event an impenetrable object is encountered, is also shown in Fig. 3. The normal filter has a self-boring, self-tapping stem. The drill-type tip of this filter is made of hardened tool steel (high-speed steel). The housings of both units, which are otherwise identical, are fabricated of 316 stainless steel. The

filter media is a carbon-bonded-carbon material and is performance tested to provide >99.97% removal of 0.3m to 0.7m particles. The air delivery capacity of the filter is 200 ml/minute at 1 inch of water column.

OPERATIONAL SEQUENCE

The following is a description of the operational sequence upon which the DVS design and programming of IPC software is based.

Power Up. All DVS subsystems operate on 125 VAC power, except the air handling train which is 240 VAC, three phase. System power is initiated manually.

Fig. 3

Change Operating Parameters. After powering up, the IPC displays currently set system process parameters which a qualified operator, by means of a password, can adjust if needed based upon prior knowledge of the drums to be processed.

Insert Filter In Socket. In preparation for DVS operation, the operator manually, through the glovebox gloves, places a filter into the specially designed, close-tolerance fitting, slotted round-head recess in the socket.

Drum Load. The operator opens the drum cabinet door and places the bridge platform in the recess to secure the door in the fully opened position. This allows the drum and dolly to be loaded into the cabinet and prevents the door from closing prematurely. The operator then removes the bridge platform and secures the door.

System Initialization. The IPC directs the HEPA blower to start up and reach its setpoint speed.

Drum Weigh. The IPC initiates a flow of pressurized air into the air stroke actuator. As the actuator slowly inflates, the drum is lifted until the beam of a photoelectric sensor is broken. This indicates the drum is sufficiently raised to be weighed by the force transducers.

Drum Lift. Upon receiving a signal from the operator, the IPC reinitiates the flow of air to the actuator which lifts the drum further to where the lid is pressed against the molded rubber seal of the seal assembly in the cabinet ceiling with the appropriate lift force, as measured by the force transducers.

Form Vacuum Chamber. Upon achieving the appropriate lift force, the operator initiates a signal causing the powerhead socket to descend into the seal assembly cavity. As this occurs, a vacuum pump begins to evacuate the now sealed chamber.

Seal Test. With the chamber formed and evacuated, the DVS tests the integrity of the chamber by monitoring any in-leakage of air. If the reduced pressure in the chamber is seen to increase by less than a predetermined amount over a short period of time, an adequate chamber seal has been achieved. If an adequate chamber seal is not achieved, the operator can either retry to form the chamber or proceed to the next step, noting that the headspace gas analysis results may be compromised by outside air.

Bore to Sample Depth. Upon receiving a signal to proceed, the nutrunner is caused to descend and rotate at "boring speed." In this operation the drum lid first is penetrated, and subsequently the liner lid is penetrated. However, the self-tapping threads of the filter assembly have not yet engaged with the drum lid. At this point, all downward and rotational motion of the filter ceases. As soon as the drum lid is penetrated, headspace gas fills the small evacuated chamber created above.

Sample Analysis. The hollow stem and access stem holes of the filter allow flow of headspace gases into the sealed chamber. The operator now initiates the gas sampling and analysis operation of the GC. Here, headspace gas is slowly pumped from the drum, through the small chamber, to the GC. Upon completion of the gas sampling and analysis, the PC displays the analytical results.

Headspace Gas Purge. When an undesirable concentration of combustible gas is measured, an evacuate/purge cycle is initiated. Here, the drum headspace gas is partially evacuated, followed by refilling with dry nitrogen. This cycle can be repeated as needed to reduce the combustible gas concentration. Subsequently, the headspace gas is resampled and analyzed for verification that the drum is safe to handle.

Install Filter. Following completion of an analysis showing acceptable headspace gas concentrations, the operator initiates the filter installation process. The IPC synchronizes the nutrunner controller and the linear drive controller to operate at the optimum thread speed and feed (descent) rate, based on the filter's thread characteristics. Using the nutrunner's torque transducer output, the IPC slowly lowers the powerhead socket until the torque transducer output indicates the threading operation is in progress. Rotary and downward motion continues until the torque transducer output achieves the predetermined setting of 18 ft-lbs, thus indicating that the filter is properly installed and seated.

Impenetrable Object. If an impenetrable object is detected at any point in the boring or filter installation process, such an indication will immediately stop all powerhead motion, with the IPC indicating the status to the operator. Should this occur during the boring operation, the subsequent gas sampling/analysis and evacuate/purge (if needed) cycles will be completed before any corrective action. Following the operational stoppage (and the continuation of operations, as appropriate), the operator can direct the powerhead assembly to retract the filter into the glovebox. At this point, the long-stem filter is replaced with a short-stem filter (see Fig. 3), which then is installed in the normal manner.

Lower and Unload Drum. Following completion of the filter installation, the operator lowers the drum in the drum cabinet. At this point, the operator opens the small access door and can check the drum lid near the filter for radioactive contamination. If any is found, the drum lid can be wiped clean to a level acceptable for removing the drum from the cabinet. The drum then can be removed from the cabinet through a reversal of the "drum load" step, and another drum subsequently can be loaded.

PERFORMANCE TESTING AND RESULTS

The DVS Test Plan was developed and accomplished to demonstrate the overall abilities of the system. Specific tests included the following:

- Ability to penetrate a "normal" drum lid and install a filter assembly. Testing included leak tightness and torque verification.
- Testing as above on drum lids containing "irregularities" such as a rusted surface or minor denting.
- Ability for system detection and recovery from an impenetrable object.
- Ability to purge the headspace volume with an inert gas.
- Ability to measure the weight of a drum, determine the required lifting force, and apply that force.
- Ability to perform representative headspace gas sample acquisition and analysis.

Normal Drum Lid and Liner Penetration/Filter to Drum Lid Seal

Testing was performed by installing filter assemblies, in the normal operational sequence, into lids of 55-gallon drums. Each drum contained a 90-mil high density polyethylene (HDPE) liner. Following the installation, each assembly was visually inspected to verify the required penetration of both the drum and liner lids, and the apparent proper installation of the filter. To verify a leak tight seal between the filter housing and drum lids, each assembly was tested utilizing Dioctyl Phthalate (DOP). Here, a dual chamber assembly was used to seal both sides of the lid around the filter for application of the DOP testing to ensure that DOP particle leakage through the assembly met testing requirements for HEPA grade. A calibrated torque wrench was used to measure the torque for the filter installations.

A total of thirty five HEPA grade filter vents were installed by the DVS in full operational mode into "normal" drum lids. Subsequently, each was visually inspected and shown to have properly penetrated both the drum and liner lids. Of these, 34 passed the DOP HEPA testing standard of a minimum 200 ml/minute flow rate and less than 0.030% penetration of DOP aerosol at one inch of water pressure. The one assembly not passing was found to be properly installed, and had DOP penetration of 0.6%. Disassembly and examination of the gasket indicated that a few cutting chips had not been properly blown out of the way and had lodged under the gasket. The installation torque measured in these tests was 18 + 0.4 ft-lbs.

Irregular Drum Lid and Liner Penetration/Filter to Drum Lid Seal

Testing was conducted exactly as described above. Here, a total of eleven filter vents were installed. Of these, ten subsequently passed all tests, with the self-tapping threads of one filter "stripping out" the hole in the drum lid. Subsequent investigation showed the lid area in which the filter was installed to be "slick", resulting from a possible oil-like residue. Stripout in this one instance occurred because of the reduced friction between the neoprene gasket and drum lid, thus not permitting the DVS to achieve the target torque of 18 ft-lbs. Threading in this case continued until the drum lid sheet metal stripped out. As a result of this testing, it was recommended that (1) drum lids be inspected to ensure that wet, oily, or otherwise slick surfaces not be chosen for filter insertion locations, and (2) all installed filters should be secondarily sealed with a fillet of silicon adhesive between the drum lid and filter head side wall.

Detection and Recovery from Impenetrable Object

In this test, a metal impenetrable obstruction was placed within 1/2 inch of the drum lid. The resistance measured on the powerhead controls was the measure of the force exerted on the object. Here, the DVS attempted to install a filter in the drum lid, where the lid was positioned such that the filter would encounter the steel object below the lid. Following the encounter with the "impenetrable object", the DVS would need to recover the original long-stem filter and install a short-stem filter in the hole in the drum lid.

In each replication of the test, the installation of the long-stem filter was stopped automatically by the excessive force caused by encountering the impenetrable object. The operator then was able to sequence the DVS operation through the subsequent retrieval of the long-stem filter, replace this filter with a short-stem filter, and properly install the short-stem filter in the drum lid.

System Purge

In this test, the DVS programming was tested to demonstrate the ability to remove and replace an undesirable headspace gas mixture. Following the penetration of a drum lid and sample analysis, the system demonstrated the ability to evacuate the drum headspace repeatedly to 11 psia followed by the purge cycle of refilling the drum headspace with nitrogen back to normal atmospheric pressure (14.7 psia)

Weigh Drum/Determine Required Lifting Force

Six drums of differing weights were tested, with a "seal force" of 200 lbs set for the test. Total measured lift force (drum weight plus seal force) varied by a maximum of + 3 lbs from the expected calculated value in each case, thus indicating an acceptable air-tight seal was achieved between the drum lid and seal gasket.

Headspace Gas Sample Acquisition and Analysis

Testing in this case was dependent on the ability of the system to establish and maintain a good seal between the seal plate gasket and drum lid (see above test), and on the proper calibration, function, and complete variance analysis of the GC sampling system. In this test, a sample drum was placed in the DVS and was lifted to seal against the seal gasket. The proper seal lift force first was verified. Following the lowering of the socket assembly to form the vacuum chamber, a vacuum was drawn and the chamber then isolated for one minute. The vacuum pressure decay within the chamber then was measured to verify the establishment of acceptable seals. Finally, the drum lid used in these tests had a reservoir attached containing a sample gas. One sample of this gas mixture was withdrawn using an injection syringe for direct injection into the GC. Subsequently, a filter assembly penetrated the drum lid in a normal DVS operation, thus allowing a simulated headspace gas sample to be obtained and analyzed. All analyses were repeated three times.

In the testing accomplished, twenty replications of measurements of total lift force showed a variation of + 2.3%. Vacuum pressure decay during the one minute hold time was negligible, well within the 5% reduction acceptance criteria, in all cases. Test acceptance criteria for the analysis of hydrogen, methane, and total combustible VOCs in these tests was to have no greater than a 25% deviation from the injected constituents of the calibration gases. The calculated Relative Percent Differences of the mean of the simulated headspace gas concentrations versus the mean of the actual (injected) concentrations for hydrogen, methane, and VOCs were <5%, < 5%, and < 10%, respectively.

SUMMARY

Trial and performance testing operations of the DVS have shown a normal cycle time of approximately 25 minutes/drum, with an increase of this time to approximately 45 minutes/drum if the evacuate/purge and resampling steps are mandated. The extensive testing of the DVS has demonstrated the accomplishment of all system performance objectives, including:

- Safe and efficient filter installation in a manner to ensure leak tightness of the filter seal to the drum lid;
- Reproducible and representative sampling of headspace gases; and
- Effective purging of unsafe levels of headspace gases, when necessary.

All other operational objectives of the DVS also have been demonstrated. The NFT DVS has been shown to provide a safe and effective method for venting previously filled drums of TRU waste.

USE OF GRAFTED MACROCYCLES IN THE TREATMENT OF ALPHA CONTAMINATED LIQUID WASTES

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ABSTRACT

The high selectivity and strong coordination of polyazacycloalkanes with transition-metal ions have been well established and make these ligands ideal candidates for use as reagents in performing metal ion separations. One way to achieve this aim is the covalent bonding of these macrocyclic ligands to a solid support. Indeed, during the last decades, numerous researches have been devoted to the surface modification of various supports used in dioxygen transport, waste-water treatment, metal-ion chromatography or recovery of trace-metal ions. The solid support can be a synthetic organic polymer or an inorganic material. Silica gel has been selected in most applications. Macrocyclic ligand attachments to silica are relatively simple reactions, especially when compared to immobilizations involving organic polymers.

Our primary interest has been the study of free tetraazamacrocycles in the complexation of lanthanide- and actinide-ions and the X-ray determination of the configuration of their corresponding complexes. Obviously, this type of ligands offers new potential due in this field to the high stability of the formed complexes.

Our secondary interest has been the preparation of modified silica gel as a polymeric support for alpha waste-water treatment. The use of these macrocycles grafted to silica showed clearly the high efficiency of these materials to extract actinide traces contained in alpha liquid wastes.

INTRODUCTION

The "Centre du Commissariat l'Energie Atomique de Valduc" has launched for several years, a research program on the development of new macrocyclic ligands which are able to pull out actinides from industrial liquid wastes.

The continued interest and research for new macrocyclic ligands stems mainly from their use as models for protein-metal binding sites in biological systems 1), as selective complexants 2) of metallic ions, i.e therapeutic reagents for the treatment of the metal intoxication 3), radioactive waste water treatment 4), functional groups for chelating ion-exchange materials 5) and to study host-guest interactions. Tetraazamacrocycle ligands represent a class of macrocycles known for their ability to complex transition and non transition metal cations. The macrocyclic complexes of lanthanides are now currently used for other medical applications 6), radioimmunotherapy, contrast-enhancing agents in Magnetic Resonance Imaging (MRI), as NMR shift and relaxation reagents and in many other novel clinical techniques. The main target in molecule design is to synthesize macrocycles which are able to discriminate among the different metal cations.

One of the major problems in using macrocyclic ligands in such applications is the maintaining of the very expensive ligand in the water or organic phases. This makes their commercial use expensive, difficult to engineer, and a potential environmental hazard. One way to overcome these problems is the covalent bonding of the macrocyclic ligands to a solid support enhancing greatly at the same time their usefulness.

Our primary interest has been on the one hand, the synthesis of new tetraazamacrocyclic ligands in close alliance with Prof. R. Guilard Laboratory (Universit de Bourgogne) and on the other hand the study of their reactivity with lanthanide- and actinide-ions. Our second interest has been the macrocyclic ligand attachment on polymers for the recovery of ion-traces.

RESULTS AND DISCUSSION

The synthesis and characterization of new and original molecules have been carried out at the University. The DOTA and TETA compounds (cyclic tetraazatetraacetics acids) were prepared by reaction of the corresponding cyclic amines with chloroacetic acid in aqueous alkaline solution as described in the literature (4). A smooth and convenient one step synthesis of the other compounds is reported on Fig. 1. These ligands are synthesized by Michael addition of tetraazacycloalkanes to acrylic acid in absolute ethanol under reflux. All the compounds are obtained in a high yield (varying from 52 to 77%) after recrystallization in distilled water.

Fig. 1

The alpha emitters contained in Valduc Wastes are essentially plutonium-americium- and uranium-elements. In order to implement our extraction methodology, we have used respectively cerium and europium as models to plutonium- and americium-ions. The stability constants of the different metallo-macrocycles were determined from the potentiometric titrations using the "MICMAC" program (7). Therefore, these new macrocycles offer new potentialities due to the high stability of the formed complexes (pKM > 1020)

Table I

Moreover, their coordination is critically dependent on the nature of metallic ion and nitrogen substituents. Consequently, the determination of the conformation of these ligands is a key point.

The X-Ray structure of the DOTA/Gadolinium complex published by Dubost & al. (8) shows at the solid state a metal coordination at once by nitrogen and oxygen atoms. In this way, the metal-ion is encapsulated by the macrocyclic ligand (see Fig. 2a). On the contrary, a crystallographic study of the TETP/Cerium complex (9) exhibits at the solid state a very different coordination mode : in this case, the metal-ion is coordinated to the ligand only via oxygen atoms belonging to four different macrocyclic ligands (see Fig. 2b).

Fig. 2a,b

Whatever the coordination mode may be, the high stability of formed complexes prove the great potentialities of such molecules. Therefore, these results allow us to consider their use in alpha decontamination processes.

These macrocycles have been covalently bound to polymers (essentially Silica Gel) for using in solid-liquid extraction process. Macrocyclic ligand attachments on Silica are relatively simple reactions especially when compared to immobilizations involving organic polymers. Moreover, Silica Gel is a low-cost starting material and exhibits a good mechanical strength and swelling required for use, as an example, in high pressure liquid chromatography systems. For instance, macrocycles have been attached to Silica Gel by amide linkages (10) and by covalent carbon-silane bonds through a hydrosilylation process. Three different methods were used for capacity determinations (in complexed forms) : quantitative

E.S.R. measurements of copper (II) complexes, nitrogen, carbon and copper elemental determinations and UV-visible measurements.
Fig. 3
Industrial and synthetic solution decontaminations concerning uranyl nitrate have been realized specifically with a material (Si2323TrP : compound with Silica Gel). This compound is a cyclam (trisubstituted with propionate groups) attached by nitrogen linkage to Silica.
At a laboratory scale, synthetic effluent treatments show a complete decontamination of the treated effluent. In addition, successive extractions and elutions (using nitric acid) of uranium on Silica Gel or Si2323TrP prove the regeneration of these materials and consequently an easy recovery of the uranium-ions present in the effluents. Industrial solution decontaminations concerning plutonium- and americium-ions induce the same results (see Table II) : ion concentration < 5 Bq/m3 and equal to detection limits of analysis measurements.
On the basis of such results, the CEA has decided to build up a pilot facility to treat one part (1/10) of the annual effluent flow generated by Valduc Nuclear Center to confirm the preliminary study.
Table II
CONCLUSION
The tetraazamacrocycles offer large potentialities in various domains. The macrocyclic chemical research generate target molecules which are able to complex selectively a specific metal. Few ligands of this series, with appropriate geometry and substituents could permit the selective complexation of various metal ions and provide a real contribution in the protection of the environment.
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Session 08 -- SAVANNAH RIVER SITE - PROGRESS IN WASTE MANAGEMENT AND ENVIRONMENTAL RESTORATION
Co-chairs: Lou Papouchado, WSRC
Rosemary Aldridge, USDOE
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TECHNOLOGY IMPLEMENTATION AND CLEANUP PROGRESS AT SAVANNAH RIVER SITE
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ABSTRACT
The integrated high level waste system at Savannah River was designed to convert 34 million gallons of liquid waste to glass and saltstone. Feed pretreatment and saltstone have started up and vitrification is scheduled

to start up in early 1996. New waste disposal vaults and startup of several other facilities such as the Consolidated Incinerator Facility and a mixed waste vitrification facility will help completion of the integrated system to treat and dispose of SRS wastes. Technology was utilized from industry, other laboratories, or was developed at the Savannah River Technology Center if it was not available. Many SRTC developments involved academia and other labs. SRS also has over 400 waste sites (400 acres) units in its characterization/remediation program. To date over 90 acres were remediated (23%) and by 1997 we plan to remediate 175 acres or 44%. Thirteen groundwater facility treatment sites will be in operation by 1997. SRS has provided and continues to provide unique test platforms for testing innovative remediation, characterization and monitoring technologies. We are currently testing DNAPL characterization and remediation and an in-situ Inorganic remediation technique for groundwater.

INTRODUCTION

The Savannah River Site (SRS) has operated since 1952 to produce Nuclear Materials for National Security. The site included five production reactors, two chemical separation plants, a fuel and target fabrication facility, and a tritium processing facility. For 35 years the site produced tritium and plutonium for the nuclear weapons stockpile. With the end of the Cold War, the site has increased its focus on waste treatment and environmental remediation to solve problems generated from decades of production. Supporting these operations and changing missions is an applied research and development laboratory - the Savannah River Technology Center (SRTC).

The Savannah River Technology Center's (SRTC) main mission has been to support site operations and, as a result, has developed a strong customer orientation and an understanding of site needs. Many of the processes and facilities that have operated at the site over the years, or that have recently started up, have been developed or enhanced by SRTC. In most cases, the development scientists and engineers participated from inception to startup. This has been one of the ingredients in the successful technology development approach used at the site. Utilizing integrated teams throughout the system life cycle to include all contributing groups - currently referred to as Integrated Product/Process Development Teams - has also helped. Another successful philosophy has been to couple small scale laboratory work with actual radioactive material with large scale or pilot facilities with simulated materials. We have found pilot scale testing to be invaluable for process and equipment development, process integration, materials evaluation, and finally, startup and troubleshooting of the actual plant. In several cases field demonstrations of the technology proved to be a critical element of the development cycle. Another principle has been to understand the fundamental science of the processes we operate and to retain that knowledge on site.

The legacy of this long period of nuclear materials production has resulted in over 34 million gallons of stored high level liquid waste (HLW), low level, hazardous, mixed, and transuranic solid wastes, and over 400 waste and groundwater units. In the early '80s, visionary site management pushed for and obtained support for the Defense Waste Processing Facility (DWPF) to treat the highest risk legacy waste - the stored high level liquid waste. This was followed by several other facilities to treat other waste streams and has now evolved to a detailed

and integrated Site Treatment Plan. A summary of the major waste streams, their treatment facilities and ultimate disposal is shown in Fig. 1. Many of the facilities are interconnected or depend on one another to provide an integrated approach to waste treatment.

Fig. 1

The integrated high level waste system at Savannah River was designed to convert 34 million gallons of liquid waste to glass and saltstone. Feed pretreatment and saltstone have started up and vitrification is scheduled to start up in early 1996. The system, shown in Fig. 2, consists of five separate but interconnected facilities (1). The waste sludge, which consists of metal hydroxides and oxides is first washed with water to reduce the alkali salts going to vitrification. Some batches will be washed with hot caustic first to reduce the aluminum content (reduce glass volume and viscosity). The first batch, Tank 51, has been washed and is ready to go to vitrification. The waste soluble salts will be treated in the "In-Tank Precipitation" facility. Sodium tetraphenyl borate is added to precipitate the major radioactive component, Cesium-137, and sodium titanate is added to sorb minor traces of strontium and plutonium. The slurry is then filtered by cross flow filtration. The resultant radioactive precipitate is sent to vitrification while the decontaminated salt solution is sent to "Saltstone" (grout) for treatment and vault disposal. The ITP facility started radioactive operation in September 1995 and has worked very well. A DF of 45,000 was obtained on the first batch, exceeding the target value of 1,500.

Fig. 2

In developing these pre treatment processes, pilot plant tests were run and combined with small scale tests on actual HLW to achieve a high degree of confidence. For ITP, a full scale plant demonstration was also performed which generated about 500,000 gallons of decontaminated salt solution.

The Saltstone plant was started up in 1990 and since startup has continually grouted the bottoms from the F/H Effluent Treatment Facility and the salt solution from the demonstration. With ITP in operation, the plant production rate will be greatly increased.

The DWPF vitrification plant has completed its waste qualification runs and is scheduled to receive its first radioactive sludge feed in early 1996. The cesium precipitate stream will be fed starting in mid 1996 when the entire integrated system will be in coupled operation. The vitrification process was extensively tested with an integrated pilot plant, multiple prototypes, and small scale processing in shielded cells with actual waste. Developers of the process are assisting in the startup and pilot plant engineers are working in the plant. The Saltstone process, on the other hand, was more conventional technology and was not piloted. It did require development in formulating the waste forms (2) and in conducting a performance assessment on the disposal vaults (3). The waste from (salt solution, cement, slag and flyash) and facility were designed such that groundwater from monitoring wells adjacent to the facility meets EPA drinking water standards. The facility is permitted as an industrial Waste Water Treatment Plant by South Carolina Department of Health and Environmental Control. Saltstone would be classified by the NRC as a Class A waste (least radioactive).

The F/H Effluent Treatment Facility was started up in 1987 and was designed to handle effluents (evaporator overheads) from the F and H areas (Separations and Waste Management) which were previously sent to

seepage basins. These low level liquid waste streams contain a wide range of inorganic and some organic contaminants. The process consists of reverse osmosis to remove 90% of the dissolved salts, mercury removal columns, and a polishing resin to remove residual Cs and Sr. The front end consists of a pH adjustment and filtration to remove solids followed by a carbon bed to remove organics. The concentrates are evaporated and the bottoms sent to Saltstone for disposal. This versatile cleanup system has board applicability at other sites. It has worked very well to meet NPDES discharge limits. It does not, however, remove any tritium contamination. The process was developed at SRTC and key steps in the process tested on a small scale with actual waste.

The Consolidated Incinerator Facility (CIF) was conceived as a versatile treatment for combustible liquid and solid wastes. It can handle low level, hazardous and mixed wastes. This 13 million BTU plant can burn 5.8M lbs solids/year and is a key facility in the SRS Site Treatment Plan. It consists of a rotary kiln, an afterburner and uses a wet scrubber system. Commercially available technologies were used in this project. SRTC did conduct pilot tests in areas of higher uncertainty such as the rotary kiln seals and the off-gas scrubbing system (4). The CIF is scheduled to destruct the benzene generated in DWPF from the hydrolysis of tetraphenyl borate salts. The ash and blowdown by products from the CIF will be immobilized in cement and disposed of on site in vaults. Hot operation is scheduled for early 1996.

For the disposal of low level waste, SRS has transitioned from direct disposal in trenches, to B-25 steel boxes in trenches, to B-25 boxes in vaults starting in 1994. The E-Area vaults are the first disposal system within DOE to have an approved performance assessment (5,6). We currently have vaults designed for low level, intermediate level, and tritiated wastes. A RCRA vault is planned as a future facility. Waste minimization is strongly emphasized on site. In addition, volume reduction using onsite compactors and offsite vendor treatment is receiving focused attention. The CIF will greatly contribute to volume reduction of the combustible waste as it comes on line.

For the numerous on-site streams of non-combustible, mixed waste, several treatments are being pursued for the most cost effective disposal. For example, one major stream, 6.5 x 106 gallons of waste sludge remaining from the metallurgical area operations, will be treated by a commercial vendor on-site. The novel contract is based on a fee per gallon of waste vitrified. SRTC assisted in the glass formulations to be used with the vendor's commercial melter/offgas system. This approach is applicable to other SRS mixed and low-level wastes and to the other DOE sites.

Transuranic (TRU) waste at SRS is stored on covered concrete pads while older TRU waste has been mounded over with earth. The site has 9.5 x 103m3 of TRU waste containing 5.75 x 105Ci. The high activity is due to the Pu-238 production operations at the site. The plan is to retrieve the waste and to repackage and characterize for shipment to WIPP. The site is also looking at OTD for treatment technologies for non-shippable material. This is a high priority need at SRS.

In summary, SRS has in place most of the facilities needed to treat and dispose of its wastes. Several key facilities have started up such that the whole integrated system is in operation.

Shifting over to the remediation side, SRS has over 400 waste units totaling approximately 400 acres which are in the characterization/remediation program. To date, 14 sites totaling 90 acres

(23%) have been remediated and by 1997, 23 sites will be closed totaling 175 acres or 44%. In addition, it is expected that a significant number will not need to be remediated after characterization is complete. The site has been aggressive in closing most of the major disposal basins. The basins adjoining the F and H Separations areas have been closed as has the Fuel and Target Fabrication (M) Area basin. In addition, a 58 acre mixed waste disposal site has been closed. The closures in all cases have consisted of a low permeability clay cap with drainage layers and a vegetative cover. The design of each closure was negotiated and approved by the state. With the basins, a sludge stabilizing step was included prior to capping. These have included grout, calcium carbonate stabilizer, and limestone/flyash additions. In the case of the mixed waste disposal site, dynamic compaction was performed to prevent future subsidence from damaging the clay caps. In this process, a 20 or 40 ton weight is repeatedly dropped on the surface using a crane.

Groundwater contamination at SRS consists principally of VOCs and radionuclide/metal contamination at several locations. Thirteen groundwater facility treatment sites will be in operation by 1997.

SRS was the location of the successful DOE-OTD Integrated Demonstration "Remediation of VOCs in Groundwater and Soils at Nonarid Sites." This demonstration, which has been extensively described, showed how to conduct successful collaborations between DOE laboratories, universities, industry and other government agencies. Integrated development and field deployment teams, using some of the best national resources, successfully demonstrated and later commercialized technologies and equipment that address characterization, monitoring and remediation of VOCs. Two notable remediation technologies were in-situ air stripping using horizontal wells, and in-situ bioremediation using methane and PHOSter injection (7). This well characterized site provides an ideal test platform and has received regulatory and stakeholder buy-in to test and implement innovative technologies. This experience has allowed SRS to develop several other test platforms to address different contaminants and environments. Extensive characterization and data sets are available as well as technical support and field analytical support for demonstrations or vitrification of technologies (8).

SRS is currently involved in several other demonstrations: Characterizing and remediating Dense Non-Aqueous Phase Liquids (DNAPLs), remediating metal ions contamination in groundwater, and testing innovative sanitary landfill remediation. Partnering with academia, industry and other labs has become a routine operating mode. This allows leveraging of DOE resources, utilizes the best available experts, and facilitates commercialization of new technologies.

Good progress is being made at SRS in Environmental Management because of visionary early planning, a bias for action and technology implementation, a needs-driven systems approach, and integrated process development teams.

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THE DWPF: RESULTS OF FULL SCALE QUALIFICATION RUNS LEADING TO RADIOACTIVE OPERATIONS

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ABSTRACT

The Defense Waste Processing Facility (DWPF) at the Savannah River Site in Aiken, SC will immobilize high-level radioactive liquid waste, currently stored in underground carbon steel tanks, in borosilicate glass. The radioactive waste is pretreated and then combined with a borosilicate glass frit. This homogeneous slurry is fed to a Joule-heated melter that operates at 1150C. The glass is poured into stainless steel canisters for eventual disposal in a geologic repository. The DWPF canistered waste forms must comply with detailed waste acceptance specifications to be acceptable for eventual disposal.

The DWPF has completed Waste Qualification Runs which demonstrate the facility's ability to comply with the waste acceptance specifications. During the Waste Qualification Runs seventy-one canisters of simulated waste glass were produced in preparation for Radioactive Operations. These canisters of simulated waste glass were produced during five production campaigns of varying melter feed composition that also exercised the facility prior to beginning Radioactive Operations. The results of the Waste Qualification Runs are presented.

INTRODUCTION

Approximately 130 million liters of high-level radioactive waste, currently stored in underground carbon steel tanks at the Savannah River Site (SRS), will be immobilized in stable borosilicate glass in the Defense Waste Processing Facility (DWPF). The glass is poured into stainless steel canisters for eventual disposal in a geologic repository. To be acceptable for disposal the DWPF product (i.e. the canistered waste form) must comply with the Department of Energy Office of Environmental Management's Waste Acceptance Product Specifications (WAPS). (1)

The DWPF has recently completed the production of seventy-one canisters of simulated waste glass as part of it's Startup Test Program in preparation for Radioactive Operations. The Waste Qualification Runs portion of the DWPF Startup Test Program was designed to demonstrate that the DWPF could comply with the WAPS prior to the start of Radioactive Operations. Fifty-five of the seventy-one canisters of simulated waste glass were produced during these runs. Varying feed compositions were used to demonstrate that the DWPF could control the glass product over a range of materials using the DWPF Glass Product Control Program. (2) The simulated waste was transferred into the DWPF and processed using the same methods as for the radioactive waste. The glass and canistered waste forms produced during Waste QualificationRuns were extensively characterized. The results of this characterization demonstrate the DWPF's ability to comply with the WAPS.

PROCESS/PRODUCT OVERVIEW

The radioactive waste in the SRS Tank Farms has been separated into a water soluble salt solution and saltcake, and an insoluble sludge of metal hydroxides and oxides. The salt solution and saltcake are decontaminated for disposal as low-level radioactive waste by removing the radionuclides by precipitation and sorption. Sodium tetraphenyl borate is added to precipitate soluble salts of non-radioactive potassium and cesium. Sodium titanate is added to adsorb residual strontium and plutonium. The resulting slurry is filtered and the decontaminated filtrate is blended with cement, slag and flyash for disposal at SRS as a low-level radioactive waste. The slurry of the concentrated solids is transferred to DWPF for immobilization. The sludge portion of the waste is washed to remove soluble salts. If necessary, insoluble aluminum is removed through high temperature caustic dissolution. Thus, the radioactive waste from the SRS Tank Farms is transferred to the DWPF in two forms: precipitate slurry and sludge slurry. The waste is then processed and blended in the DWPF before it is vitrified, poured into canisters, sealed, and placed in interim storage. (3) See Fig. 1 for a schematic of the DWPF process.

The sludge is transferred directly into the Sludge Receipt and Adjustment Tank (SRAT) while the precipitate must first be processed in the DWPF Salt Process Cell to remove most of the organic material. The tetraphenyl borate compounds in the precipitate react in the presence of formic acid and copper (II) catalyst. The products of this reaction are aromatic organic compounds (benzene, phenol, and minor amounts of higher boiling aromatics) and an aqueous phase known as Precipitate Hydrolysis Aqueous (PHA). The PHA contains the cesium, soluble formate salts, boric acid and excess formic acid.

The sludge is neutralized with nitric acid in the SRAT. The PHA is then added to the sludge (at boiling). The excess formic acid in the PHA reduces the mercuric oxide in the sludge to elemental mercury. The elemental mercury is then steam stripped from the SRAT into a holding

tank from which it is later pumped and decontaminated. After the PHA and sludge are blended and processed in the SRAT, this SRAT product is transferred to the Slurry Mix Evaporator (SME) where a borosilicate glass frit is added and the slurry is concentrated to produce melter feed. The amount of sludge and PHA to be blended in the SRAT and the amount of SRAT material and frit to be blended in the SME is determined by the Product Composition Control System (PCCS). (4) The PCCS is a computer program that uses glass property models and statistical algorithms to develop blending strategies and to determine the acceptability of the melter feed before it is transferred to the melter. The PCCS uses lab analyses of tankcontents, tank volumes and existing tank heel volumes to determine an appropriate tank blending region. It chooses a target that minimizes the amount of frit to be added in the SME. However, any point in the acceptable blending region can be chosen by DWPF engineers as the blending target. PCCS is also used to predict SRAT and SME compositions based on existing tank compositions, tank transfers, and the uncertainties associated with sampling and processing. These predicted compositions for the SRAT and SME can be used to diagnose processing problems. This capability was used during Waste Qualification Runs as is described in a later section.

Fig. 1

The SME is the hold point in the process where feed acceptability is determined as part of the Glass Product Control Program (see below). The analyses of samples from the SME are used by the PCCS to determine the acceptability of the melter feed. The acceptability of the melter feed is determined using glass property models and statistical algorithms which take into account analytical uncertainty to ensure the glass product will be acceptable and that processing constraints (such as viscosity and liquidus) are met.

Once the melter feed material in the SME is determined to be acceptable, it is transferred to the Melter Feed Tank (MFT) and then fed to the joule heated melter. The DWPF melter has two pairs of electrodes. The feed slurry is introduced from the top of the melter and forms a crust, or cold cap, on the surface of the melt pool as the water is evaporated and removed via the off-gas system. The cold cap melts from the bottom and forms theborosilicate glass matrix. The nominal glass melt pool temperature is 1150C. The mixing behavior of the melter was evaluated during Waste Qualification Runs as described later. The glass is removed from the melter near the bottom through a riser and pour spout. A vacuum is drawn on the pour spout to pour the glass. A glass pour stream sample is taken occasionally during filling of a canister to confirm that the glass durability (as determined by the Product Consistency Test (5)) is acceptable. After a canister is filled, a temporary seal is installed to prevent free liquid from entering the canister during the decontamination process. Decontamination of the canister surface consists of blasting an air-injected frit slurry against the canister. The frit slurry from the decontamination process is used in the next SME batch as part of the required frit addition. The canister is then welded closed and transferred to an interim storage building. The DWPF canistered waste form contains approximately 1800 kg of glass. It is 300 cm in length and 61 cm in diameter.

WASTE ACCEPTANCE SPECIFICATIONS

To ensure that the DWPF product is acceptable for final disposal the Department of Energy Office of Environmental Management has developed the

Waste Acceptance Product Specifications (WAPS) (1) which the DWPF product (i.e. canistered waste form) must meet. These specifications are divided into five sections: waste form (borosilicate glass), canister, canistered waste form, quality assurance, and documentation. The most important of the glass specifications is the product consistency specification which states that the DWPF must control its process so that the glass produced is more durable than the DWPF Environmental Assessment glass (6) as measured by the Product Consistency Test (PCT). The PCT is a crushed glass durability test in which the results are expressed as the amount of boron, lithium, and sodium measured in the leachate. DWPF has developed a Glass Product Control Program (see below) to ensure that this specification is met. During Waste Qualification Runs, the DWPF demonstrated that it can comply with this specification as well as the other glass, canister, and canistered waste form requirements.

GLASS PRODUCT CONTROL PROGRAM

The DWPF has developed the Glass Product Control Program to ensure that the DWPF consistently produces a glass which satisfies the product consistency specification, and that there is demonstrable evidence that this has been achieved. In developing this strategy for producing an acceptable glass product, the DWPF has considered the following:

The only parameters which the DWPF can directly control that affect the results of the PCT are the chemical composition of the glass, and the uniformity of the feed to the melter. It has been shown in laboratory and large scale testing that chemical composition is the prime determinant of a glass's PCT results. (7)

It is not possible to recycle or rework glass which does not satisfy the specification in the current DWPF processingequipment.

DWPF waste glass is highly radioactive. Any testing to be performed for control or verification must be performed in shielded cells and, thus, must be simple and reliable.

The waste stream coming into the DWPF will vary. Thus, the program must be able to handle varying waste compositions.

The DWPF Glass Product Control Program ensures that the glass product satisfies the WAPS by controlling the composition of the melter feed. As described earlier, control of the melter feed is ensured by the hold point at the SME. No material is allowed to be transferred from the SME to the MFT until it has been determined to be acceptable. The PCCS is used by DWPF engineers to ensure that the SME material is acceptable. A glass property model that uses a thermodynamic hydration approach to represent composition is used to predict PCT results based on the SME composition. The free energy of hydration reaction for each glass component is multiplied by that component's mole fraction in the glass. These partial quantities are summed to represent the free energy of hydration of the glass. The free energy of hydration is related to the glass PCT results. The ability of the Glass Product Control Program to produce an acceptable glass product was demonstrated during the Waste Qualification Runs portion of the Startup Test Program.

STARTUP TEST PROGRAM

The DWPF Startup Test Program was modeled on the testing required for startup of a commercial nuclear reactor. The Startup Test Program:

Demonstrated the DWPF's ability to reliably produce an acceptable product that meets the requirements of the WAPS. This was accomplished during the Waste Qualification Runs portion of the Startup Test Program.

Demonstrated the operability, reliability, and integrity of the major process systems.

Provided operating experience to operations, maintenance, and engineering personnel.

Baselined equipment and system operating parameters.

Melter Startup

Prior to Waste Qualification Runs the melter was heated up and one melter campaign (FA-13) was performed. Two thousand pounds of startup frit were loaded into the melter. The startup frit was formulated to avoid corrosion and other problems during startup. The melter dome heaters were turned on to melt the top surface of the startup frit. Once the glass became molten enough to allow joule heating, the lower electrodes were energized. Once joule heating was established the first batch of melter feed material was fed into the melter until the normal melter level was reached. Once the melter glass level reached the upper electrodes they were energized. One canister was filled using the melter feed remaining from the batch used to fill the melter. Two more melter feed batches were produced and eleven morecanisters were filled. A fourth melter feed batch was produced during this campaign and held in the Melter Feed Tank. The DWPF then entered an extended outage during which modifications were made to the process vessel vent system. Following the outage four more canisters were filled using the remaining feed from the first campaign. The canisters produced during FA-13 were processed as shown in Table I. The melter feed for this first campaign was Blend feed (i.e. a blend of all waste types in the SRS Tank Farm). The glass and canisters from this campaign were characterized to ensure that the facility was ready for Waste Qualification Runs as well as the procedures and personnel for glass and canister characterization.

Table I

During this first melter campaign problems with pour spout pluggage occurred. The blockage was cleared but this problem continued to occur during Waste Qualification Runs and initiated an effort to design and procure a manipulator arm that could remotely clear the glass blockages.

Waste Qualification Runs

During the Waste Qualification Runs, fifty-five canisters were filled, over four melter campaigns, with simulated waste glass which was produced in accordance with the Glass Product Control Program. During these four melter campaigns, the feed coming into the DWPF went through abrupt changes in composition. The purpose of making such abrupt changes was to demonstrate that the Glass Product Control Program is able to control the glass product even when the feed composition is rapidly changing. This should enhance the confidence in the use of the program during normal operations, when changes in feed composition will be more modest.

The specific compositions processed during the Waste Qualification Runs were selected based on the thesis that if the composition/PCT correlation has been used properly to judge the acceptability of the feed, the only possible cause of failure of the Glass Product Control Program is segregation of the feed. This will occur only if the rheological properties of the material are not consistent with good mixing in the process vessels. Thus, the primary purpose of varying the feed composition during Waste Qualification Runs was to test a rangeof rheological properties, particularly for the melter, and not to qualify a set of compositions. The compositions made step changes from a Blend (design basis) composition with a dopant to track melter mixing behavior,

to a low viscosity feed, to a high viscosity feed, and back to a Blend composition. The range of the major components in the sludge and then the range of the major components in the melter feed (SME composition) over Waste Qualification Runs is shown in Table II. Although, DWPF is not qualifying a set of compositions, these compositions represent the extremes of the waste currently present in the SRS Tank Farms.

The first campaign of Waste Qualification Runs, WP-14, was the Blend feed doped with neodymium to serve as a tracer to monitor melter mixing. Neodymium was added to the SME. One batch of melter feed was produced which yielded seven canisters of simulated waste glass. Glass pour stream samples were taken from the first six of the seven canistered waste forms produced. These seven canistered waste forms were processed as shown in Table I.

An example of the importance of the SME hold point occurred during the production of this first batch of material. During the evaluation of this batch for acceptability, the liquidus criterion (glass property related to processing) was not met. However, the glass durability (i.e. predicted PCT results) was acceptable. It was determined that not enough frit had been added to the batch. The PCCS was utilized to confirm that the addition of 3600 pounds of frit to the SME would bring the batch into the acceptable region for the liquidus criterion and keep the durability acceptable. Thus, following the Glass Product Control Program, 3600 lbs of frit were added to the SME batch, the batch was agitated, and then sampled again. The results of the sample analyses were evaluated and the batch was determined to be acceptable. The SME batch was then transferred to the MFT.

Table II

During the second campaign of the Waste Qualification Runs, WP-15, four batches of high iron feed were prepared, which yielded twenty canistered waste forms. Glass pour stream samples were taken from all but the first and the eighth canistered waste forms produced. These twenty canistered waste forms were processed as shown in Table I. Prior to the performance of this campaign, problems had been experienced in pilot plant tests with melter feeds which were high in iron, and contained alkali metal ions in relatively high concentrations. The glass produced from these feeds was much less durable (as measured by the PCT) than predicted by the PCCS. Many of the glasses produced in the pilot plant testing were also phase separated, indicating that this might be the cause of the discrepancy. As a result, a variability study was undertaken with this feed. This study showed that the PCT results for the PCCS target glass were reasonably consistent with the PCCS predictions, but that there were large discrepancies between the predicted and actual PCT results for most of the other glasses tested. As called for by the GPCP, a model was developed for use on this material, called the Purex model. The resulting model essentially fits a straight line between two clumps of data at either end of the free energy of hydration range, and overpredicts PCT results of glasses in the middle of the range. This provides additional conservatism in PCT predictions for glasses with intermediate free energy of hydration values. Thus, its use would prevent production of unacceptable glass.

The decision was made to proceed with the campaign, but to control the composition of the feed through the model discussed above; through the controls already in place in the Glass Product Control Program (including the use of the PCCS); and by taking samples of each SME batch, vitrifying

them, and subjecting the vitrified samples to the PCT. If any of these indicated a problem with a SME batch, the SME batch would be remediated as called for in the GPCP, again using the same three control measures. During the third campaign of the Waste Qualification Runs, WP-16, four batches of high aluminum feed were prepared, which yielded nineteen canistered waste forms. Glass pour stream samples were taken from all but the sixteenth and seventeenth canistered waste forms produced. These nineteen canistered waste forms were processed as shown in Table I. Mercury was added to the precipitate in the salt process cell and to the sludge in the SRAT during the last batch of WP-16 to successfully demonstrate the mercury removal system.

During the production of the last WP-16 batch of melter feed the predicted composition for the SRAT was used by the DWPF engineers to diagnose an operating concern. The measured composition of the SRAT product was not consistent with the predicted composition from the PCCS. The analyzed sample results were lower in aluminum, iron, and silicon. After further investigation, it was found that foaming occurred in the SRAT during processing and sludge was carried over to the condensate tank. It is believed that this foaming occurred as a result ofthe addition of mercury to the precipitate and sludge. As a result, changes were made during processing of the remaining WP-16 batches to prevent reoccurrence. These changes included lower steam rates and an increase in antifoam additions.

During the fourth campaign of the Waste Qualification Runs, WP-17, two batches of feed were prepared. This feed transitioned back to the Blend feed from a high aluminum feed. These two batches of feed yielded nine canistered waste forms. Glass pour stream samples were taken from all but the first canistered waste form produced. These nine canistered waste forms were processed as shown in Table I. No problems had been observed with this type of feed, which was similar to the feed used in WP-14, so no special control measures were used. Mercury was added to the precipitate in the salt process cell. Noble metals and mercury were added to the sludge in the SRAT to further demonstrate the mercury removal system and simulate operations with noble metals in the sludge. The noble metals added were: ruthenium, rhodium, palladium and silver. No significant processing problems were encountered during this campaign. The mercury was successfully removed from the melter feed and no unusual problems were encountered with the addition of noble metals.

Glass and Canister Characterization

After processing was completed the filled canisters were transported to a test facility for destructive examination. The canistered waste forms were either sectioned or a portion of the canister wall removed and glass samples taken (except for two of the canisters produced during WP-17 which were archived intact). Sectioning consisted of using a band saw to slice through the entire canister at three levels. One of the slices was made at the height corresponding to the level of glass in the canister when the pour stream sample was taken. Glass samples were taken at four radial locations for each of the three levels. All the glass samples, including the DWPF pour stream samples, were characterized for chemical composition, devitrification, and durability (measured by the PCT). The chemical composition results from the glass samples were compared to the results of the melter feed sample analyses. The PCT results of the glass samples were compared to the PCT results predicted by the PCT/chemical composition correlation in PCCS. Also, the glass pour stream sample

results were compared to those of the glass samples taken directly from the canisters. These comparisons demonstrated that an acceptable product had been produced under the controls of the Glass Product Control Program.

The PCT results for all the glass produced during the Waste Qualification Runs were far below the values for the benchmark EA glass (see Table III). The mean PCT values for each of the 55 canisters produced clearly lie far below the values for the EA glass. For each of the campaigns, there is greater than 99% confidence that at least 99.9% of the least durable glass produced has better PCT results than the EA glass. This demonstrates that use of the Glass Product Control Program by theDWPF will ensure production of a product which satisfies the product consistency specification.

For all of the campaigns, there were statistically significant variations in the PCT results as a function of sample location in the canister, and sometimes from canister-to-canister. This is not too surprising, because the feed was varying for each of the last three campaigns. However, none of these affected the ability of the product to satisfy the product consistency specification. Figure 2 compares the mean PCT results for each canister to the predictions for each batch of melter feed.

Table III

The figure shows that the actual PCT results were adequately predicted and shows the lag in the glass as compared to the melter feed. This lag is due to heels in the process tanks and the melter volume.

Characterization of the canisters produced during the Waste Qualification Runs was also performed. The following activities were performed:

six canistered waste forms were tested for foreign materials by sampling the free air space above the glass level to demonstrate that DWPF can exclude foreign materials from the canister

nine canister welds were leak tested, burst tested, and microstructurally examined to demonstrate DWPF's ability to make an acceptable weld

five canistered waste forms were dimensionally measured before and after filling to demonstrate that canister filling does not significantly affect canister dimensions

Melter Mixing

The neodymium which was added as a tracer in WP-14 was tracked in the glass product and evaluated against models for a continuously stirred tank reactor and a plug flow reactor. As expected, excellent agreement was obtained between the predicted Nd concentration in the glass using the continuously stirred tank reactor model and the actual Nd concentration in the glass product. A four day hold was placed on glass pouring to allow the melter to mix after a portion of the third canister was filled. For a continuously stirred tank reactor the average concentration in the melter is the same as the outlet concentration. For a plug flow model the average concentration would be greater than the outlet. After the four day hold there was no significant effect on the difference between the predicted and actual concentrations further indicating that the DWPF melter was best represented by a continuously stirred tank reactor.

Fig. 2

TRANSITION TO RADIOACTIVE OPERATIONS

Due to SRS Tank Farm processing and blending strategies the DWPF will begin Radioactive Operations with a sludge only (no PHA)process. A frit

composition higher in alkali, to compensate for the missing PHA, will be mixed with this first radioactive sludge batch. A dilute formic acid solution will be added to the SRAT in place of the PHA. The major components of this first batch of radioactive sludge (on a dried solids wt% basis) are: Al - 6.39; Fe - 24.6; Na - 8.74; U - 3.42; Mn - 2.53; Ca - 2.38; Mg - 1.16.

This first batch of sludge has been washed five times to remove the soluble salts. Inhibited water has been added to the tank and mixing accomplished using four long-shaft slurry pumps. The wash water (containing the soluble salts) was transferred out of the tank and fresh water brought in five different times.

Currently, the DWPF is processing two sludge-only batches of simulated feed using a sludge simulant that closely matches the first batch of radioactive sludge. After the completion of these batches and startup authorization has been given, the DWPF will introduce radioactive sludge into the plant to begin the process of immobilization of high level waste in borosilicate glass.

CONCLUSIONS

The DWPF Startup Test Program has allowed the DWPF to demonstrate the functionality of its systems, exercise and refine operating procedures, and provide operators and engineers an opportunity to run the facility prior to introducing radioactive feed. During the Waste Qualification Runs portion of the Startup Test Program, DWPF has demonstrated its ability to comply with the WAPS providing assurance that the facility can produce a product acceptable for disposal. The results of this testing provide valuable experience for future operations and provide assurance that DWPF can effectively immobilize the SRS liquid high level waste in borosilicate glass.

ACKNOWLEDGMENT

This paper was prepared in connection with work done under Contract No. DE-AC09-89SR18035 with the U.S. Department of Energy.

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BENCH-SCALE VITRIFICATION STUDIES WITH SAVANNAH RIVER SITE MERCURY CONTAMINATED SOIL

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ABSTRACT

The Savannah River Technology Center (SRTC) has been chartered by the Department of Energy (DOE) - Office of Technology Development (OTD) to investigate vitrification technology for the treatment of Low Level Mixed Wastes (LLMW). In fiscal year 1995, LLMW streams containing mercury were targeted. In order to successfully apply vitrification technology to LLMW containing mercury, the types and quantities of glass forming additives necessary for producing homogeneous glasses from the wastes had to be determined and the treatment for the mercury portion had to also be determined. Selected additives had to ensure that a durable and leach resistant waste form was produced, while the mercury treatment had to ensure that hazardous amounts of mercury were not released into the environment.

The mercury containing waste selected for vitrification studies at the SRTC was mercury contaminated soil from the TNX pilot-plant facility at the Savannah River Site (SRS). Samples of this soil were obtained so bench-scale vitrification studies could be performed at the SRTC to determine the optimum waste loading obtainable in the glass product without sacrificing durability and leach resistance. Vitrifying this waste stream also required offgas treatment for the capture of the vaporized mercury. Results indicated that a durable, leach resistant glass waste form capable of passing the Product Consistency Test (PCT) and the Toxicity Characteristic Leaching Procedure (TCLP) could be produced. The optimum glass feed composition contained 60 weight percent soil and produced a soda-lime-silica glass when melted at 1350C. The glass additives used to produce this glass were 24 weight percent Na_2CO_3 and 16 weight percent $CaCO_3$.

The proposed mercury capture method was a Na_2S wash bottle followed by a NaOH wash bottle. The volatilized mercury entered the first wash bottle through a bubbler with the intent of converting it to Hg_2S, a very stable form of mercury. If successful, no further treatment of the mercury would have been needed. However, attempts to capture the volatilized mercury in a Na_2S solution wash bottle were not as successful as anticipated. Maximum mercury captured was only about 2% of the mercury contained in the feed. Efforts then shifted to condensing and capturing the volatilized mercury. Condensing attempts were much more successful at capturing the vaporized mercury. This captured mercury could then be treated on a mercury specific resin after digestion of the volatilized mercury.

INTRODUCTION

The DOE has chartered the Mixed Waste Focus Area (MWFA) to investigate waste forms for LLMW. Vitrification or high-temperature thermal treatment of the wastes is a main focus of the MWFA investigations. The MWFA has funded the SRTC to perform vitrification and high-temperature thermal treatment studies on LLMW. The SRTC's efforts have focused on treatment of LLMW sludges, soils, debris, resins, and other solid wastes.

A large focus of the efforts has been treatment of soils due to the large volume of contaminated soil that exists at the SRS and other DOE sites. Soil at the SRS has been contaminated with both radioactive and hazardous constituents as a result of accidental spills and storage of liquid wastes. Some of these soils have been exhumed and containerized, but most have not and will not be until a treatment method is determined.

A small amount of contaminated soil was exhumed at the TNX pilot-plant facility during routine maintenance operations. This soil was characterized and found to contain elevated levels of mercury. Two samples of this soil, one with mercury levels below TCLP limits and one with elevated levels of mercury, were obtained by SRTC for bench-scale vitrification studies. Previous studies by SRTC had shown that vitrification of soil was a viable option (1) and that vitrification offgas systems could successfully capture mercury.

For the mercury soil vitrification viability studies at the SRTC, bench-scale studies were performed with the two soil samples to determine the necessary glass additives for producing homogeneous glasses. The homogeneous glasses produced were subjected to leach testing to determine glass durability, since it was important to ensure that the hazardous/radioactive constituents were incorporated in the glass matrix. Since the waste contained mercury which is not incorporated in the glass matrix, an offgas system had to be installed on the furnace. The intent of this system was to collect the vaporized mercury, as well as convert the mercury to a stable form.

The composition of a sample of the TNX soil with less than TCLP levels of mercury was chemically characterized to determine the chemical constituents. It was suspected that the SRS soil contained mostly SiO_2 and this was confirmed by the chemical analyses. The chemical composition of the soil is given in Table I on an oxide basis. The previously determined total mercury in the less than TCLP limit sample was 1.92 ppm, while the high mercury sample contained 264 ppm total mercury.

Table I

Previous bench-scale studies with simulated SRS soil had determined that durable and homogeneous glasses could be made by using the soda-lime-silica ternary system. Waste loadings in these glasses were 55 and 58 wt%. The glass forming additives used were Na_2CO_3 and $CaCO_3$, and all glasses were melted at 1150C (1). Since these glasses were made using simulated soil, bench-scale tests with actual soil had to be performed to determine the validity of the glass compositions.

LOW MERCURY SOIL EXPERIMENTS

In order to determine the applicability of the previously developed glass compositions for actual SRS soil, two of the glass compositions were selected for bench-scale testing. A higher waste loading batch composition was also derived to ensure that the optimum waste loading had been found. The compositions tested on an additive basis are shown in Table II. Two additional batch compositions were also tested based on the results of these first bench-scale trials and these batch compositions are also shown in Table II.

Table II

The batches listed in Table II were mixed using the low mercury soil sample and reagent grade chemicals. The batches were then placed in high purity alumina crucibles and covered. Batches 1 - 3 were melted at 1150C for 4 hours and visually examined after air quenching. A description of the resulting products are given in Table III. Since it appeared that batch 3 resembled homogeneous glass the most, a higher waste loading was tested as batch 4. This batch was melted at 1175C and its appearance is also described in Table III. As noted in Table III, some of the glasses appeared to have some unreacted material around the crucible edges, it was believed that this material was unreacted SiO_2. This was confirmed by Scanning Electron Microscopy (SEM). Since the material was confirmed to be SiO_2, batch 3 was remelted at 1350C to determine if higher temperatures would fully react the feed material. This resulted in a very homogeneous looking glass with only minimal unreacted particles of SiO_2. Attempts were also made to diminish the formation of this SiO_2 layer by reducing the CaO:Na_2O ratio, since the presence of Na_2O tends to make SiO_2 more soluble. This composition was batch 5 and was also melted at 1350C. The appearance of the final product is also given in Table III. Since batch 3 produced the most homogeneous glass, it was characterized for chemical composition and phase assemblage. Durability tests were also performed using the PCT. The composition of the batch 3 glass is given in Table IV. The only substantial oxide components found in the glass were the SiO_2 and Al_2O_3 from the soil and Na_2O and CaO from the glass additives. X-ray diffraction (XRD) of the glass indicated that the resulting glass was amorphous. SEM analyses verified the amorphous state of the glass, as well as confirmed the presence of small amounts of unreacted SiO_2 at the glass surface.

Table IV

In order to determine the durability of the batch 3 glass, the PCT was performed. The PCT is the standard test used for determining the durability of High Level Waste (HLW) glasses and is performed in an alkaline driven environment. It is a 7-day test performed at 90C in ASTM type I water. The test is performed on 100-200 mesh glass particles and the resulting leachate is analyzed for elemental concentrations (2). These concentrations are then normalized for the elemental glass constituents.

At present no PCT acceptance criteria exist for LLMW glasses. However, the acceptance criteria have been established for HLW glasses. The measured normalized releases for the Environmental Assessment (EA) glass, which is the benchmark for the Defense Waste Processing Facility (DWPF) HLW glass, are 3.922 g/L Si, 13.346 g/L Na, and 16.695 g/L B. The measured leachate pH is 11.91 (3). The normalized PCT results for the batch 3 glass were 0.35 g/L Si, 5.90 g/L Na, and 0.00 g/L B. The measured pH was 11.85. Normalized PCT results for the batch 3 glass were significantly less than the limits for Si and B, and the result for Na was 2.5 times less. These results indicate that the resulting glass product was a durable wasteform.

The TCLP was not performed on the glass since no hazardous constituents other than mercury were present in the soil. Since it was known that the mercury would volatilize during vitrification, no mercury should have remained in the glass to leach during the TCLP.

ELEVATED MERCURY SOIL EXPERIMENTS

Before the treatability studies with the elevated mercury soil sample could be performed in the bench-scale furnace, the offgas collection system had to be fabricated and installed. A detailed drawing of the offgas system fabricated is contained in Fig. 1. The intent of the system was to contain all of the volatile mercury in the quartz lines of the system. Using forced air on the seal of the crucible and vacuum pressure on the end of the system line, mercury was forced through the quartz tubing to the first wash bottle containing Na2S. The vapors entered through the dip tube and were bubbled in the Na2S. Vapors from this tube were drawn through the NaOH wash bottle by the vacuum at the end of the line. In this second bottle, the sulfuric acid gases generated from the first wash bottle were neutralized by bubbling through the NaOH solution. All vapors generated from this bottle were vacuumed through a hood exhaust that was in line with the building offgas system.

Fig. 1

The offgas system was seated in a Thermolyne furnace. For bench-scale vitrification of the elevated mercury soil, batches were heated in a 50 mL platinum crucible to a minimum of 690C and then the final product was transferred to an alumina crucible and melted in a Lindberg high temperature furnace at 1350C. Due to chemical hood space constraints and the desire not to drill a 2 inch hole in the Lindberg furnace, batches were heated using the Thermolyne furnace and the offgas system until temperatures substantially above the mercury vapor point were achieved. Three batches of approximately 50 grams of the same batch composition tested with the low mercury soil (batch 3) were mixed. These batches contained 60 wt% of the elevated mercury soil, 16 wt% CaCO3, and 24 wt% Na2CO3. Three separate trials were performed with these feeds, with two of the batches of feed (Trial 2 and 3) containing 500 ppm of Cs and Ce as radioactive surrogates. In each trial, the batch was placed in a platinum crucible, which was placed inside the quartz containment vessel. The inner containment section was placed directly on the crucible, where it actually overlapped the crucible top by about a 1/4 inch. The vacuum and air were started and then the furnace was turned on. The furnace was heated at a rate of approximately 10C/min until it reached the temperatures shown in Table V. Once at temperature, the temperature was maintained for 2 hours. After the 2 hours, the furnace was turned off and the system was allowed to cool. During the heat-up, maintain temperature, and cool-down cycles, the temperature of the furnace and the thermocouple in the offgas line were recorded. Maximum recorded offgas line temperatures were 269C. Once the furnace had cooled, the air and vacuum supplies were turned off.

Table V

The amounts and concentrations of the wash bottle solutions are also contained in Table V. Trial 3 contained a stronger concentration of Na2S in an attempt to capture more of the mercury. Samples of the two wash bottles were taken so the chemical constituents could be analyzed.

The platinum crucibles of feed from each trial were removed from the mercury collection system in the Thermolyne furnace and placed in a programmable Lindberg furnace and heated to 1350C. After 4 hours at temperature, the crucibles were removed from the oven and the glasses were air quenched to room temperature.

Glass Analyses

Once the glasses had cooled, they were broken out of the crucible for chemical composition, phase assemblage, and durability determinations.

All glasses were blue-green in appearance with small amounts of unreacted SiO_2 at the surface. Durability was determined in alkaline and acidic conditions using the PCT (2) and TCLP (4), respectively. The PCT results were compared against the EA glass accepted values for HLW (3), while the TCLP results were compared to the more restrictive of the TCLP limits, Resource Conservation and Recovery Act (RCRA) Land Disposal Limits, or the Universal Treatment Standards (UTS).

The chemical composition of the glasses produced from each trial are contained in Table VI. Compositions were fairly consistent between each trial, with the exception of the Cs_2O and CeO_2, since these were not added to the batch for the first trial. Some soil variability was evident by the elevated Al_2O_3 and lower SiO_2 numbers compared to the low level mercury soil glass. Part of this variability can be attributed to differences in soil composition and part can be contributed to incomplete mixing of the feed material. Glass analyses results indicated that almost all of the Cs_2O was encapsulated in the glass, which was shown by a comparison of the feed and glass analyses results. For both trials, more Cs_2O was actually found in the glasses, which was probably due to incomplete mixing in the feed sample analyzed. In both cases, the amount retained was greater than the theoretical loading. CeO_2 concentrated in the glass matrix for trial #2, proven from a comparison of the feed to glass results, but results for trial #3 were inconclusive since both the feed and glass levels for Ce were less than the detection limit. HgO concentrations in the feed for each trial were 0.055, 0.077, and 0.015 wt% respectively. For the final glass products, the HgO levels were at the detection limits and were roughly the same irrelevant of the feed concentration levels. This was expected since all of the mercury in the feed was expected to volatilize from the feed. The Fe^{2+}/Fe^{3+} ratio was also measured for the glasses produced, and the results indicated that the melting conditions were oxidizing (average ratio was 0.0494).

Table VI

Phase assemblage was determined using XRD and SEM analyses. In all cases, the glasses were amorphous when analyzed by XRD. SEM analysis of the glasses confirmed the presence of small amounts of unreacted SiO_2. This SiO_2 would likely be fully reacted at elevated temperatures or in a melter environment due to the mixing that occurs.

The PCT was performed on the glasses from each trial. The normalized results are given in Table VII. Normalized releases were comparable to the releases for the low mercury soil glass, which were better than the HLW EA glass PCT limits (3). Na release was once again the highest relative to the HLW EA acceptance criteria, but it was still much better than the EA glass. Trial #2 glass had the lowest Na release, which was consistent with the lowest Na_2O and SiO_2 content and the highest CaO and Al_2O_3 content in the elevated mercury glasses. However, when compared to the low mercury glass, trial #2 glass had higher Al_2O_3, CaO, and Na_2O concentrations and a lower SiO_2 content, which resulted in a more durable glass. These results would seem to indicate that the effects of high concentrations of Na_2O on durability can be offset by increases in Al_2O_3 and CaO concentrations and decreases in the SiO_2 content. This better durability may also have been the result of the increases in Fe_2O_3 content in the trial #2 glass since Fe_2O_3 is known to have a positive effect on durability. Glasses from the second and third trial had minimal releases for Ce (<0.020 ppm detection limit) and Cs (<1 g/L), which were the radioactive surrogates.

Table VII

The TCLP was performed on the first and second glasses produced from the trials because of the elevated levels of mercury in the feed. The TCLP was mainly performed to determine the leaching behavior of mercury since it was the only hazardous element of concern. As expected, mercury leaching was less than the detection limit of 0.008 ppm. The TCLP was not performed on the third glass, since the releases for the first two were below the detection limit.

Offgas System Analyses

The aqueous products contained in the wash bottles after the three trials were analyzed for chemical content. Each solution was analyzed using Inductively Coupled Plasma Emission Spectroscopy (ICPES) to determine the major cation concentrations, Atomic Absorption (AA) to determine the Cs content, Ion Chromatography (IC) to determine the major anions present, and cold vapor techniques were used to determine the mercury content. The components of each wash bottle are presented in Table VIII.

Table VIII

Results presented in Table VIII indicate that the mercury was not sufficiently being captured and converted to Hg_2S as anticipated. Total mercury captured for the trials were 0.03%, 1.97%, and 0.09%. The second trial captured the most mercury and it was the only solution which changed colors. The solution in the Na_2S bottle exhibited a blue-green color by the end of the trial. The higher concentration of Na_2S in the wash bottle in the third trial did not seem to help capture the mercury. Even though the Na_2S wash bottle was not effective in capturing the mercury, the NaOH bottle was successful in scrubbing the acid gases generated from the first bottle which is indicated by the general decrease in the pH and the capture of sulfate. No Ce was detected in the offgas system, which helps support the theory that it is all entrained in the glass waste form. Total Cs detected in the offgas system for each trial was less than 0.5% of the total Cs in the feed.

In order to more efficiently capture the volatilized mercury, the wash bottles were replaced with a mercury condenser kept below 10C and a KOH final wash bottle. The same glass formulations and melting schedules were used from the earlier tests. The resulting product was a homogeneous and durable glass similar in composition to the glass from trial #3 and with similar PCT results. Almost all of the volatilized mercury was captured and/or condensed. No mercury carried over into the KOH wash bottle. The mercury condensed was dissolved into an aqueous media for treatment on a mercury specific resin. This resin is capable of incorporating the mercury so it does not leach from the resin when subjected to the TCLP.

CONCLUSIONS

Crucibles studies with mercury contaminated SRS soil have shown that the soil can be converted to a durable, leach resistant glass wasteform. Optimum waste loading was determined to be 60 wt%, with 24 wt% Na_2CO_3 and 16 wt% $CaCO_3$ used as the glass forming additives. Attempts to capture and convert the volatilized mercury into stable Hg_2S were not as successful. Only 2% of the total mercury was captured in the offgas system. However, it was shown that by using conventional condensers, mercury could be successfully captured and/or condensed in the offgas system. Once captured, the mercury could be converted to a stable form on leach resistant mercury specific resin.

ACKNOWLEDGEMENTS

Funding for this research and development work was provided by the Department of Energy - Office of Technology Development under the auspices of the Mixed Waste Focus Area and Technical Task Plan SR1-3-20-04 and contract No. DE-AC09-89SR18035 with the U.S. Department of Energy.

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8-4

DEVELOPMENT OF AN INTEGRATED FACILITY SPECIFIC WASTE ACCEPTANCE CRITERIA AT THE SAVANNAH RIVER SITE

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ABSTRACT

The Department of Energy's (DOE) Savannah River Site (SRS) has developed an integrated facility specific Low Level Waste (LLW) Waste Acceptance Criteria (WAC) (1) designed to meet DOE Order 5820.2A (2) requirements and implement limits and assumptions established in a facility specific radiological performance assessment (RPA) (3) and safety analysis report (SAR) (4). As the SRS E Area Vault (EAV) facility is the first in the DOE complex to have an approved PA, the EAV WAC is the first in the complex to fully implement PA results into LLW acceptance criteria. Enforcement of PA derived limits ensures long-term compliance with DOE 5820.2A performance objectives. Integration of SAR based limits into the WAC ensures protection of workers and the public during facility operations. The composite WAC establishes the foundation for safe LLW disposal at SRS.

In developing facility specific acceptance criteria, SRS has necessarily stepped away from implementation of industry standard limits promulgated in the Nuclear Regulatory Commission's Title 10 CFR Part 61 (5). Development and implementation of facility specific LLW acceptance criteria provides for special consideration of unique waste streams produced in the site's numerous waste generating activities. Of equal importance, development of facility specific acceptance criteria has enabled SRS to employ a technically sound, graded approach for isolation and disposal of LLW. SRS's transition to fully compliant LLW disposal activities represented a significant change in operations and infrastructure. The facility and site specific approach applied in development of the acceptance criteria facilitated generator implementation and offered an optimal balance of technical requirements and cost efficient disposal options.

BACKGROUND

With issuance of Order 5820.2A in 1988, DOE established performance objectives to protect the public, intruders, and groundwater from LLW disposal activities. In response to the Order, SRS initiated the E Area Vault project to provide concrete barrier isolation between LLW and the environment. As DOE Headquarters prepared guidance for implementation of new performance assessment requirements, SRS proceeded with design and construction of three sets of disposal vaults. The Low Activity Waste Vault (LAWV) was designed for disposal of contact handled (<200 mrem/hr) waste. The Intermediate Level Tritium Vault (ILTV) was constructed for disposal and grout encapsulation of tritium contaminated wastes. The Intermediate Level Non-Tritium Vault (ILNTV) was designed for disposal and grout encapsulation of remote handled waste (>200 mrem/hr). A later addition to the EAV facility provided for shallow land disposal (non-vault) of slightly contaminated soil. Shallow land disposal of large highly activated components generated by the Department of Defense's Naval Reactor program has also been proposed.

The intent of both DOE 5820.2A and 10CFR61 is completion of a site specific PA in parallel with facility design so as to allow the PA to influence design requirements. Due to the near term need for additional disposal capacity and the infancy of performance assessment requirements, vault designs proceeded without the benefit of PA results. As such, the PA and SAR evaluated actual facility designs and operating practices. So as not to restrict waste disposal, the EAV PA was drafted to assess the maximum allowable inventory within constraints of DOE 5820.2A performance objectives. The PA initially considered 730 radionuclides that could potentially exist in SRS waste streams. Simple screening analyses based on half-lives and dose contributions reduced the number of radionuclides requiring detailed pathway analyses to 58. Detailed analyses further reduced this number, establishing maximum single isotope facility inventory limits for 37 radionuclides. As expected, the majority of the resulting limits were for long-lived radionuclides. Two notable exceptions were H-3 limits derived through the atmospheric pathway analysis and Cs-137 limits from the intruder scenario.

Enforcement of these 37 single isotope, performance based limits required utilization of sum-of-fractions inventory controls. Given the nature of changing missions at SRS, this approach for establishing and managing performance limits provided increased flexibility versus analyzing a fixed facility inventory.

The SAR necessarily evaluated potential exposures from a range of credible natural phenomena, criticality, and accident scenarios. An entry point for SAR preparation is establishment of a facility hazard classification in accordance with the Hazard Assessment Document (HAD) requirements defined in DOE Standard 1027 (6). Projections of facility inventories established a facility hazard classification and set an upper limit on total facility inventory.

As accident analyses typically involve individual or multiple containers, the SAR also required development of estimated container inventories. Due to the number of historical and ongoing nuclear processes at SRS, the composite of facility and container inventories included 48 radionuclides. The SAR produced container and facility inventory limits for each of these 48 radionuclides. The SAR also established container and facility inventory limits for fissile and fissionable isotopes and enforced regulatory driven transuranic limits.

DEVELOPMENT OF INTEGRATED WASTE ACCEPTANCE CRITERIA

The EAV WAC provided the vehicle with which to compile the 37 performance and 48 safety based limits and communicate resulting limits to waste generators. In developing the WAC, key objectives included:

- enforcement of PA and SAR limits, including controls on fissile and transuranic isotopes
- encompass majority of SRS waste streams, minimizing need for special deviations to acceptance criteria
- communicate key requirements needed to uphold assumptions made in the PA and SAR
- facilitate generator efforts to comply with DOE 5820.2A characterization and certification requirements

The composite of 37 performance and 48 safety based limits produced a total of 68 radionuclides requiring consideration for acceptance criteria radionuclide limits. In order to provide a common basis for evaluation, PA and HAD facility inventory limits were divided by the expected number of packages, producing limits based on the "average" package. Package based WAC limits were chosen over concentration limits, making implementation more compatible with generator strategies to characterize individual packages (i.e. dose-to-curie). Clearly, implementation of "average" package limits would have resulted in excessive requests for deviations to accommodate the many unique SRS waste streams. As such, three initiatives were undertaken to better refine the WAC and ease generator characterization requirements.

An extensive study of SRS radionuclide distributions was undertaken to better identify radionuclides of concern to vault operations. The study grouped the over 200 SRS radionuclide distributions into five representative waste streams (fission products, induced activity, tritium, uranium, and plutonium). A review of over 1,000 actual waste package inventories provided typical container loadings for each of the five representative waste streams. These typical container loadings were used to construct container and facility inventories which should reasonably bound future receipts. This study provided a basis for manipulating the "average" container limits to better accommodate SRS wastes.

A separate study was conducted to evaluate the range of special nuclear materials produced at SRS, many of which were retained in the PA and SAR analyses due to their long half-lives and dose contributions. A search of records maintained within the accountable material controls program identified upper bounding site-wide inventories for a number of special nuclear materials. These bounding inventories established that some special nuclear materials, such as curium and californium, were not produced in quantities that could impact long term vault performance. The study provided a basis for eliminating these radionuclides from the WAC.

A third study evaluated the fixed relationships of many radionuclides within SRS waste streams. In such cases, including the obvious parent-daughter relationships, limit controls for one radionuclide restricted the inventory of one or more associated radionuclides. As an example, the PA limit for Cs-137 and its fission product relationship with Cs-134 and Cs-135 necessarily limited the inventory of Cs-134/135 well below levels of concern in either the PA or SAR. This study provided the basis to further reduce the number of radionuclide limits retained in the WAC.

When paired with PA and SAR limits, the results of these three studies enabled the authors to reduce the number of radionuclides retained in the

WAC to 22 radionuclides. Comparison with the projected facility inventories permitted adjusting package limits to best accommodate the wide range of expected SRS radionuclide distributions while still upholding the basis for the facility limits. Figure 1 below identifies the basis documents for the 22 radionuclide limits in the final WAC.

Fig. 1

This strategy for development of acceptance criteria has been applied to specific waste streams, further "personalizing" the SRS LLW acceptance criteria. Examples of this unique application include earthen trenches for slightly contaminated soil and proposed mounded disposal of DOD's Naval Reactor components. In both examples, waste stream specific analyses present technically defensible disposal options to the more costly vault disposal. Implementation of these waste form specific disposal options dictates that the WAC impose more stringent waste form requirements to uphold assumptions in the PA and SAR analyses.

RESULTS AND CONCLUSIONS

Implementation of waste acceptance criteria for the E Area Vaults has brought SRS into compliance with DOE 5820.2A disposal requirements. The final set of radionuclide limits necessarily differs from those established in 10CFR61. As expected, intruder based limits were reasonably similar as environmental and geographical differences have limited impact on these scenarios. EAV limits derived from the groundwater analyses did vary significantly from limits established in 10CFR61. Again, this was expected given the humid SRS environment and the limited groundwater analyses supporting 10CFR61 limits. Table I below provides a comparison of some of these limits (EAV limits converted to concentration basis for comparison purposes).

Table I

Initial drafts of the acceptance criteria contained all of the composite 68 radionuclides identified in the PA and SAR. The concept of 68 radionuclide limits for each of the multiple disposal options in addition to the more stringent LLW controls promulgated through 5820.2A, brought generator implementation of the Order to a near stand still. The studies outlined above produced a streamlined set of 22 facility and site specific radionuclide limits. This approach substantially eased generator implementation while still enforcing Order requirements and critical PA and SAR assumptions.

Implementation of the PA and SAR limits has been further facilitated through a site-wide Waste Information Tracking System (WITS). WITS enables generators to select approved containers, create shipments, check container inventories against radionuclide limits, and print waste shipping manifests. Within the disposal facility, WITS provides capabilities for ensuring compliance with the entire range of container and facility inventory limits, including PA, SAR, criticality, transuranic, and greater-than-class C.

The facility and site specific approach requires added attention when accepting offsite waste shipments as some assumptions and evaluations would not apply to offsite generators. This increased scrutiny is accomplished through the generator certification process and evaluation of individual waste stream characterizations. The concept of developing acceptance criteria for specific waste forms has been applied to offsite generators as exemplified in the ongoing development of unique limits for Naval Reactor components.

The EAV WAC has been successfully implemented at SRS. The strategy taken has enabled SRS to enforce a multitude of requirements through a single set of radionuclide limits. The final WAC limits are unique to EAV and bound the majority of SRS waste streams. The approach can be applied to specific waste streams, providing for continued flexibility for future waste receipts. The concept of facility and site specific acceptance criteria does require increased monitoring to ensure key assumptions are upheld.

OPPORTUNITIES FOR FURTHER STUDIES

Many opportunities exist for continued enhancements of the acceptance criteria. Recently completed performance analyses of ashcrete and blowcrete to be generated in SRS's Consolidated Incineration Facility indicate this stabilized waste form can be disposed in earthen trenches. If approved and implemented, this waste stream specific acceptance criteria will optimize utilization of existing disposal capacity while enforcing DOE 5820.2A performance objectives. Other opportunities for waste stream specific analyses include vitrified waste forms and increased utilization of earthen trenches for organic and non-organic rubble and debris.

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LOOKING AHEAD: HOW TO MEET INCREASING WASTE CERTIFICATION REQUIREMENTS WITH LESS BUCKS

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ABSTRACT

Construction of the Savannah River Site (SRS) was started by the U.S. Government in 1950. The site covers approximately 300 square miles located along the Savannah River near Aiken, South Carolina. It is operated by the U.S. Department of Energy (DOE). Operations are conducted by managing and operating contractors, including the Westinghouse Savannah River Company (WSRC). Historically, the primary purpose of the SRS was to produce special nuclear materials, primarily plutonium and

tritium. In general, low-level radioactive and mixed waste is generated through activities in operations.

Characterization and certification of the low-level radioactive and mixed waste generated by forty seven independent operating facilities at The Savannah River Site (SRS) was completed in 1995. A generic waste characterization protocol and methodology enabled the efficient characterization of over 140 waste streams. This strong program base has allowed the site to meet increasing waste certification requirements cost effectively.

After approving the waste certification programs for 47 generators of low-level radioactive and mixed waste, Westinghouse Savannah River Company (WSRC) moved forward to implement a performance-based approach for assuring that approved waste generators maintained their existing waste certification programs during a time that they were also expanding those same programs to qualify for additional disposal and treatment facilities at SRS. WSRC implemented the Waste Certification Review Program, which is comprised of two sitewide programs, waste generator self-assessments and Facility Evaluation Board reviews, integrated with the WSRC Solid Waste Management Department Waste Verification Program Evaluations. The waste generator self-assessments ensure compliance with waste certification requirements, and Facility Evaluation Board reviews provide independent oversight of generators' waste certification programs. Waste verification evaluations by the TSD facilities serve as the foundation of the program by confirming that waste contents and generator performance continue to meet waste acceptance criteria prior to shipment to treatment, storage, and disposal facilities.

TRANSITION TO A WASTE CERTIFICATION REVIEW PROGRAM

The WSRC Solid Waste Management Department (SWMD) has implemented an integrated and cost effective Waste Certification Review Program for approved generators of low-level radioactive waste and low-level mixed waste. The program assures SWMD that approved waste generators maintain their approved waste certification programs.

New waste generator certification programs are approved through the established Waste Certification Approval Program. Once approved, waste generators undergo periodic review by the TSD under the protocol of the Waste Certification Review Program described below.

Both programs require review of waste generator certification programs for compliance with waste acceptance criteria of treatment, storage, and disposal (TSD) facilities. The Waste Certification Review Program integrates the use of existing SRS programs and waste verification to provide the assurance that approved generators maintain their required waste certification programs. It also integrates the Facility Evaluation Board (FEB) reviews, waste generator self-assessments and SWMD waste verifications. Waste verification evaluations coupled with FEB reviews and self-assessments provide a cost effective approach to waste certification for approved waste generators.

The foundation of the Waste Certification Review Program, the Waste Certification Waste Verification Program Evaluations, permits the SWMD to address the DOE Order 5820.2A requirement that waste generator certification programs undergo periodic audit by the operators of the facilities to which waste is sent. Waste verification evaluations are the method used to confirm that waste contents and waste generator performance meet the DOE Order requirements for waste certification.

Waste verification evaluations involve the inspection and verification of waste generator shipments by various methods to ensure consistency between the shipping documentation and the received wastes. As performed at most DOE and commercial facilities, waste verification confirms that the waste from a particular generator meets both general and specific verification criteria.

WASTE VERIFICATION PRACTICES

SWMD waste verification procedures check and document that waste generators' compliance with TSD waste acceptance criteria. These procedures require collecting and reporting inspection results for feedback on waste generator performance for use in the Waste Verification Program Evaluations. Procedures for visual inspection of low-level waste call for the selection of waste containers based upon generator shipments. SWMD establishes the frequency for verification of containers from waste generators. Verification frequency is adjusted according to the generators' performance. It is noteworthy that an initiative was implemented sitewide to improve the effectiveness of visual inspections of waste bags. Clear polyethylene plastic bags are being procured for improved visibility of waste bag contents. The new bags are being used by waste generators for improved inspection capability.

SWMD receipt inspection results are collected from all waste generators in the Waste Verification Program. Non-destructive examination (NDE), non-destructive assay (NDA), and chemical screening techniques similar to those employed at other DOE and commercial facilities will be phased into the program. Inspection and verification are results included in the Waste Verification Program. Nonconformances found during verification are identified, tracked, and corrected with waste generators prior to formal acceptance of the waste.

NDE, NDA and chemical screening are important features of any waste verification program. These methods provide a non-intrusive means of screening and quantifying radiological and chemical constituents in a waste container. SWMD compared NDE, NDA and chemical screening methods at other DOE and commercial TSD facilities and found the emphasis and level of scrutiny vary between facilities, based on federal and state regulatory requirements. Some features of radioisotopic assaying, radiography, and chemical screening are:

Radioisotopic Assaying - a method of identifying and quantifying radioisotopes in a given waste container. The purpose of the assay is to verify radioisotopic content to the extent and accuracy reported on the waste characterization forms for the waste stream being checked. Assaying may also be used to verify that low-level waste is below the transuranic threshold limit (100 nCi/g).

Radiography - a means of inspecting a container for prohibited items listed in the waste acceptance criteria. There are two technologies available, real-time radiography and digital radiography. Real-time radiography allows for a virtual "real-time" image of waste container contents and typically includes videotaping capabilities. digital radiography takes an image of the container contents and digitizes the picture such that it can be viewed and stored electronically.

Chemical Screening - analytical tests performed on representative samples of the waste. Typically, these analyses are performed to determine the presence of polychlorinated biphenyls, Resource Conservation Recovery Act or Toxic Substance Control Act controlled substances which, if found, would categorize the waste as mixed rather

than low-level waste. The specific analysis performed is dependent on the waste acceptance criteria. A field-deployable lab cart is used to perform gross screenings and, if necessary, samples are then sent to an analytical laboratory for more in-depth analysis.

These NDE, NDA and chemical screening methods are available both onsite and offsite; however, the associated costs and logistics vary greatly. After evaluating the various options and considering the requirements of DOE Order 5820.2A, SWMD selected a combination of assay, radiography and limited chemical screening to verify that incoming waste is in compliance with the waste acceptance criteria. Radioisotopic assay and digital radiography were the most cost effective and logistically feasible approaches. A portable kit is being obtained for chemical screening, and personnel are to be trained accordingly for use on a limited basis and under special circumstances to analyze various homogeneous waste forms.

PERFORMANCE BASED EVALUATIONS

Information on waste generators is collected and analyzed from receipt inspections, random visual inspections, shipment and characterization verifications, and NDE and NDA results when available. Other performance information, such as nonconformance reports, is considered as appropriate. Overall, this information provides a performance-based method of reviewing generator waste certification programs and a method for adjustment of waste verification.

In general, SWMD evaluates approved waste generators on a periodic basis. Waste Certification Assessment and Minimization procedures are developed to describe the evaluation process, evaluation criteria, reporting, and department required actions, as appropriate. In the evaluation process, waste verification results are discussed with waste generators, documented for trending purposes, and serve as a tool for continuous improvement. Performance areas requiring improvement or correction are identified on a case-by-case basis and presented to waste generators for resolution.

SWMD uses performance evaluation criteria to review the approved status of generator waste certification programs. The performance criteria are improved with its application to assure consistent evaluations and provide a mechanism to identify waste generator strengths and areas needing improvement. The performance criteria are structured to provide a graded method for evaluation of performance. Each waste generator's performance is graded as satisfactory, needing improvement, weak, or a strength area based upon an evaluation of the information collected over a specified evaluation period. The evaluation criteria define the basis for the grading of waste generator's performance from waste verification results.

CONCLUSION

After waste generator certification programs were initially approved and operating, SWMD needed a real-time waste verification program to confirm waste contents and evaluate generator performance. Waste verification is the foundation of Waste Certification Review Program, which includes waste generator self-assessments and independent oversight by the FEB. Random visual inspections of waste shipments, digital radiography, and radioisotopic assay are performed and results are evaluated with receipt inspection information to confirm generators are maintaining their waste certification programs. SWMD performs a review of approved waste generator status using information from the Waste Verification Program as part of the Waste Certification Review Program.

Therefore, as the SRS Waste Certification program matured with the initial waste certification assessments for 47 facilities at the SRS, innovative methods were developed that provided the following cost effective steps for the re-assessment and new assessments: 1) The Consolidated Incinerator Facility and the required periodic assessments were combined, 2) Waste generators were grouped by division programs to expedite completion of re-assessments and 3), Tiered oversight from three existing site assessment programs was established to gain economy of effort. In combination this will result in an annual cost saving of approximately $600,000 for the site.

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DRIVING DOWN THE COST OF CLEANUP: ER INITIATIVES AND INNOVATION IN PLACE AT THE SAVANNAH RIVER SITE

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ABSTRACT

This paper will describe how four (4) major initiatives developed at the Savannah River Site (SRS) have combined to save millions in their Environmental Restoration Program. The four initiatives are 1) streamlined sampling approaches; 2) approved standard designs and remedies; 3) minimization of investigative derived wastes; and 4) reduced groundwater monitoring requirements.

Specifically streamlined sampling approaches will show how early scoping of waste site characterization has resulted in more efficient sampling to reach a site profile. In addition, this initiative has generated the grouping of sites into characterization envelopes which can be used to further reduce sampling and facilitate the selection of a remedy. SRS will share its experience in this area.

The concept of the Approved Standard Corrective Action Design (ASCADtm) has been developed at SRS and makes full use of streamlining. This concept is an innovative way of defining characterization conditions and matching them with standard environmental restoration remedies. SRS has this concept endorsed by its regulators and is using it.

Minimizing the cost of investigative derived wastes is an area where SRS can show success in managing a complex-wide issue. Purge water from groundwater wells in radioactive and mixed waste burial grounds can be treated in a permitted on site facility. SRS will share this experience and has regulatory approval to move ahead with this initiative.

Quarterly groundwater well monitoring requirements were costing the SRS ER Program over $10 million annually. By focusing data needs and communicating with regulators, permit requirements have been changed to allow a more cost efficient program without significantly sacrificing environmental information. Other DOE Programs should benefit from this experience.

INTRODUCTION

This paper describes how four (4) major initiatives developed at the Savannah River Site (SRS) have combined to save millions in the Environmental Restoration Program. The four initiatives are 1) reduced groundwater monitoring requirements; 2) minimization of investigation derived wastes; 3) streamlined sampling approaches; and 4) approved standard designs and remedies.

Quarterly groundwater well monitoring requirements were costing the SRS ER Program $12 million annually. By focusing data needs and communicating

with regulators, permit requirements have been changed to allow a more cost efficient program without impacting key environmental information. The cost today of this program is now less than $8 million annually.
Minimizing the cost of investigation derived wastes is an area where SRS can show success in managing a complex-wide issue. Purge water from groundwater wells in radioactive and mixed waste burial grounds can be treated in a permitted on site facility. SRS will share this experience and has regulatory approval to move ahead with this initiative.
Streamlined sampling approaches will show how early scoping of waste site characterization has resulted in more efficient sampling to reach a site profile. In addition, this initiative has generated the grouping of sites into characterization envelopes which can be used to further reduce sampling and facilitate the selection of a remedy.
The concept of the Approved Standard Corrective Action Design (ASCADTM) has been developed at SRS and makes full use of streamlining. This concept is an innovative way of defining characterization conditions and matching them with standard environmental restoration remedies. SRS has this concept endorsed by its regulators and is using it.

GROUNDWATER MONITORING REDUCTIONS

Strategy

SRS-ER conducted an intensive review of the groundwater monitoring program required under the Resource Recovery and Conservation Act Part B Permit. The purpose of this review was to identify areas in which cost reductions could be made, while simultaneously remaining in compliance with the applicable regulatory requirements.
As a starting point, all cost elements were identified and organized into one of seven cost categories: task management, sampling, analysis, data management, report reparation, well maintenance, and DOE requirements.
The major stakeholders-the DOE customer, the South Carolina Department of Health and Environmental Control (SCDHEC), four internal Westinghouse departments, and several subcontractors-were identified and included in the process from the beginning. Through intensive discussions and negotiations, major cost savings opportunities were identified and implemented.

Streamlining

Over $2 million annually were saved by reducing the frequency of sampling and eliminating unneeded analytes. An additional $250,000 were saved by eliminating over 100 groundwater monitoring wells providing redundant information. This was done utilizing quantitative analysis of groundwater data. This statistical tool was developed as a computer algorithm by SRS with Lawrence Livermore National Laboratory and Georgia Tech. This tool allowed efficient use of groundwater data.

Cost Savings

The reduction in the frequency of sampling and the number of analytes resulted in a savings with respect to data management and report preparation. With fewer pieces of data to verify, validate, manage and report, subcontractor costs were reduced. Overall, these groundwater monitoring costs were reduced from $12 million in 1993 and $11 million in 1994 down to $7.4 million in 1995. This is shown in Table I.

Table I

INVESTIGATION DERIVED WASTES (IDW) MINIMIZATION AND TREATMENT

Strategy

IDW includes potentially contaminated environmental media such as well purge water, well pumping test and development water, drilling mud, and

soil drill cuttings. Also included in the definition of IDW is decontamination and rinse waters as well as equipment and personnel protective equipment. The SRS IDW management strategy is to minimize the quantity of IDW while cost effectively managing the IDW generated.
Two management programs are encompassed. IDW derived from contact with non-listed hazardous waste (non-listed IDW) and that derived from contact with listed waste (listed IDW).
SRS is implementing a program for the management of IDW from listed sources to achieve consistency with the Contained-In Policy. For both aqueous and non-aqueous IDW, a phased-in implementation approach is in progress. SRS treatment systems and infrastructure have been established for full implementation.
Permission from State and EPA regulators has been received to use an existing effluent treatment facility to treat mixed and radioactive purge water from monitoring wells. An approved air stripper treats non radioactive purge water.
IDW Minimization
In addition, SRS is pursuing innovative technologies and practices to reduce the volume of IDW that is generated. These innovative technologies and practices either have been or will be introduced to the EPA and the South Carolina Department of Health and Environmental Control (SCDHEC) for implementation at SRS.
Purge Water Reduction
Several methods are being evaluated to reduce the amount of purge water generated during sampling operations at SRS:
1) Micropurging uses dedicated flow sampling devices to selectively remove only water in the well directly opposite the well screen versus conventional methods of evacuating the entire column of water standing in the well and mixed in flow with stagnant water. 2) Two inch diameter wells can be used and require less water to be purged during well development and prior to groundwater sampling. SRS will continue to install two inch diameter monitoring wells where applicable. 3) The use of a casing packer to seal the upper portion of wells from the screened interval reduces the amount of stagnant water that requires purging prior to sampling. This technology is limited to four-inch diameter wells screened below the water table contaminants in concentrations above prescribed criteria.
Closed-Loop Purge Water Management
SRS is currently working with a prototype closed-loop purge water management system for aqueous IDW volume reduction. This system may eliminate the need for containerization of aqueous IDW and related transportation to a treatment system. The concept is based on capturing the water purged from a well during sampling into a collapsible tank dedicated and affixed to the well standpipe. Upon completion of the sampling event the captured water is returned to the well.
Less Invasive Drilling Techniques
SRS has conducted a successful field evaluation of a prototype device called the HydropunchTM which provides a less invasive technique for collecting groundwater samples. Direct push methods of data collection (e.g. cone penetrometer testing) and sonic drilling are being evaluated. These technologies provide real time geophysical, hydrological, and geochemical data that allow for design improvements in monitoring well networks. Improved monitoring well network designs will result in fewer

permanent wells being needed to obtain the required data, thus reducing the amount of IDW.

Cost Savings

Overall, the approval to use existing systems to treat IDW and initiatives to minimize generation had a major cost reduction of $2.5 million in FY '95 versus previous budget projections.

STREAMLINED SAMPLING APPROACHES

Strategy

SRS is now implementing streamlined sampling by scoping work plans with core team members and regulators. Every effort is made to evaluate and incorporate existing data where available. The Streamlined Approach for Environmental Restoration (SAFER) is being used at virtually all sophisticated characterization projects.

Data collection is driven by Data Quality Objectives determined by data users for future documents. There is a focus on end use determination and utilization of presumptive remedies.

The program utilizes non-intrusive sample techniques (ground penetrating radar, soil gas surveys, and magnetic surveys) and historical knowledge of known waste disposal practices at operable units. Preference exists to sample existing wells when possible to determine contaminants for groundwater and soil versus installing new wells.

Streamlining

SRS is implementing other faster characterization techniques as well. Expedited site characterization is now possible with mobile laboratories. Dynamic work plans contain built in contingencies for pre-approved decision-making. Better use of screening in-field techniques reduce laboratory costs. Use of cone penetrometer hydropunching and temporary wells reduces need for permanent groundwater wells. Appropriate analytical levels have been examined to reduce data needed.

Streamlining development of Remedial Investigations (RI) and Baseline Risk Assessments (BRA) has resulted in development of one combined RI and BRA document per operable unit. This yields savings on document production and reduced document reviews and revisions.

Cost Savings

These initiatives have combined to produce $3.35 million of savings in FY '95 versus previous budget projections. The average RI/FS cost per site is depicted in Table II.

Table II

APPROVED STANDARD CORRECTIVE ACTION (ASCADTM)

Strategy

The ASCADTM concept uses WSRC's remediation experience to develop a standard design for a specified set of contaminants and the approved remedy can be applied to any other waste site demonstrated to be within the specific envelope of contaminants. Table III graphically displays the ASCADTM process.

An ASCADTM is developed in the following way:

A single proposed plan and record of decision are planned for a grouping of similar sites.

A profile is established for a waste grouping that identifies the envelope of characterization conditions that match each proposed plan.

A proposed plan or corrective action plan is developed for the specified waste envelope.

Engineering parameters and technical requirements for the specific waste envelope are developed to ensure the record of decision will be

fulfilled for subsequent designs for remediation of waste sites that meet criteria of the specified waste envelope.

The regulators approve the proposed plan or corrective action plan for a lead site or groups/sites. Public review and comment is conducted and the record of decision is issued. Sites subsequently falling within the characterization envelope can be added to the record of decision. This concept is shown in Table III.

Table III

Streamlining

ASCADTM has the following potential benefits:

1) The overall characterization scope is streamlined by focusing on known contaminants of concern for a larger number of units 2) One treatability study workplan is developed and laboratory scale testing is conducted for a group of sites. 3) One CERCLA feasibility study, one proposed plan, and one record of decision (ROD) document are developed for a given waste unit grouping. 4) Only one SRS design review process (i.e., independent design review, value engineering study, etc.) is required. 5) One conceptual design (CD) package and one remedial design (RD) package are developed for a given waste unit grouping.

Cost Savings

The grouping of similar waste sites provides the opportunity to obtain regulator approval of a single or multiple steps of the remediation process for a given group of sites. The overall cost savings projected in 1996 and 1997 is $810,000.

CONCLUSION

SRS continues to pursue cost savings in all phases of environmental restoration. This constant improvement is necessary in an era of tightening budgets but often leads to streamlined ways of doing business that accelerate the real mission of cleanup.

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PRIVATIZATION OF WASTE MANAGEMENT PROGRAMS AT THE SAVANNAH RIVER SITE

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ABSTRACT

An aggressive effort to implement privatization initiatives at the Department of Energy's Savannah River Site (SRS) was initiated in 1991. This effort has resulted in the evaluation of potential contracts valued at $171 million, placement of $59 million in subcontracts, and capital cost savings of $142 million. This effort responds to evolving economic, political, and social changes, as well as a continuously shrinking DOE budget. Like other DOE facilities, SRS was conceived, built, and operated as a self-contained entity capable of fulfilling its mission in meeting the challenges of the Cold War. The move toward privatization required that SRS overcome the remnant Cold War barriers and required the development of cost-efficient, customer-defined products and services that also stimulated economic development in the communities surrounding SRS. This paper provides an overview of this initiative by presenting three specific case studies on waste management projects (Contaminated

Laundry Services, Mixed Waste Stabilization, and Sanitary Waste Disposal) that exemplify the successes and challenges of the SRS privatization initiative.

INTRODUCTION

SRS was constructed near Aiken, South Carolina, during the early 1950's to produce the basic materials used in the fabrication of nuclear weapons, primarily tritium and plutonium-239. Five heavy water reactors were built on the site to produce nuclear materials by irradiating target materials with neutrons. Support facilities, including two chemical separations plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities were also constructed at the site.

SRS has adjusted through the years to meet the major changes in defense requirements. All five of the original SRS production reactors are shut down, a reflection of the improved U.S. relations with the former Soviet Union. While production of tritium will not be necessary for many years, recycling and reloading of tritium (due to its 12.5 year half-life) to keep the nation's supply of nuclear weapons operational is a continuing site mission. In addition to the tritium mission, SRS is developing new missions for the site. These new missions include community outreach, assistance in the economic diversification programs of the surrounding communities,and privatization.

The overall privatization effort at SRS includes four major elements:

1. Outsourcing - fixed price subcontracting of goods and services that were traditionally provided by DOE or by Management and Operating Contractor (M&O) employees,
2. Technology Transfer - the transfer of DOE developed technologies to the private sector,
3. Asset Reuse - the transfer of DOE assets through sale, lease, permit, or other means to the private sector for reuse to satisfy commercial objectives, and
4. Dual Use Commercialization - making available DOE assets through sale, lease, permit, or other means to the private sector for dual use to satisfy DOE and commercial objectives.

Outsourcing, or fixed-price subcontracting, will be the focus of this paper. Of the $59 million in subcontracts that have been placed, three of the subcontracts in the waste management area are significant, both in terms of dollar value and as examples for the rest of the DOE Complex. Those three projects are:

1. Contaminated Laundry Services: This example involves the radioactively-contaminated laundry services that were once done on the site in a forty year old facility. That service has now been contracted off the site for a capital cost saving of $13 million. As an additional benefit, a new facility is being constructed in the local community, employing local personnel and transferring additional work from an older facility. The construction of this facility was encouraged by a DOE grant that was provided as an incentive for construction in the local area.
2. Mixed Waste Sludge Stabilization: A second major effort is the stabilization of 2,500,000 liters of a mixed waste sludge from a former nickel plating and aluminum forming operation. At a cost savings of $26 million, a vendor will come on site, erect a temporary facility, vitrify the waste, and close out the existing storage tanks.
3. Sanitary Waste Disposal: SRS has traditionally disposed of all sanitary waste in landfills located on site. Due to the low volume of

sanitary waste generated, the cost per ton for disposal was very high. In addition, landfill space was being depleted, and the cost for design, permitting, and construction of a new landfill appeared prohibitive. Subcontracting for hauling and offsite disposal of SRS generated sanitary waste has resulted in a cost savings of $103 million.

For each of these examples, a brief description of the old and new way of doing business, a discussion of the obstacles faced, and a candid evaluation of results will be presented.

CONTAMINATED LAUNDRY SERVICES

Background

As part of the original construction of the Savannah River Site, a laundry for contaminated clothing and equipment, primarily respirators, was built in 1952. The laundry was located near the center of the 800 square kilometer site and operated on three shifts, seven days a week. The facility processed an average of 225 metric tons of laundry and 10,000 respirators a month.

By 1992, several motivating factors existed to support the concept of replacing the laundry facility. First and foremost, the original facility had several deficiencies, including:

- Inability to separate radioactive and non-radioactive clothing
- Lack of a waste water treatment process at the facility
- An ergonomically obsolete design
- A roof structure that prohibited installation of modern handling equipment
- Location of the facility within a secure area, requiring security clearances for workers
- Severe heat stress conditions
- Lack of modern air emissions equipment

As a consequence of these deficiencies, providing future laundry services at SRS would have required either investing in a new laundry facility, estimated to cost $13 million, or privatizing the laundry services. The privatization option was explored and selected because it met DOE objectives to reduce capital expenditures and to stimulate regional economic development by creating an opportunity for businesses to locate or expand in the area.

Description of Privatization

Outsourcing the site laundry services was one of the first SRS attempts at privatization. The success of this initiative was highly dependent on "out-of-the-box" thinking and interactions with organizations who had successfully privatized traditional DOE owned operations. These organizations included the Hanford Reservation, the Idaho National Engineering Lab, and the Oak Ridge National Lab. Several details from those interactions were applicable, but the major lesson learned was that, if the facility is to be located off-site, it is important to locate the facility in the local region.

The first approach investigated a turnkey operation, with a commercial entity building and operating a facility on SRS land that would be leased to the successful bidder. Subsequently, an alternative life cycle cost analysis, done by the M&O Contractor, showed that it would be more cost effective to contract for a laundry service with a commercial radioactive laundry. This was initially supported by the fact that Interstate Nuclear Services (INS), a commercial firm in the radioactive laundry business for thirty years, had two facilities within a reasonable vicinity of SRS. As a result of this cost analysis, Westinghouse and DOE agreed to proceed with development of a specification and a procurement package for an off-

site service, as well as a Request for Expressions of Interest that was published in the Commerce Business Daily and several regional newspapers. Prior to publication of the Request for Proposal (RFP), DOE proposed to incentivize privatization of the site laundry services by offering an award of $500,000 or 10% of construction costs (whichever was less) to the successful bidder if he were willing to construct a new facility in one of the five counties surrounding the SRS. This incentive was authorized by the FY-1994 National Defense Authorization Act, which provided for funding to stimulate local economic development. The local construction option was not mandatory; the most cost effective proposal would still be selected. Respective bidders were notified of the incentive in the cover letter issuing the RFP, with a detailed explanation of the incentive package provided at the pre-bid conference.

Evaluation of Results

The incentive for local construction of the laundry proved to be a major factor in the procurement. Even though Interstate Nuclear Services has two other facilities near SRS (one located in Columbia, SC, about 80 kilometers away), they selected the local construction option and submitted the lowest bid for the services. The contract was awarded in January, 1995. INS has obtained all necessary permits from the State of South Carolina Department of Health and Environmental Control and Aiken County and has purchased land near Interstate 20, about 30 kilometers from the SRS, for the facility. The new facility will also service commercial industry, thus contributing to the local economy by providing jobs and increased tax base for the county.

In addition to benefits to the local economy, DOE and the SRS also achieved two major benefits. The first was an estimated $13 million savings in construction of a new laundry on-site, with additional operational savings possible when the new facility becomes operational. An additional positive impact for the SRS is a 90% reduction in the amount of low-level waste generated by the old laundry facility. The old facility used respirator boxes which required disposal; the new facility will utilize respirator bags which are reusable. In addition, use of the newer commercial process for handling lint and lint filter media also reduces generation of low-level waste.

MIXED WASTE SLUDGE STABILIZATION

Background

From 1985 to 1988, nickel-plating and aluminum- forming operations were performed in the M-Area of the Savannah River Site. These operations supported four reactors operating on-site during that time and resulted in the generation of 2,500,000 liters of a plating line sludge containing depleted uranium. Treatment of this sludge, stored in double-lined tanks, is required by the site's Federal Facility Compliance Agreement and is regulated pursuant to the Resource Conservation and Recovery Act (RCRA). Original plans for stabilization of this sludge involved construction of a Filtration/Stabilization Facility for initial treatment, followed by final stabilization, in a cement form, in a proposed Mixed Waste Treatment Facility. As part of the design effort for both the pre-treatment facility and the Mixed Waste Treatment Facility, a project team was formed to address the interface between pre-treatment and final treatment. A life cycle cost study done in 1991 by the project team indicated that on-site treatment by a vendor could be significantly more cost effective than construction and operation of new facilities by the M&O Contractor.

Description of Privatization

The results of the life cycle cost study convinced site management that privatization was, by far, the preferable option. Once again, success depended upon overcoming the obstacles associated with a new, innovative process at the site. These obstacles included determination of the DOE Orders that would be applied (this was the first instance in which DOE Orders were not applied in their entirety to an on-site activity), re-negotiation of milestones with the EPA and the South Carolina Department of Health and Environmental Control (SCDHEC), and demonstration that the vendor process would meet the requirements for the resultant waste form. The first obstacle was addressed during preparation of the specification for the Request for Proposal, where agreement was reached that only "applicable" DOE Orders would be applied. During conduct of the Operational Readiness Review, all DOE Orders were reviewed, recommendations on applicability were made, and DOE concurrence obtained. Major cost savings were made through the waiver of several DOE Orders, including DOE 4700.1, "Project Management" and DOE Order 6430.1A, "General Design Criteria."

Overcoming the second obstacle, re-negotiation of the Land Disposal Restriction-Federal Facility Compliance Agreement milestone for stabilization of the sludge with the EPA and SCDHEC, required five months. The third obstacle, adequate demonstration that the final waste form would meet requirements, was overcome by requiring that the vendor successfully perform treatability studies on actual plating line sludge before award of the final contract for treatment.

Evaluation of Results

Following successful treatability studies on the plating line sludge, a contract was awarded to GTS Duratek in November of 1993. The award is a fixed-price contract, with both bonus and penalty clauses for performance. In order to expediate performance, Readiness Assessments were split into three sections (sludge transfer, construction, and operation). The Sludge Transfer Readiness Assessment was completed in May, 1995, sludge transfer was initiated in June, and the first phase of sludge transfer was completed in September, 1995. The Construction Readiness Assessment was completed in July, 1995, construction of the temporary vendor facility initiated in July, and construction was completed in January, 1996. Operation is currently scheduled for May, 1996.

In addition to the benefits of providing jobs in the private sector, the main benefit was a cost savings for DOE. The original cost estimate for the pre-treatment facility, plus the Mixed Waste Treatment Facility, plus the treatment and disposal efforts, was $46 million. The contract cost for the vendor treatment is $16 million, with infrastructure support plus disposal costs of $4 million, for a total of $20 million. This results in a cost savings of $26 million.

SANITARY WASTE DISPOSAL

Background

The Savannah River Site had always disposed of its sanitary waste in landfills located on-site. Since the sanitary waste generation rate at SRS is low (approximately 36 metric tons per day), the landfill itself is small in comparison to a commercial landfill. However, the costs associated with landfill operation are roughly the same regardless of size (e.g., the same equipment, monitoring, and close-out requirements, paperwork associated with permit compliance and renewal, etc.). Because

these fixed costs are spread over a much smaller volume, the cost per ton for disposal at a small landfill is much higher. For SRS, this means a disposal cost of approximately $1150 per metric ton, compared to a commercial cost of $55-$110 per metric ton.

Description of Privatization

Based on the high unit cost of sanitary waste disposal at SRS, the decision was made by DOE and WSRC to advertise for sanitary waste disposal services. This decision was strengthened by the fact that the existing landfill would be filled in 1998, and the design, permitting, construction, close-out, and monitoring of a new, on-site landfill would be cost prohibitive, especially for such a small operation.

In March of 1994, a Request for Proposals for Sanitary Waste Disposal Services was issued. In July of 1994, four firms responded. In September, 1994, a fixed-price contract was awarded to CDS of Charleston, Inc. CDS is a small business and utilizes the Hickory Hill Landfill in Jasper County, South Carolina, for disposal. The contract has a fixed base price for establishing a route with (currently) 311 dumpsters. A fixed price per dumpster for adding and deleting dumpsters was also established in the contract. Thus far, the price has averaged $42-$45 per metric ton of waste.

Evaluation of Results

The major impact of the change from on-site handling and disposal of sanitary waste to a privatized service for pick-up and disposal was an estimated cost savings of $103 million. This savings included the capital costs for construction of a new landfill, operating costs of the landfill, transportation costs, closure costs, and costs for 30 years of monitoring following closure.

The response from Jasper County residents concerning the landfill initiative has been very positive. Besides creating jobs for the community, the revenue assists the county with landfill operation costs and lowers taxes for the residents.

CONCLUSIONS

The three initiatives discussed above have demonstrated that the DOE can achieve significant cost savings through privatization. The major savings, roughly $140 million, in the cases shown above result from shifting the cost of construction from DOE to the private sector. Although some savings result from operating efficiencies, the majority are savings in capital costs.

Other lessons learned include:

1. The importance of properly handling the human element of privatization. The employees who are potentially affected by a privatization initiative should be informed and involved to the maximum extent possible. Where possible, employment for the affected individuals should be assured, or at least considered, in the resultant subcontract. Other options include priority transfer of affected employees to other openings on site.
2. The importance of working with local and regional economic development organizations in the private sector, and doing it as early as possible in the process. Not only will these organizations assist in recruiting companies into the area, they can often provide tax and other incentives that will make the privatization initiative even more cost effective for DOE.
3. Contrary to widely-held beliefs, not all activities are automatically performed more cost effectively in the private sector. NEPA and other

permitting activities, safety analyses, and readiness assessments are often performed more cost effectively by the M&O Contractor. This may be due to lack of experience with the DOE Orders and implementing procedures that often cannot be waived when work is done on a DOE Site. Especially in smaller companies, the required expertise, or experience with a particular state regulatory agency, may mean it is cost effective to have the M&O perform the permitting. The main point is that a valid Make-Buy Analysis is necessary before assuming that the private sector is cheaper for any and all tasks.
4. Use of existing infrastructure is usually more cost effective than having the private sector provide duplicative services. However, the rates to be charged, and the methodology of measuring the services provided, are often major obstacles to be addressed.
5. An independent technical review is essential for certain technical services that are outsourced. At the time of bid preparation and contract award, details of the process are yet to be determined and a thorough review of the completed design is essential.
6. Finally, attitudes must be changed! Probably the biggest obstacle to privatization that must be overcome are the attitudes of many DOE and M&O employees towards doing things in a new way. This culture change is slow in coming, but it must occur for privatization to truly be successful.

Session 09 -- WASTE MANAGEMENT & REMEDIATION IN RUSSIA TODAY
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9-1
DECONTAMINATION OF THE TERRITORY OF BELARUS CONTAMINATED WITH RADIONUCLIDES AS A RESULT OF THE CHERNOBYL ACCIDENT
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ABSTRACT
The total area of the territory of Belarus contaminated with radionuclides amounts to 46.5 thousands km2. It is more than 20% of the whole territory. There are 3668 populated areas in it. After the Chernobyl accident Belarus has become a zone of ecological disaster. There are four special features of the Chernobyl accident:
1. Large amount of released activity from the reactor. According to the present-day knowledge this value accounts for 25.9 billions GBq. The main weight of the initial period has been a heavy burden to Belarus.
2. Large scale contamination and the prolonged period of the accidental event. The overall activity at the territory of Belarus amounts to 37 millions GBq. 137Cs, 90Sr, 239,240Pu are the main contributors to the radiation situation in Belarus. Belarus is characterized by the following data on contamination:
From 37 to 185 GBq/km2 - 29.9 thousands km2.

From 185 to 555 GBq/km2 - 10.2 thousands km2.
From 555 to 1840 GBq/km2 - 4.2 thousands km2.
More than 1840 GBq/km2 - 2.15 thousands km2.
Figure 1 presents the maps of contamination with radionuclides.
Fig. 1
3. Uneven character of contamination of territories. Even at small areas the density of contamination varies considerably. Thus, for instance, in the village Kolyban of the Gomel Region the density is in the range from 185 to 2590 GBq/km2. At the area of 1 m2 there have been the differences in the concentrations by a factor of 100. Even if dividing the portion of the soil into separate fractions with the weight up to grams, there is 10 times difference in activity between them. It is called the inhomogeneity phenomenon. There is high specific activity in the upper layer of the soil at the major part of the territory. This soil is itself the radioactive waste. Hence, these territories are the exposed radioactive waste disposal sites.
4. Radioactive contamination is combined with dangerous chemical contaminants. Thus, in Mogilev there is the high content of NO2 and H2S. Therefore, it is necessary to consider their combined effect.
The work on elimination of the consequences of the accident are carried out in accordance with the State program.
The main attention has been paid to building new populated areas. The people have been resettled on land. The large-scale industrial works on decontamination of the territories haven't been carried out. The following works have been completed:
1. Evacuation and resettlement of the population.
2. Local decontamination.
3. Disposal of radioactive wastes.
4. Division of the territory into the restriction zones.
415 populated areas in Belarus are resettled. The analysis shows that this solution cannot be considered as completely effective.
The work on decontamination of the territories, buildings and constructions bega`n in May 1986. It has been executed by subunits of chemical military troops and civil defense forces. Decontamination included local removal of the contaminated soil, demolishing the contaminated buildings and constructions.
Over the years the works on decontamination are conducted only for the most important objects: kindergartens, schools, hospitals, etc.
The main problem of Belarus is the lack of effective technologies for decontamination of the soils and large forest areas.
During 1991-1995 the clean-up of 150 objects, 360 ventilation systems has been conducted.
Depending on the density of contamination, the territory of Belarus is divided into five zones:
1. The evacuation zone - 30-kilometer zone
Sr-90 > 111 GBq/km2
Pu-238,241 > 3.7 GBq/km2
controlling and carrying regime.
2. The zone of immediate change of residence
CS-137 > 1480 GBq/km2
Sr-90 > 111 GBq/km2
Pu-238,241 > 3.7 GBq/km2
controlling and carrying regime.
3. The zone of subsequent change of residence

1480 GBq/km2 > Cs-137 > 555 GBq/km2
111 GBq/km2 > Sr-90 74 GBq/km2
3.7 GBq/km2 > Pu-238,241 > 1.85 GBq/km2
or dose > 5 mSv (0.5 rem)/year.

4. The zone with the right of changing the residence
555 GBq/km2 > Cs-137 > 185 GBq/km2
74 GBq/km2 > Sr-90 > 18.5 GBq/km2
1.85 GBq/km2 > Pu-238,241 > 0.74 GBq/km2

5. The residence zone with periodical radiation control
185 GBq/km2 > Cs-137 > 37 GBq/km2
18.5 GBq/km2 > Sr-90 > 5.5 GBq/km2
0.74 GBq/km2 > Pu-238,241 > 0.37 GBq/km2

General characteristics of contamination of Belarus are given in Table I.

Table I

The zone of evaluation of the population occupies 1987 km2. It is the territory from which the population has been evacuated in 1986. It includes forests - 877 km2, water bodies - 40.3 km2, marshes - 32.9 km2, arable lands - 940.1 km2, roads - 24 km2.

Let us consider the experience of decontamination of Belarus in details. Table II gives the list of technologies used for decontamination in Belarus. It presents the level of development of technologies.

Table II

The removal of the soil has been used for its clean-up. Both the general-purpose and special machinery have been used. The clean-up of the zones for recreation and residence is carried out in two stages.

- decontamination of the abnormal spots with high density of activity;
- decontamination of the whole area.

The exposure does rate is the criterion for carrying out the works. There are the following criteria, adopted in Belarus;

1. Territories of kindergartens, schools and hospitals 35 mR/h
2. Territories of private plots of land 40 mR/h
3. Interior rooms of kindergartens, schools and hospitals 25 mR/h
4. Working places in offices:
   - with the continued residence 50 mR/h
   - with temporary residence 200 mR/h

The internal surfaces of private buildings, kindergartens, hospitals, etc. are controlled with respect to contamination.

2.6 thousands of square kilometers have been excluded from use. 12 thousands of arable lands are exposed to erosion. Radionuclides are mainly in the upper layer of the soil at the depth from 5 to 80 cm. The value depends on the type of soil and its ploughing. In the clay soils the migration is low. In the sand soils it is faster.

The content of water soluble caesium in the soil amounts to some percents. Thus, for instance, in the Vetka area it is 3.8-5%. For the similar conditions water soluble strontium amounts to 11-26% and plutonium-239 to 2.8-9.5%. With the increase of the distance from the damaged Chernobyl reactor the fraction of mobile strontium increases too.

The average rate of migration of radionuclides in the soil amounts to 1 cm/year. It depends on the type of radionuclide. Strontium-90 has the largest rate of migration. And ruthenium-103, the least one. As to the migration rate, there is the following sequence of radionuclides; strontium-90; cerium-144; caesium-134; zirconium-95; caesium-137; ruthenium-106; ruthenium-103. Permeability of the soils to radionuclides can be presented in the following sequences: sand, sand loam, drained

peat bogs, loam. The rate increases with the increase of the moisture content.

It should be noted again, that the contamination of the soil is of uneven character. The general variation is shown in the map. The uneven character of the forests contamination has been shown earlier.

The area of the contaminated tracts of forests amounts to more than 19 thousands km2. Pine trees are the main species. They occupy 58% of forests, fir trees - 11%, birch trees - 17% and alder - 8%. The age of forests exceeding 60 years amounts to only 13%. Presently, there are the large areas of unthinned forests. The total forest reserve accounts for 804 millions m3. According to the density of contamination the damaged forests are divided into 3 types:

37-555 GBq/km2 - minor after-effects for forestry;

555-1480 GBq/km2 - restrictions for forestry;

more than 1480 GBq/km2 - prohibition of all types of felling the trees.

In 1988 the Polesje radiation forest reserve has been made. It is the most contaminated region with the total area of 1.4 thousands km2.

The main activity of the forest areas is concentrated in the forest litter. There are from 60 to 80% of radionuclides in it. It should be noted that the concentration of radionuclides in the forests is substantially larger than in meadows, marshes and other types of the territories. The largest specific activity is in the lower layers of the forest (litter, mosses, mushrooms). Bushes and grasses are the second in respect to activity. The trees have the lowest activity. In case of fires, the most active forest litter is burnt down. In addition, the radioactive particle, sorbed at the surfaces and included into the plant compositions, turn into gaseous and aerosol state. According to the data of the High Engineering Fire-Technical School, the time of life of such radioactive cloud is 7 days at the height up to 1.5 km, in the upper troposphere it is about a month and in the stratosphere, 1-5 years. New fire-extinguishing agents have been developed in Belarus. They possess high fire-retarding and fire-extinguishing efficiency. They preserve the fire curtain properties for more than 3 weeks. The consumption of the agent is 1.5-2 l/m2.

The contaminated forest has the following characteristic properties. The distribution of the activity with respect to plant organs is uneven. Table III gives the data on distribution of caesium in some parts of wood types.

Table III

Leaves and bark are the main accumulators of caesium.

Thus, it is necessary to solve the problem of processing the radioactive wastes so as to obtain the commercial wood.

Let us consider the coefficients of accumulation for various types of wood. The coefficient of accumulation is the ratio between the concentration in the wood and the concentration in the soil. Table IV presents the values of the coefficient of accumulation.

Table IV

It should be noted that there is the process of the increase of the contamination of the wood. For instance, the specific activity of the pine wood has increased twice in 1990-1992.

The volume of the wood radioactive wastes is given in Table V.

Table V

In Belarus the permissible level of contamination of firewood is 740 Bq/kg in respect to caesium-137.

There are more than 500 thousands m3 of wastes generated from pulling down wood buildings.
Therefore, there are two problems of contaminated forests. The first is connected with the decontamination of radioactive wastes, generated in the process of forestkeeping. The second one is connected with changing the contaminated forests into an ecologically safe system.
The following variants of decontamination are considered in Belarus:
1. Reducing in size and breaking up near the felling areas.
2. Collection at the open sites.
3. Thermal treatment.
Burning the wastes at the open places in forests is unacceptable owing to high activity of ash and smoke.
Presently, the possibility of using the technologies of the French company FROMATOME is considered. These technologies will make it possible to obtain clean gas and alcohol from the contaminated wood.
The feasibility study is now considered jointly between Byelorussian, Russian, Ukrainian and French specialists. The primary results are real in waste processing and changing the large forest areas into an ecologically safe system.
Another source of the radioactive waste should be mentioned. In heating the villages, 18300 tons of ash are generated yearly. Its specific activity reaches for 137CS - up to 550 KBq and for 90Sr - up to 23 KBq/kg. It can lead to the secondary contamination. Therefore, the technologies of the centralized collection of ash from the population is organized. Ash is transported to the processing and disposal sites.
The technology of immobilization of the ash wastes has been developed on the basis of phosphogypsum. It is the by-product of fertilizer production plants.
The contamination of water basins with radionuclides is defined by two regions.
The first southern region includes the basins of the Pripyat, Braginka, Vita rivers. In the initial period of the accident there have been high dose rates at the expense of short-lived radionuclides. Downstream of the Pripyat and Bragin rivers the region is characterized by contamination density of 137CS from 3.7 GBq/km2, 90Sr, 185 GBq/km2 and plutonium 3.7 GBq/km2.
The second northern region includes the basins of the Dnepr, Sozh, Iput, Besed rivers. The radiation situation has been formed here mainly at the expense of atmospheric precipitations. 137Cs contamination density reaches up to 5550 GBq/km2, 90Sr to 37 GBq/km2.
Immediately after the accident the permissible concentration (4.10-10 Ci/l in the downstream water of the Pripyat river has been considerably exceeded for the limited contingent of population but at the end of May 1986 the concentration had been reduced to 10-10 Ci/l.
The average annual 137Cs concentration in the Pripyat, Dnepr, Sozh, Iput rivers is lower than the permissible levels and does not exceed the Republican control levels (Republican control levels are 5.10-10 Ci/l).
If in the initial period of the accident the contamination of the water has been determined by the deposition of the radioactive aerosols on the water surface, later the level of contamination has been determined by wash-off from the watershed surface, exchange with bottom sediments.
According to the data of the State Department for Hydrometeorology, the total wash-off of 137Cs by the five above rivers has amounted to: 22900

GBq in 1987; 12580 GBq in 1988; 7030 GBq in 1989; 3034 GBq in 1990; 2664 GBq in 1991.
The main contribution has been obtained at the Sozh river. Its wash-off has accounted for 10360 GBq in 1987; 5180 GBq in 1988; 2220 GBq in 1989; 925 GBq in 1991.
The analysis of the wash-off characteristics has shown that in 1991 the activity has been determined by caesium-137. The presence of anomalies should be noted. Thus, in the Kulazhin village on the shore of the Nesvich river the contamination with strontium does not exceed the contamination with caesium and amounts to 1130 GBq/km2. The substantial contribution from 106Ru and 144Cs is seen (225 and 52 GBQ/km2, respectively). In this area the increase of density of contamination has been observed: 5660 GBq/km2 in March; 19280 GBq/km2 in October. Similar phenomena have been seen in other areas.
The main fraction of activity is in the bottom sediments. They have a specific activity much higher than water. It amounts to 40000-700 Bq/kg and 4000-10 GBq/kg for 137Cs and 90Sr, respectively.
There are also the seasonal changes.
The works on protection from the water transport of radionuclides and decontamination of bottom sediments are not carried out in the Republic.
Large works on decontamination of the industrial equipment are conducted. 390 ventilation systems have been decontaminated. The total cleaned-up area amounts to 57 thousands m2. More than 1300 pieces of industrial equipment are subject to decontamination.
In Table VI the detergents for decontamination of such systems are given.
Table VI
The works on creation of the pastelike compositions for decontamination of unpainted surfaces are carried out. They can decontaminate 97% of 137Cs and 95% of 90Sr. In the process of decontamination liquid and solid radioactive wastes are generated.
In the Institute of Radioecological Problems of the Academy of Sciences of Belarus two installations have been designed for treatment of liquid radioactive waste.
The first installation is operated on the principle of chemical co-precipitation followed by separation of solid and liquid phases.
The second installation takes the evaporation of rotatable liquid flow as the basis.
The process of volumetric evaporation is carried out in this apparatus. Liquid is preheated and then the evaporation of whirled liquid occurs, when pressure is reduced. It should be noted that such eddy evaporator is the separator at the same time. It decontaminates liquid radioactive wastes of any composition.
Figure 2 gives the basic elements of the installation for processing liquid radioactive wastes. The installation embodies the following principle:
Liquid is heated and conveyed by the tangential channel into the evaporation chamber. On the certain radius of rotation liquid cones to the boil. Vapor bubbles move to the axis of the periphery. The main technical characteristics of the installation are as follows:
Fig. 2
1. Initial temperature of liquid radioactive wastes is 278-298 K.
2. Capacity of 0.05 - 0.01 kg/sec.
3. Operating conditions for pH are not specified.
4. Decontamination factor is 105.

Let us consider the problems of sewage waters. The selective investigation of the radiation situation of the sewage works have been conducted. In the fields of filtration of the Slavgorod town the specific activity of 60000 Bq/kg has been found out.

It should be noted that the state of radionuclides in the fields of filtration differs from that in the soil. The mobile form of 137Cs amounts to 21-46% and 90Sr to 68-85%.

The preliminary works on decontamination of sewage works are carried out. In 1986-1988, 69 interim repositories have been created for disposal of radioactive wastes. The works on determination of radiation danger of these repositories are conducted. The system of radiation monitoring has been created. The observation wells have been made around the repositories. There is no danger now. Mathematical modeling of Belarus and Sweden specialists have shown that in the future the ingress of strontium into the aquifer is possible. There are seven repositories for disposal of the radioactive wastes generated in the process of decontamination at the territory of the zone of changing the residence. The volume of each of them is 30-50 thousands m3. presently, 120 thousands m3 are without backfilling. 120 thousands m3 of repositories will be constructed to 2000.

Work on selection of sites for perspective disposal have been carried out. The principles and the methods of shallow ground and surface repositories have been worked out. Six types of sites have been assigned, ranging from favorable to useless. Ecologically safe sites have been found. The following areas have been chosen for radioactive waste repositories:

The Gomel Region:

- Narovlya area - "Karpovichi" sites;
- Khojniki and Bragin area - "Babchin" - 4 site;

The Brest Region:

- Stolin area - "Olmyanskaya Koshara" site;
- Luninets area - "Dobraya Volya" site.

Figure 3 shows the location of the repositories on the map. The codes of practices have been worked out for carrying out the work on decontamination. They determine the sequence and technologies for work on decontamination, demolition of buildings and waste management.

Fig. 3

The works on decontamination need enormous expenditures. In the laboratory of Radioecological Problems of the Institute of Radioecological Problems of AS B, complex methods have been developed. They determine the expediency of carrying out the work. The cost-benefit analysis is the basis of the methods.

Figure 4 gives the schematic diagram of the program of evaluation f the efficiency of decontamination.

Fig. 4

With the help of the program the following operations can be made:

1) Evaluation of the efficiency of decontamination of soils and the ground;
2) Comparison with the alternative technologies and the choice of the optimal one;
3) Comparison of the optimal technology of decontamination of soils and the ground with the alternative methods of rehabilitation and the choice of the best variant.

The evaluation of the efficiency of the technologies of decontamination of soils and the ground has been carried out according to the criterion:

$$Ed = R + MS = Cd \max,$$

where Ed - efficiency of decontamination, roubles/km2;
R - reduction of the integral detriment to health of the population as a result of decontamination;
MS - Additional production of products attributed to the decontamination of the territory, roubles/km2;
Cd - cost of decontamination, roubles/km2,
DF - final detriment after decontamination;
Dp - detriment attributed to exposure to radiation of operational personnel.

Table VII presents the economic efficiency of decontamination of agricultural lands with the help of removal of the soil by deep ploughing to the depth of 70 cm.

Table VII

Removal of the soil is more efficient than deep ploughing.

Decontamination has the positive efficiency only with utilization of wastes and conversion into energy. In addition, the economic effect is smaller by a factor of thousand than the removal of the soil.

On the basis of the above the following conclusions should be made:

1. In Belarus the tactics of observation and changing the residence have been mainly carried out. The large-scale decontamination of the territories has not been carried out.
2. The scales of decontamination are to be determined by the cost-benefit analysis methods. The criteria of the density of decontamination, dose rates, etc., are only the internal parameters of the methods. They do not determine the expediency of the decontamination works.
3. Taking into account high cost of decontamination, it is advisable to create the international standard computational programs for "cost-benefit" analysis.
4. It is necessary to unite international efforts for development of the efficient technologies on clean-up of the soil, forest areas and water systems.
5. It is advisable to create the international data base on technologies and methods of decontamination of large territories.

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9-5

DEVELOPMENT OF A SOPHISTICATED COMPUTER BASED DATA SYSTEM FOR EVALUATION OF THE RADIATION LEGACY OF THE FORMER USSR AND SETTING PRIORITIES ON REMEDIATION AND PREVENTION POLICY

(Project of International Science and Technology Center # 245)

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RADLEG

Phase 1:

Creation of Simple Operational Data Base Able to be Connected to GIS Describing Currently Available Information on Radiation Legacy of the Former USSR

Phase 2:

Public Accessible Prototype Data System Including Radiation Legacy Data Base Linked to GIS

ABSTRACT

The objective of the Project is to create a public accessible database linked to geographical information system (GIS) that will describe availableinformation on the radiation legacy of the former Soviet Union (FSU). This system will be developed so as to aid policy makers in two principle areas:

1. identify and set priorities on radiation safety problems and
2. provide guidance for the development of technically, economically, publicly, and institutionally sound policies to reduce the impact of radioactively contaminated sites.

Over the past 50 years, vast quantities of radioactive waste (RW) and numerous radioactively contaminated sites resulting from the production of nuclear weapons and civilian use of nuclear energy have accumulated in several countries. Although there are studies and cooperative efforts on aspects of this overall issue, there has been no systematic, comprehensive and interdisciplinary effort to exchange information and coordinate activities, to achieve technically sound and cost effective management, and cleanup of the RW.

The aim of this project is to provide usable information that can aid to make decisions for radiation hazard management. It will aid also in conducting risk estimate studies, analyzing the transboundary and transjurisdictional aspects of waste and spent fuel transportation, determining future land uses, and evaluating radionuclide material flows through the environment.

In accordance with the funding directives of the ISTC, this project has been separated into two phases, the first phase will last one year and is being carried out now.

Phase 1 consists of :

the development of the methodology necessary for a systems approach to the creation of the database and GIS technology;
collection and analysis "first order" data;
determination of the computer application methodologies;
development of a simple operational prototype data system linked to 2 or 3 GIS examples.

Phase 2 will consist of :
an evaluation of the completeness of data collected during Phase 1;
collection and analysis additional, "second order" data including socio- institutional, geographic, demographic, and climatic characteristics allowing to develop a public accessible data base and GIS;
development of the map blocks as GIS components;
creation a public accessible data system including data base and GIS on radiation legacy of the former USSR.

DESCRIPTION OF THE DATA

The data are separated into three classes, rated according to its accessibility: sector-overview data, first-order data, and second-order data.

Sector-overview data represents a review of information about radioactive waste accumulated and about the plans for nuclear materials management available in organizations taking part in this project. Each participating organization have prepared an overview for its sector of professional activity; for example, nuclear test sites, civilian NPPs, reprocessing plants, etc. General Sector Overview based on materials by participating research groups in participating institutions (along the whole project) is prepared and delivered to ISTC.

First-order data represents

1) physical, chemical, technical and general characteristics of the facilities of interest, and

2) socio-institutional data

The data from facilities are based on information currently available in organizations participating in the project and include specific archives, data published in the literature as well as data contained in existing local databases.

Socio-institutional data are based on polls that were conducted by INTEST in 1993 in regions of major social anxiety and concerns about local radioactive problem (Urals, regions around Chernobyl, Russian North).

Second-order data embraces

1) additional information on the physical, chemical, technical and general characteristics that will be considered as necessary after evaluation of the completeness and quality of the first-order data (Sources of this additional information will be unclassified archives and additional calculations and verification of the models used for quantitative estimations.),

2) data on the socio-institutional characteristics pertinent to the radioactive objects and not covered during Phase 1, and

3) data on geographical, geological, demographical, and climatical characteristics that permit the database collected during Phase 1 to be organized into a Geographical Information System (GIS).

The public accessible prototype database and GIS will be developed on computer centers located at VNIICHT and RRC "Kurchatov Institute". All data that is collected and stored on this system will be accessible to the public via the networks RELCOM / EUNET / INTERNET.

EXPECTED RESULTS

A public accessible prototype database system on the FSU radiation legacy. The database will include:

- available physical, chemical, technical, and general information about civil and military objects of the nuclear fuel cycle, obtained from all areas of the FSU; and
- socio-institutional, geographic, geologic, demographic and climatic characteristics of civil and military objects of the nuclear fuel cycle and surrounding areas.

A computerized GIS linked to public accessible prototype database system on the FSU radiation legacy. GIS will include three categories of multilayer thematic maps:

- Survey (3 semi-global for the whole area of the FSU);
- Highlights (20 regional with description of radiation situation);
- Radioecology (20 local with description of radiatively dangerous objects).

A video information about some objects of radiation legacy

SCOPE OF ACTIVITIES

To reach the goals of identification and setting priorities on radiation safety problems, analytical tools must be developed. The proposed study will include the following three activities:

Activity 1: Methodology of Systems Approach to Database Creation

The main part of this activity have been carried out in the first phase of the project (collection of key reports and information about radiation legacy of the FSU, descriptions of the most serious problems caused by this legacy, as well as current and expected efforts to cope with them, documentation and examination of existing methodologies and experience with assessment data needs and data base creation).

The outcome of this activity is the development of the version of data base structure necessary for the Activity 2 realization.

Activity 2: Data Collection and Analysis

This activity will entail the collection and verification of the data for providing a review of extent and seriousness of radiation problems left by the cold war and accidents at nuclear facilities of the FSU, as well as those caused by its disintegration.

The basic idea is to describe the radiation risk sources: potential radioactive releases, contained radioactive wastes, and contaminated areas. The sources must be grouped into five categories: military, civilian, front end of the fuel cycle, back end of the fuel cycle, and non-reactor radioactive waste burial sites.

As a first step of the Project first phase, the Sector Overview has been prepared on what data are currently and potentially available from the Russian Federation and the other NIS. Then the process involve the collection of "easily" available data ("first order data"), finalized by the creation of a simple operational data base.

The second phase will start from the evaluation completeness and quality of the data collected during the first phase. The purpose is to identify gaps in the data and provide this information together with the requirements for additional data ("second order data") collection.

The second phase will result in the collection of the second order data and their analysis.

Activity 3: Generation of Public Accessible Prototype Data System

The collected and verified data will be organized in the computerized public accessible data system.

This activity involves the following steps:

The data base on the radiation legacy of the FSU is being built on a number of existing data bases that are already in use in the NIS or in the phase of creation.

An analysis is being made of computer methodologies currently used for these specific data bases.

The task to determine which GIS methodologies might be effectively linked to the data base must be done. This is followed by the development and demonstration the feasibility and capabilities of the conceptual design.

Realization a conceptual design of a public accessible computer-based data system of the radiation legacy of the FSU by multilayer map blocks development and input the collected data into these blocks will be the ultimate goal of the Project second phase.

TECHNICAL APPROACH AND METHODOLOGY

Nineteen main sectors of the nuclear fuel cycle, both civilian and military, are framed for this project and are investigated by participating institutions, which have responsibility for collecting data on the radiation legacy. Each institution obtain and analyze the information in those sectors closely connected to its professional activities.

The database describes the types and characteristics of objects containing radioactive materials, physical, chemical and technical characteristics as well as geographical, geological and socio-institutional data on the radiation legacy. The data will be organized in a public accessible computerized database which will be linked to a multilayer Geographic

Information System (GIS) consisting of semi-global, regional and local map blocks.

The semi-global block represents a survey GIS saturated by data which give an overview of radiation legacy objects arrangement. The regional blocks represent a highlights of the FSU areas containing the significant radiation sources. Among these the first line constitute the regions as follows: Kola Peninsula; Novaya Zemlya; Tomskaya Oblast; Urals; Center of Russia European Part (at first areas exposed by Chernobyl accident pollutions and NPPs location sites); Semipalatinsk; North-West (Lithuania, Leningrad Oblast); Pacific Navy.

The local blocks will give a detailed radioecologic picture for sites where the major amounts of radionuclides are arranged.

The map blocks being created will be saturated by factographic (attributive) data collected during the first and second phases of Project efforts.

The project activities will be carried out in nineteen Participating Institutions of the Russian Federation:

Name -- Acronym

Chief Coordinating Institution - All-Russian Scientific Research Institute

of Chemical Technology -- VNIIKHT

Russian Ecological Center of Influence Assessment -- RETSOV

Ministry of Russian Federation for Atomic Energy -- MinAtom

All-Russian Scientific Research Institute for Non-Organic Materials -- VNIINM

Mining-Chemical Combine -- GKHK

Radium Institute -- NPO "RI"

All-Russian Scientific Research and Designing Institute for Production Engineering -- VNIPIPT
All-Russian Scientific Research Institute for Nuclear Power Plants Operation -- VNIIAES
Russian Academy of Sciences -- RAS
Central Economic-Mathematical Institute of -- RAS TSEMI
Institute of the Geology of Ore Deposits-- IGEM
Institute of Global Climate and Ecology -- IGKE
Institute of Geochemistry and Analytical Chemistry -- GEOKHI
Nuclear Safety Institute IBRAE
Russian Research Center "Kurchatov Institute" -- RRC "KI"
Russian Ministry of Defense -- MO RF
Scientific & Industrial Firm -- "Radon"
Ministry for Environmental Protection and Natural Resources of Russian Federation -- Min Priroda
State Institute for Applied Ecology -- GIPE
Moscow State University -- MGU
Interindustrial Innovational Research Association "Technological Risk and Human Safety" -- IIRA "INTEST"
Consulting organizations:
All-Russian Scientific Research Institute of Experimental Physics -- VNIIEF
All-Russian Scientific Research Institute of Technical Physics -- VNIITF
Industrial Firm -- "Mayak"
Siberian Chemical Combine -- SKHK
Kola Science Center -- RAS.
Institute of the North Industrial Ecology Problems -- NIEP
Foreign Collaborators:
British Nuclear Fuels, limited -- BNFL
Dutch Nuclear Corporation -- DNC
International Institute for Applied System Analysis, Austria -- IIASA

9-6

RADLEG - DEVELOPMENT OF A SOPHISTICATED COMPUTER BASED DATA SYSTEM FOR EVALUATION OF THE RADIATION LEGACY OF THE FORMER USSR AND SETTING PRIORITIES ON REMEDIATION AND PREVENTION POLICY

N. Egorov
A. Iskra
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Minatom, RAS, RRC-"KI," IIASA

ABSTRACT
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2) data on the socio-institutions characteristics pertinent to the radioactive objects and not covered during Phase 1, and
3) data on geographical, geological, demographical, and climatical characteristics that permit the database collected during Phase 1 to be organized into a Geographical Information System (GIS).
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2) a computerized GIS linked to public accessible prototype database system on the FSU radiation legacy. GIS will include three categories of multilayer thematic maps:
  Survey (3 semi-global for the whole area of the FSU);
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Realization of a conceptual design of a public accessible computer-based data system of the radiation legacy of the FSU by multilayer map blocks development and input of the collected data into these blocks will be the ultimate goal of the Project's second phase.

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Table I
Table II
Table III

Session 10 -- INTERNATIONAL PROGRAMS & PROGRESS
Co-chairs: John D. Hurley, WSRC
Michele Laraia, IAEA
10-1
THE IMPLICATIONS OF POLITICAL, LEGAL, AND LEGISLATIVE INITIATIVES UPON THE DOE SPENT
NUCLEAR FUEL PROGRAM STRATEGY
Mark W. Frei
U.S. Department of Energy

Elizabeth C. Saris
Adam E. Barringer
Science Applications International Corporation

Ralph Stein
Golder Federal Services, Inc.
ABSTRACT
The DOE Spent Nuclear Fuel Program has made considerable progress over the past year in defining a path forward for the management and disposition of DOE-owned spent nuclear fuel. Key elements of the program strategy include 1) assuring safe existing conditions, 2) achieving safe interim storage, and 3) preparing for ultimate disposal in a geologic repository. DOE-owned spent nuclear fuel is spread throughout the DOE Complex in a variety of storage modes and conditions which have been factored into an integrated program management strategy. However, legal challenges and federal and state agreements, National Environmental Policy Act (NEPA) documentation and legislative initiatives have the potential to influence the future direction of the program strategy. Examples include: the outcome of ongoing or new litigation including consent orders; NEPA documentation and records of decision concerning DOE-owned spent fuel storage, processing, and ultimate disposition; and proposed legislation that may significantly affect the management of DOE-owned spent nuclear fuel. This paper seeks to examine the impacts of these factors and other emerging programmatic issues on the current and future course of the program.

INTRODUCTION

The December 1994 DOE-Owned Spent Nuclear Fuel Strategic Plan (1) (to be updated in 1996) articulates the mission, vision, objectives and strategies for the management of DOE-owned spent nuclear fuel spent nuclear fuel (SNF). The DOE Spent Nuclear Fuel Program has made considerable progress over the past year in further defining a path forward for the management and disposition of DOE-owned SNF. Key elements of the program strategy include 1) assuring safe existing conditions, 2) achieving safe interim storage, and 3) preparing for ultimate disposal of approximately 2,800 MTHM of DOE-owned spent nuclear fuel in a geologic repository. DOE-owned spent nuclear fuel is spread throughout the DOE Complex in a variety of storage modes and conditions which have been factored into an integrated program management strategy. This strategy is outlined in the Programmatic Spent Nuclear Fuel and Idaho National Engineering Laboratory Waste Management and Environmental Restoration and Waste Management Programs Environmental Impact Statement Record of Decision (2) which identifies "regionalization by fuel type" as the management alternative to be implemented for spent nuclear fuel in the DOE Complex.

Specifically, aluminum-clad spent fuel will be managed at the Savannah River Site and stainless-steel/zircaloy-clad spent fuel, with the exception of the production fuel at the Hanford Site, will be managed at the Idaho National Engineering Laboratory. These fuels would be stored in existing facilities with modifications and upgrades as needed until new interim dry storage facilities become operational. Ultimately, DOE-owned spent fuel would be transferred to a permanent geologic repository unless otherwise dispositioned. Current agreements with States and Consent Orders have been accounted for in the program strategy. Further details of the DOE spent nuclear fuel program strategy are articulated in both the Interim Storage Plan (3) and Program Plan (4) issued by the Office of Spent Fuel Management in November 1995.

Further programmatic guidance concerning the implementation of the "regionalization by fuel type" strategy is found within the DOE-Owned SNF Program Plan. This document is supported by other publications, such as the DOE-Owned SNF Interim Storage Plan, which provide greater detail concerning the various elements of implementation. The storage plan acts to integrate those national and site-specific plans needed to effectively manage DOE-owned SNF during its interim storage. The scope of the DOE-Owned SNF Interim Storage Plan includes the integration of planning and implementation activities associated with the handling, conditioning, storage, and transport of DOE-owned SNF.

Despite the establishment of a clear waste management strategy and set of objectives as announced in the mentioned plans, the Department continues to recognize the increasingly dynamic climate in which future waste management policy decision shall be made and to identify influences upon these established implementation activities. This paper seeks to examine the implications of a number of influences, specifically, pertinent political, legal, and legislative initiatives upon DOE's spent fuel program strategy. These influences include: the outcome of ongoing or new litigation and State and Federal government agreements concerning spent nuclear fuel; proposed National Environmental Policy Act (NEPA) documentation and Records of Decision concerning DOE-owned spent fuel storage, processing, and ultimate disposition; and legislation that may significantly affect the timing of repository availability.

LEGAL ACTIONS/STATE AND FEDERAL GOVERNMENT AGREEMENTS

The key agreements impacting the DOE spent fuel program include the Hanford Tri-Party Agreement, the DOE/Navy/Idaho Agreement on Spent Fuel and Nuclear Waste, and the DOE/Public Service of Colorado Agreement (In Principle) to Manage Spent Fuel at Fort St. Vrain. Each of these agreements sets forth terms and conditions that affect the manner in which DOE is able to manage its spent fuel in the complex.

The Hanford Tri-Party Agreement (5)

Over the past few years, the mission of the Hanford Site has changed from nuclear material production to environmental cleanup. The framework for this cleanup effort is outlined in the Hanford Federal Facility Agreement and Consent Order, commonly referred to as the Tri-Party Agreement signed by the Environmental Protection Agency (EPA), the Washington State Department of Ecology, and DOE. The Hanford Site manages the largest amount of DOE-owned spent nuclear fuel (by weight) with a current inventory of approximately 2,133 MTHM, representing approximately 81% of the Department's entire spent nuclear fuel inventory, the vast majority of which is production fuel.

In the early years of the Hanford Site, a number of onsite facilities were utilized for the processing and storage of spent production fuel. From the early 1970's through 1987, PUREX, a major processing facility, remained off-line while undergoing necessary refurbishment and awaiting restart. During this time, the K Basins, originally designed for the temporary storage of K-Reactor fuel, served as temporary storage for spent production fuel generated at the now inactive N-Reactor. Following PUREX's return on-line in 1988, much of the N-Reactor spent fuel was processed as planned. However, a 1992 Department action permanently deactivating the PUREX facility left approximately 2,100 MTHM of N-Reactor spent production fuel in the K Basins with no immediate strategy for removal.

The Tri-Party agreement set milestones and target dates to encapsulate fuel and sludge from K Basin East, treat and dispose of contaminated water, and to remove all fuel and sludge from the K-Basins by 2002. In the Department's strategy, the completion date for the removal of spent production fuel from the K Basins is scheduled for December 1999. The fuel will then be conditioned, repackaged, and stored in a dry storage area within the planned Canister Storage Building at the Hanford Site.

DOE/Navy/Idaho Agreement on Spent Fuel and Nuclear Waste (6)

Following several months of intensive negotiations, the Departments of Energy and Navy reached agreement with the State of Idaho regarding spent fuel and radioactive waste issues at the Idaho National Engineering Laboratory. The terms of the agreement were incorporated into a Consent Order bringing long-standing litigation to a close. Negotiations were driven by a set of guiding principles and interests on the part of the Federal government and the State. DOE's interests centered on continued national security, future environmental protection, effective investment of public funds, and equitable treatment of other concerned states by the DOE and the Department of Navy. As for the State of Idaho, key negotiation interests included obtaining firm commitments by the Federal government regarding dates for the removal of all spent nuclear fuel and other wastes from Idaho, acceleration of environmental restoration at INEL, and economic investment in southeastern Idaho. It should be noted that DOE and the Navy negotiated in good faith and entered into this

agreement with the State of Idaho for the explicit purpose of preserving U.S. national security interests.

Under the agreement, the Navy is to resume spent fuel shipments to Idaho immediately, up to a total of 575 Navy fuel shipments (55 MTHM) through the year 2035. Prior to December 31, 2000 , the receipt of DOE-owned spent nuclear fuel shipments at the Idaho National Engineering Laboratory shall be contingent upon the United State's adoption of a policy to manage spent nuclear fuel from foreign research reactors to support U.S. nuclear weapons nonproliferation objectives. The number of shipments would be limited to 61 foreign research reactor spent fuel shipments. Following the year 2000, no more than 55 MTHM of DOE-owned spent fuel, approximately 497 truck shipments, would be made to Idaho. Furthermore, the constraint on receipt of only essential DOE shipments at the Idaho National Engineering Laboratory shall continue only until such time as a repository or interim storage facility outside of Idaho is accepting spent nuclear fuel from INEL.

As part of the settlement, the State of Idaho has agreed to support fully DOE's efforts to obtain the permits and approvals necessary to meet its initiatives under the Agreement, and in any future third party lawsuits brought against the Government concerning the adequacy of the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement and Record of Decision.

In return for these shipments to INEL, DOE has made a number of commitments to the State of Idaho. Failure to meet these commitments would result in suspension of DOE shipments. Key DOE commitments include:

Begin removal of transuranic waste from INEL by April 30, 1999, with complete removal by 2018

Complete removal of all spent fuel from INEL by 2035

Complete treatment of all calcined high-level waste at INEL by 2035

A mixed waste treatment facility for transuranic and alpha-contaminated mixed low-level waste at INEL operational by 2003

Transfer of all spent fuel at INEL from wet storage to dry storage by 2023

Transfer equivalent numbers of spent fuel shipments between Savannah River Site and INEL annually

Establish INEL as DOE's lead laboratory for spent fuel research and development

Over the next five years, DOE estimates the cost of its initiatives under the Agreement to be about $200 million; Navy initiatives are estimated at $110 million. In addition, DOE will provide Idaho with $30 million to assist in the diversification of the southeastern Idaho economy. If the Government fails to remove all the spent nuclear fuel at INEL from Idaho by January 1, 2035, the Government is obligated to pay (pending Congressional appropriations) $60,000 a day (unescalated) to the State.

DOE/Public Service of Colorado Agreement (In Principle) to Manage Spent Fuel at Fort St. Vrain (7)

Under the Idaho Agreement, DOE agreed not to ship Fort St. Vrain spent nuclear fuel from Colorado to INEL unless a permanent repository or interim storage facility outside of Idaho is opened and accepting spent nuclear fuel from INEL and the shipments are needed for the purpose of treating the spent fuel to make it suitable for off-site storage or disposal. As a result, DOE and the Public Service of Colorado have executed an agreement in principle for DOE to take title of the spent

fuel from the Fort St. Vrain nuclear power plant. Under this agreement, DOE will pay the Public Service of Colorado $16 million for the Independent Spent Fuel Storage Facility, plus future operating and maintenance costs. Approximately 1,464 spent fuel elements (about the volume of a railroad boxcar) are in storage. Once the agreement becomes final, and the transfer of licensing and closure of other legal matters is complete, DOE will take title to the spent fuel and transfer it off-site when DOE interim storage or the repository becomes available.

In 1965, DOE signed a contract to take the fuel and ship it to INEL for storage. However, strong opposition from the State Legislature of Idaho and Indian tribes have continually precluded DOE from doing so. Fort St. Vrain shut down in 1989. DOE saves approximately $2.0 million annually at INEL by consolidating spent nuclear fuel storage into space originally reserved for the Fort St. Vrain spent nuclear fuel.

NATIONAL ENVIRONMENTAL POLICY ACT

DOCUMENTATION AND DECISIONS

The type and nature of activities in implementing the DOE spent nuclear fuel program strategy are affected by decisions made under related National Environmental Policy Act documents. As discussed earlier, the framework for the DOE-owned spent-nuclear fuel program strategy is the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (SNF&INEL PEIS) Record of Decision issued on June 1, 1995. The SNF&INEL PEIS Record of Decision stated that DOE aluminum-clad spent nuclear fuel would be managed at the Savannah River Site and other DOE spent nuclear fuel will be transported and managed at the Idaho National Engineering Laboratory. The exception to this management regime is the Hanford Site, where all production fuel would continue to be managed onsite. The shipment limitations contained in the Agreement with the State of Idaho will cause some perturbation in carrying out the Record of Decision; in particular, it is likely that non-production spent nuclear fuel at the Hanford Site, with the exception of sodium-bonded fuel, will remain at the Hanford Site for continued storage.

DOE arrived at a key policy decision with respect to the ultimate disposition of DOE-owned spent nuclear fuel in developing the SNF&INEL PEIS Record of Decision. Specifically, some or all of DOE-owned spent nuclear fuel that is not otherwise dispositioned (i.e., chemically separated, with the high level waste being converted into a vitrified waste form for repository disposal), is authorized for disposal in the first repository. This is subject to the total quantity of DOE spent nuclear fuel and high-level waste not exceeding ten percent of the first repository capacity limit of 70,000 metric tons of heavy metal, and meeting the DOE-owned spent fuel acceptance criteria.

The SNF&INEL PEIS Record of Decision provides an important framework from which to manage DOE-owned spent fuel. In addition, other ongoing DOE NEPA documentation is under development on related spent fuel management activities. These include:

The Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (8)

Reducing the threat of the proliferation of nuclear weapons is one of the foremost goals of the United States. Proper management of spent nuclear fuel from foreign research reactors supports this goal, since much of

this spent fuel contains highly-enriched uranium which can be directly used in simple nuclear weapons. The proposed action being considered in the Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (FRR SNF EIS) is for DOE and the Department of State to jointly adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent fuel containing uranium enriched in the United States would be covered by the proposed action.

It is unclear at this writing what the Department's preferred alternative will be. Up to 19.2 MTHM of foreign research reactor spent nuclear fuel could be accepted and managed in the United States for national security purposes or managed at one or more overseas facilities under certain conditions. If managed in the United States, the spent nuclear fuel would be stored at the Savannah River Site and/or at INEL. A portion of the foreign research reactor spent nuclear fuel could be chemically separated at the Savannah River Site and the remaining aluminum-clad spent nuclear fuel would be stored. The non-aluminum spent nuclear fuel would be accepted and placed in dry storage at INEL.

The Draft EIS was issued in April 1995. At this writing, the Final EIS is expected to be issued in February 1996 with a Record of Decision following at least 30 days thereafter. The Final EIS and Record of Decision will be based, in part, on the outcome of the Interim Management of Nuclear Materials Environmental Impact Statement discussed below.

Interim Management of Nuclear Materials at the Savannah River Site (9)

In early 1992, DOE announced an end to defense-related chemical separations activities of spent nuclear fuel to recover highly-enriched uranium. This decision was based on the end of the Cold War and a reduction in the demand for new materials for nuclear weapons. The cessation of processing operations resulted in a large inventory of nuclear materials being caught in various stages of the historic production cycle at SRS that require continued management. Some of the methods used to store these materials potentially pose risks to the environment or the safety and health of SRS workers or the public because, at the time DOE suspended the production cycle, many of these materials were either in a form or were stored in a manner that was intended for only temporary periods (i.e., one to two years). In some cases, the material's physical or chemical form poses a risk; in other cases, the material simply needs to be repackaged or moved to another location to ensure safe storage.

The Interim Management of Nuclear Materials Environmental Impact Statement (IMNM EIS) evaluates the environmental impacts that could occur from alternatives for the management of nuclear materials at SRS that would be initiated over the next ten years. Some of the nuclear material at the Savannah River Site may require near-term stabilization to ensure continued safe management. Materials that are candidates for stabilization are either in forms (e.g., liquid) that present inherent management risks or stored in facilities (e.g., reactor disassembly basins) that were not designed for long-term storage, or both. Stabilization would be achieved by chemical separation of high-risk production fuels, and other processing/conversion steps. The IMNM EIS also evaluated alternatives for the restart of facilities which could be used for chemical separation of the foreign research reactor spent nuclear fuel as well.

The IMNM EIS was issued in Draft in March 1995 and in Final in October 1995. The Final IMNM EIS identifies stabilization (i.e., processing/chemical separation) as the preferred alternative for the management of certain high-risk types of material. The IMNM Record of Decision, released in December 1995 (10), announces DOE's intention to begin operations to stabilize most of these high-risk materials, consolidate stable materials, and store all SRS materials onsite until the Department resolves issues regarding their future disposition. The Record of Decision also declares that upon completion of further analysis, DOE will issue a subsequent Record of Decision to announce the Department's final stabilization strategy for some of these at-risk materials (e.g., neptunium-237 and plutonium-239 solutions).

Further, the IMNM Record of Decision announces DOE's amended preferred alternative for the management of aluminum-clad, highly enriched uranium fuel from SRS reactors. Based upon further consideration and resolution of crucial management concerns such as cost, schedules, and technical uncertainties, DOE now considers the stabilization of these materials as the preferred management strategy. In particular, aluminum-clad (spent fuel) targets would be processed in preparation for vitrification in the Defense Waste Processing Facility. The Department will issue a subsequent Record of Decision on aluminum-clad materials in the near future.

LEGISLATIVE INITIATIVES

Fiscal Year 1996 Energy and Water Development Appropriations

The FY 1996 Energy and Water Development Appropriations (11) bill includes $400 million in total, including $85 million (to be held in reserve) towards the construction of an interim storage facility for spent fuel and $315 million for the Civilian Radioactive Waste program to, in part, continue studies at Yucca Mountain. The appropriations bill directs the Department to refocus the repository program to include interim storage development while completing the core scientific activities at Yucca Mountain. Activities include completing the excavation of the necessary portions of the exploratory tunnel and the scientific tests needed to assess the performance of the repository. Further, DOE is directed by Congress to defer preparation and filing of a license application for the repository with the Nuclear Regulatory Commission (NRC) until a later date. Instead, DOE's goal is to collect the scientific information needed to determine the suitability of the Yucca Mountain site and to complete a conceptual design for the repository and waste package for submission to NRC at an appropriate future date.

These directives significantly impact the schedule of repository program activities and, at the same time, add a new dimension to program planning, namely the prospect of a centralized interim storage facility. DOE will no longer work toward the preparation of a license application for the repository beginning in 1996 with expected submission to NRC in 2001 as previously scheduled. The potential availability of an interim storage facility could impact the DOE spent fuel program strategy in terms of the Department's current plans for new interim storage facilities. It is premature, however, for the DOE spent fuel program to consider the use of a centralized interim storage facility as a storage option for its own fuel when such a facility would be built primarily to store commercial spent fuel.

Legislative Initiatives to Amend the Nuclear Waste Policy Act

While the FY 1996 Energy and Water Development Appropriations bill directs $85 million towards the construction of an interim storage facility, the actual implementation of the directive is contingent upon the approval by Congress and the Administration of a separate measure directing DOE to build an interim storage facility. The most prominent legislation before Congress that seeks to reshape DOE's spent nuclear fuel management program are companion bills to amend the Nuclear Waste Policy Act of 1982. In February 1995, Representative Fred Upton of Michigan introduced H.R. 1020 (12), the Nuclear Waste Policy Act of 1995. In September 1995, Senator Larry Craig of Idaho introduced its counterpart, S. 1271 (13), before the Senate.

These measures would authorize DOE to construct and operate, in a phased approach, an interim storage facility for spent fuel at the Nevada Test Site. (It should be noted, however, that the Administration is opposed to the companion bills' language specifying this location for the interim storage facility until a 1998 viability assessment on whether or not Yucca Mountain is a suitable site for the geologic repository.) The Senate bill increases the maximum capacity of the facility from 40,000 metric tons authorized in the House legislation to 100,000 metric tons. Both bills would also require DOE to begin accepting commercial spent fuel and high-level radioactive waste no later than January 31, 1998. The bills provide for additional DOE-owned spent fuel and high-level waste to be stored once an annual acceptance rate from civilian power reactors has been achieved. This would include Naval spent fuel and DOE-owned spent fuel resulting from atomic energy defense activity and from foreign research reactors.

Legislative Initiatives to Amend the Waste Isolation Pilot Plant Land Withdrawal Act

Legislative initiatives to amend the Waste Isolation Pilot Plant (WIPP) Land Withdrawal Act would also impact the DOE spent fuel program since DOE spent fuel shipments to INEL are tied to transuranic waste shipments out of Idaho as specified in the Settlement Agreement with the State of Idaho. In particular, the Agreement would suspend any shipments of DOE spent fuel after April 30, 1999 to INEL unless shipments of transuranic waste from INEL to the Waste Isolation Pilot Plant (located in New Mexico) or another facility are proceeding at a specified rate.

There are two key legislative initiatives underway to amend the WIPP Land Withdrawal Act passed in 1992. These are H.R. 1663 (14) sponsored by Representative Joe Skeen, and S. 1402 (15) sponsored by Senator Larry Craig. These proposals seek to eliminate regulatory duplicity, and unnecessary budget expenditures and delays in arriving at a disposal decision. Both bills are very aggressive in attempting to reach a WIPP disposal decision.

Skeen's proposal provides that the intent of Congress is for DOE to reach a disposal decision no later than March 31, 1997. This is almost one full year ahead of the current WIPP Disposal Decision Plan schedule (the WIPP Disposal Decision Plan shows a decision by January 1998). However, Skeen proposes to reduce the Environmental Protection Agency's (EPA's) role from "regulator" to that of "advisor" by only commenting on, and not certifying by rule, the DOE application for compliance with the disposal regulations. DOE is insistent that the EPA role remain that of a regulator.

The Craig bill for the most part retains EPA's role as that of regulator by providing for EPA's certification of DOE's application for compliance

with disposal regulations. However, the Department is concerned with the limited review role that the EPA and the general public would have in the certification process. The Craig bill also calls for a firm target date for the opening of the WIPP facility by providing that the intent of Congress is that after the Administrator's review and certification, the Secretary will begin the disposal phase no later than June 30, 1997. DOE agrees with setting a date to begin disposal operations at WIPP as early as practicable within the constraints of the EPA review and certification schedule.

Legislation on External Regulation of DOE Nuclear Facilities

In January 1995, the Secretary of Energy formed the Advisory Committee on External Regulation of U.S. DOE Nuclear Safety in response to congressional interest in external regulation of DOE nuclear facilities. In its December 1995 (16) report, the Advisory Committee recommended ending DOE self-regulation and identified the pros and cons of using the Nuclear Regulatory Commission or an agency based on the Defense Nuclear Facilities Safety Board as the DOE's future regulator of nuclear facilities. Furthermore, the Committee's report outlined key principles for external regulation including enforcement approaches, public participation requirements, relationships with state regulators, and guidelines for reconciling external safety oversight with DOE's legitimate national security imperatives. The Committee's recommendations, if implemented, would substantially redefine the regulatory framework under which DOE must operate and would contribute to DOE and Congressional efforts in developing legislation for external regulation in the coming years.

SUMMARY

In 1995, the Department of Energy has made considerable strides in the Spent Nuclear Fuel Program with the establishment of a management strategy for the treatment, storage and ultimate disposition of the Department's anticipated 2,800 MTHM of DOE-owned spent nuclear fuel. This strategy is outlined in the Office of Spent Fuel Management's November, 1995 Interim Storage Plan and Program Plan. These documents include the establishment of clear time-lines for the assurance of safe existing conditions, achievement of safe existing conditions, and preparation for the ultimate disposal of spent fuel in a geologic repository.

Legal challenges to DOE activities, State and Federal government agreements concerning spent nuclear fuel, NEPA documentation and Records of Decision concerning DOE-owned spent fuel storage, processing, and ultimate disposition, and various legislative initiatives that affect the timing of repository availability, will all affect the direction of DOE's spent nuclear fuel program. This paper has attempted to introduce and examine several of these influences and provide insight concerning their impact upon the future of the Program.

Long-standing litigation with the State of Idaho concluded with a negotiated Agreement between DOE, Department of Navy and the State. These negotiations were undertaken due to paramount national security interests and were a necessary step in moving DOE's spent nuclear fuel program forward.

Current NEPA documentation for the program will lead to pending decisions affecting the State of South Carolina. Recently, the Interim Management of Nuclear Materials EIS Record of Decision announced DOE's intention to chemically separate certain high-risk materials at the Savannah River Site. One such decision pending in the near future is whether or not to

adopt a policy to manage foreign research reactor spent nuclear fuel in the United States. Such policy decisions may necessitate negotiations with South Carolina so that DOE can implement activities in a timely manner at the Savannah River Site. If necessary, negotiations would be predicated on overriding national security interests. As in the case of Idaho, such an agreement would be another example of redefining Federal-State relations to jointly implement policies at DOE management sites. It is expected that a new legislative framework will evolve that will help refine the path forward for the DOE spent nuclear fuel program, especially with regard to the timing/availability of a centralized interim storage site for civilian and DOE-owned spent nuclear fuel, a permanent geologic repository, and the Waste Isolation Pilot Plant. In addition, it is expected that Congress will act in developing legislation for external regulation of DOE nuclear facilities that will have implications to the DOE spent nuclear fuel program.

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10-2

THE EXPERIENCE OF INDUSTRIAL REPROCESSING AT LA HAGUE

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ABSTRACT
The full Reprocessing-Conditioning-Recycling (RCR) concept is an integral part of the French nuclear policy, initiated in the 1970s. COGEMA operates the World's largest reprocessing plant in La Hague, France with a 1600t/year reprocessing capacity. The plant serves the French utility and 27 electricity utilities from other countries. COGEMA has developed extremely sophisticated technologies over the past 30 years to treat the large quantities of fuel it reprocesses yearly and their diverse nature, both French and foreign.
Today, the La Hague plant has achieved a 99.88% recovery of the fissile materials contained in spent fuel and has developed safe waste conditioning through vitrification, a technology internationally recognized as the best for high level waste. In spite of its proven mastery of reprocessing on an industrial scale, COGEMA continues to apply the ALARA principle. Thus, new technologies for waste minimization are being developed, including, from now to the year 2000, the suppression of bitumen packaging, the compaction of hulls and end-pieces, and the reduction of technological waste originated in the reprocessing operations.
Decreasing the plant's environmental impact and the personnel's occupational exposure have also been major goals ever since the plant was conceived. The environmental impact is negligible. The surrounding air, land, and sea are systematically monitored. Both sample analyses and continuous measurement instruments demonstrate the extremely low level of radioactivity added to the natural level. Similarly, in applying the ALARA principle, radiation safety is a major concern, and has shown very effective. The average dose in 1995 was 0.20 mSv/man/year, and has decreased to less than 10% of the dose equivalent resulting from natural radioactivity.
The plant's design has made it possible to improve the effluent management, the waste segregation procedures, the vitrification units, crane equipment, etc... without disturbing production. Thanks to the permanent improvement of the facilities, the total volume of waste produced is now less than what it would be in the direct disposal of spent fuel option. In contributing to an optimal use of natural resources through recycling and reducing both the toxicity and the volume of its waste, COGEMA has demonstrated its commitment in favor of sustainable development.
INTRODUCTION
France opted in the 1970s to close its nuclear fuel cycle, thereby optimizing the industrial use of fissile materials and minimizing waste production. The enactment of this policy led to the construction of the World's largest reprocessing facility at La Hague in 1980s. With the start-up of UP3 in 1990 and of UP2-800 in 1994, the Reprocessing-Conditioning-Recycling concept has been completed, both as a part of the French electricity program and as an industrial service facility for foreign customers.
As the La Hague plant's operator, COGEMA has demonstrated its capacity to master large-scale reprocessing technologies. It has also led the way in the improvement of quality, safety, and waste minimization. The operational experience of the plant shows that production, maintenance and a low environmental impact are fully compatible, thanks to a carefully planned design and strict operational procedures.
NEARLY 30 YEARS OF EXPERIENCE

The Hague's total reprocessing capacity of 1600 metric tons per year has been achieved through constant improvements over three decades. The industrial mastery of reprocessing and recycling is evidenced by the progress made in attaining the full capacity : it took ten years to reach the nominal throughput for UP2-400, only five to reach double that throughput at UP3, and hardly over one year for UP2-800, commissioned in 1994, to reach its nominal capacity.

Quantitative and Qualitative Experience

To date, over 17,000 metric tons of spent fuel have been transported from European and Japanese reactors to La Hague, with over 4200 shipments by rail or sea. Every year, the French utility (EDF) sends approximately 1000 tons to the site. Figure 1 shows the combined UP2 and UP3 facilities have reprocessed 8262 tons of spent fuel as of January 1, 1996.

Fig. 1

La Hague also bases its experience on a wide variety of fuel inputs. Over the years, these fuels have included fuels from Light Water reactors (both PWR and BWR) as well as the tail end of the GCR reprocessing. Feasibility of reprocessing MOX fuel was demonstrated by a successful campaign in 1992. Lastly, fuel from the first French fast breeder reactor was reprocessed at UP2 as well. The diversity lies not only in the nature of the fuels but in the burn-up as well, spanning from 10 000 to 45000 Mwd/t. Thus, COGEMA's operational experience is both broad and deep. Through this experience, a 99.88% recovery rate has been achieved for recyclable materials (uranium and plutonium).The remainder (0.12%) is vitrified along with the fission products, which represent 99% of the radioactivity contained in the spent fuel, but only 3% of its mass.

Advanced Technology

The vitrification technology was developed in France. It was implemented in the AVM facility at Marcoule, and later on, in the R7 and T7 facilities at La Hague. It is based on a two-step process, which separates the calcination and glass melting operations. The final product, vitrified residues, are today the only HLW form licensed by the French and foreign regulatory bodies. Its confinement qualities are based on a 15 year qualification program performed at CEA (the French public research institution for atomic energy), and on the active quality control and quality assurance programs set forth by the operators of the vitrification facilities.

WASTE MINIMIZATION

Above and beyond the high degree of recovery, COGEMA is concentrating its efforts on three waste minimization programs : the suppression of bitumen packaging, the compaction of hulls and end-pieces, and the reduction of "technological waste".

Suppression of Bituminized Packages

Until now, medium and low-level effluents were routed to the waste treatment facility "STE3" and treated by a "coprecipitation process" involving coprecipitation by different additives, decantation and filtration steps, which provides for an efficient decontamination factor. The active sludges were conditioned in bitumen and considered as Intermediate Level Waste. These wastes exhibit an alpha activity above the threshold of 3.7 GBq/t (0.1 Ci/t) and are not acceptable by a surface disposal site in France. The quality of the bitumen waste form, necessary for underground storage, has been demonstrated after a comprehensive characterization program performed by the CEA and acknowledged by the French Safety Authorities and most foreign governmental authorities.

However, the bitumen matrix is no longer necessary. Feedback from the operation of the UP3 plant shows that the volume and the activity of low and intermediate level effluents was, from the beginning, significantly lower than estimated at the design stage. This comes from the very high performances of the extraction cycles, resulting in lower than expected activity levels of some of the effluents and allowing for significant improvements in liquid waste management.

The improvements are based on both a more sophisticated segregation of the wastes according to their chemical and activity content, and the implementation of additional evaporation capacities in the plant. Effluent segregation allows for the discharge of the very low active streams to the sea whenever possible, after filtration and monitoring, without increasing the released activity.

Effluents from the analytical laboratory also need to be considered. These effluents traditionally include a mix of sample excess and chemical reagents which are not present in the main process. Moreover, the analytical effluents contain different levels of alpha and beta-gamma activity. The improvement of their management first relies upon a segregation within the analytical lines. According to their nature, they will be either recycled into the plutonium purification cycles (alpha bearing effluents) or routed to the vitrification unit (beta-gamma effluents) whenever possible. For effluents containing unwanted chemicals, a special coprecipitation unit will be used to separate the alpha activity. The amount of unwanted ions such as phosphates, sulphates, and chlorides is being significantly reduced through the use of alternative analytical methods.

As of mid-1995, these improvements make it possible to route practically all the concentrated activity to the existing vitrification facilities. Thus, the need for coprecipitation and bituminization treatments has disappeared in normal operating conditions for the intermediate and low level effluents from the UP3 and UP2-800 plants. The resulting small increment of activity conditioned in the glass will not increase the volume of glass produced.

Compaction of Hulls and End-pieces

Hulls and end-pieces separated from the irradiated fuel in the head-end of the reprocessing plant are currently conditioned by cementation, a process which was licensed by the French regulators, while COGEMA continued investigating alternative possibilities. COGEMA is currently developing a compaction technology which should be operational before the end of the decade. Compacted pellets will then be loaded in a glass-type canister, and the corresponding volumes should decrease to approximately 0.15 m3/t of Rep U.

Reduction of "Technological Waste"

The last category of waste conditioned at the reprocessing plant originates not from the fuel itself, but from the reprocessing operations, and is called "technological waste". All the technological wastes are now conditioned in a single type of fiber-concrete container. A drastic minimization has already been obtained by comparison to the design waste volumes, by relying on the high equipment reliability and improved sorting, performed at the entrance of the solid waste processing facility (AD2). The sorting makes it possible to send more than 80% of the technological waste to surface disposal. Further minimization could involve a compaction, incineration and a plutonium decontamination

process based on oxidizing dissolution, as well as the implementation of new repairing techniques.
The volumes of each waste category are summed-up in Fig. 2, where are compared the design values and those derived from the actual plant operation. The figure shows that the amounts of waste produced are quite a bit lower than expected. From the beginning of the oxyde fuel reprocessing at UP3, great steps have been taken toward minimizing the different types of wastes resulting from the process. At this stage, the overall volume of long-lived waste (under 1.0 m3/t) is already lower than the volume of waste which would result from a direct disposal of the irradiated fuel in a once-through option (2m3/t). This achievement is due to innovative technologies and effective management of waste conditioning.
Fig. 2
Expected Results
After the completion of all the necessary developments, the overall volume of high level and long-lived waste will be under 0.5 m3/t of Rep U. Most of the waste will be delivered in standard, glass-type canisters. Low-level technological waste will be conditioned in standardized fiber-concrete containers, in order to facilitate handling and optimize storage.
LOW ENVIRONMENTAL IMPACT AND OCCUPATIONAL EXPOSURE
Both the plant's design and its operation are aimed at making the environmental impact negligible. Authorized limits of alpha and beta activities have been determined for the site's liquid releases, based on an environmental impact assessment. As seen in Fig. 3 the actual discharges have been continuously decreasing and are well below the authorized levels, despite the increasing quantities of fuel reprocessed yearly.
Fig. 3
A direct measurement of the site's environmental impact is provided by the Health Physics Department. Its systematic monitoring of the surrounding air, land and sea on the basis of both sample analyses and continuous measurement instruments shows the exceptionally low level of added radioactivity. A particular attention is paid to the presence of radioelements in dairy products and seafood. The results show that there is a continuous decrease of activity in these products down to less than 10% of the natural radioactivity level.
Reducing the occupational doses to COGEMA and subcontractor staff is a unending quest when one applies the ALARA principle. The level achieved shows that radiation safety procedures have been truly effective. In 1995, the average value of 0.20 mSv/year/man was reached, which is less than 1% of the value authorized by current regulations. Figure 4 shows the decreasing trend in occupational doses over the last 10 years.
Fig. 4
EFFICIENT IMPLEMENTATION OF MODIFICATIONS
It should be noted that it has been possible to introduce the improvements described above without interrupting the production process because the initial design accounted for further modifications.
The increased sophistication of the waste segregation, and the addition of several evaporation units have required substantial modifications in the existing facilities. The lay-out concept of UP3, where active units are located in specific cells, has made it possible to perform these modifications without significant interferences in the process, while

minimizing the dose rate to the workers. The connections to the operating active units have been facilitated by the waiting pipes provided outside active cells.
Other modifications have been achieved in vitrification units, with positive consequences for waste minimization. From the start-up of R7 and T7, uninterrupted progress has been achieved without disturbing production, in particular :

- implementation of a new melting equipment to increase the melter lifetime up to 3000 hours ;
- addition of a gas containment system to decrease the pouring cell contamination;
- modifications of a crane equipment and its maintenance procedures to improve reliability.

CONCLUSION

The Environmental Summit held at Rio in 1992 ended with the commitment of industrial countries in favor of sustainable development. For years, France's reliance on nuclear energy, which led to drastic reductions of toxic emissions even before the Rio decisions, has been a confirmation of such a commitment. The early choice of a closed fuel cycle is the logical continuation of its nuclear policy, and is fully consistent with the principles contained in the Rio decisions. Spent fuel reprocessing significantly contributes to:

- environmental protection : waste is safely conditioned in a form appropriate to its nature;
- conservation of natural resources : recycling of energetic materials allows the recovery of nearly 20,000 TOE per metric ton of spent fuel.

The fact that the reprocessing and recycling facilities are operated in a way which minimizes the amount of liquid and solid waste produced is also consistent with sustainable development. This has been COGEMA's unvarying strategy, exemplified by the commissioning of modern plants at La Hague, UP3 in 1990 and UP2-800 in 1994. COGEMA is completing a first important step with the suppression of bituminized waste in modern facilities, and new developments are being launched, which will bring a further minimization of high level and long lived-waste.
The total volume of high level and transuranic waste produced in UP3 is already less than what would result from direct disposal of irradiated fuel. From now on, all but an infinitesimal amount of the activity will be concentrated in glass. All the necessary modifications are performed without disturbing plant production, because their possibility is accounted for in the initial design. By this continuous improvement strategy with regard to quantity as well as quality of wastes produced, COGEMA has demonstrated its commitment in favor of sustainable development.

10-3

STATUS OF THE NEW FRENCH MOX FABRICATION PLANT: MELOX

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ABSTRACT

The MELOX plant, which is currently entering industrial operation, is an important element of the RCR strategy (Reprocessing, Conditioning and Recycling) which is steadily implemented in France. Being the first high throughput MOX fuel manufacturing facility, MELOX plant can be seen as an outstanding achievement, based on well-tried techniques, but introducing several innovative features in terms of radiation protection as well as environmental concern.

While the first decision was taken in 1985, construction started in 1990 and active operation approval was granted by Regulatory Authorities in August 1994.

Industrial start-up is currently underway, and production is progressively increasing, implying customer's involvement in a detailed and complete qualification of the whole process.

Acquired results give confidence that the objectives of full-scale plutonium recycling will be met, serving French and foreign utilities' optimized back-end strategies. As a result, further capacity and flexibility extensions are envisioned.

THE MELOX CONTEXT

From the very origin of the French nuclear program, reprocessing of spent fuel and recycling of valuable materials such as uranium and plutonium has been the thorough and coherent French strategy. Today, the possibility for the use of plutonium into MOX fuel appears economically sound, and that is why EDF decided in 1985 to recycle plutonium in some of its PWR 900 MW units. This strategy implements a set of well-mastered industrial operations, and at three places, the same flow of recycled materials is to be achieved : at the output of the reprocessing plants, at the input of the fuel manufacturing facilities and at the input of the recycling-dedicated reactors.In order to cope with EDF planned requirements (recycling in 20 to 28 PWR reactors), and to serve foreign utilities as well, a consistent industrial organization was established through industrial and commercial agreements among EDF, FRAMATOME and COGEMA on the one hand, and between COGEMA and BELGONUCLEAIRE (BN), on the other hand.

It appeared that existing facilities (BN P0 plant at Dessel and COGEMA Cadarache plant) would only partly match the projected needs and that additional capacity would be necessary. As soon as 1985, preliminary studies were undertaken for the construction of a large-scale MOX plant, MELOX, the design capacity of which is 120 tons of Heavy Metal per year (tHM/y).

The MELOX plant is located on the site of Marcoule, on a separate 50000 m2 area, and includes all manufacturing operations from UO2 and PuO2 powders reception to assembly delivery. It comprises two main buildings : a manufacturing building with production lines and laboratories and a scrap and waste treatment building including an incinerator for burnable waste.

The MELOX Company, created by COGEMA and FRAMATOME in order to supervise the construction of the plant and operate it, is now proceeding with industrial start-up activities aimed at a full qualification of the entire plant resulting in a progressively increasing production during the next months. After describing some of its specific design features, this paper gives information about some milestones and current achievements (as of mid-January 1996) of the MELOX plant industrial operation.

MELOX SPECIFICITIES

MELOX is the first high-throughput MOX fuel manufacturing facility to be put into operation in the world. While taking full benefit from the excellent performance of previous generation plants, MELOX is designed to fulfill the foreseeable requirement trends in both fuel management and modern safety and environment concerns.

An Innovative Plant Based on Well-tried Techniques

Relying on the outstanding experience of the pre-existing facilities (MOX fuel at Dessel and Cadarache and UO2 fuel at Dessel, Romans and Pierrelatte), MELOX is implementing, whenever possible, proven and qualified industrial-scale processes and techniques.

The reference process is the well-mastered MIMAS process (Micronized MASter blend) which has been successfully implemented in the Dessel plant since 1984. The PuO2 powder is micronized with a part of the UO2 powder to form a primary blend of 30% plutonium content, which is then mechanically diluted and mixed with a free-flowing UO2 powder to obtain the specified content of the MOX fuel. Sintering is then performed under temperature and gas blanket conditions adjusted to the densification and diffusion requirements for the pellets.

Research and development activities have been performed in order to confirm the choice of some equipment, devoted to homogenizing, multi-punch pressing, pellets grinding, rod welding and rod repairing. Moreover, some innovative equipment have been introduced, such as automated sorting of pellets for surface defects and rod decontamination before assembling.

A Modern Unit Coping with Future Trends

Although dedicated to a number of FRAMATOME assembly designs, MELOX offers to the utilities a flexible use of MOX fuel. As a result, it may receive and handle a wide range of basic nuclear materials. This implies the capability of using high-burnup and aged plutonium, which has consequences, through Pu238 and americium content, both on the specific thermal energy to be evacuated and on radiation shielding to be installed. On the other hand, the trend towards an increased MOX fuel discharge burnup leads to high plutonium content in the fabricated fuel.

MELOX characteristics are given below :

- Pu 240/Pu total 17 %
- Pu total/U+Pu+Am 12.5 %
- Am 241/Pu+Am 30 000 ppm
- average thermal power 17.6 W/kg PuO2

Safety and Environmental Concerns

A special design effort, while coping with the above requirements, has been devoted to set the annual dose of most factory operators to a value below 5 mSv/yr, which is one tenth of the current regulatory limit. This implies almost total automation of production and required extensive optimization studies tailored to each workstation.

Another design effort has been devoted to waste minimization. Besides the recycling of most of the scraps in the process line, the objective is to achieve chemical treatment of unsuitable scraps, to decontaminate metallic waste before conditioning and to incinerate organic waste with chemical treatment of the ashes. A significant decrease of residual plutonium in the waste should be achieved (the expected order of magnitude is about 500 g/yr in the final waste).

On the other hand, low activity liquid waste will be treated in the COGEMA Marcoule treatment unit. Radioactive discharge from MELOX will

represent one thousandth of the overall limit authorized for the Marcoule site.

The MELOX plant can be seen as an outstanding achievement, taking full benefit from previous experience, while introducing several innovative features in terms of radiation protection as well as environmental concern. It is a major input in the national energy policy, as an important element of the plutonium recycling strategy.

SOME MILESTONES

Governmental Authorities licensed the MELOX plant as a Basic Nuclear Facility in May 1990. Construction started in 1990 and civil works of the production building were completed in 1991. After main equipment installation, inactive testing program started in 1993.

Several stages have been progressively performed and, as they proceeded, increasing confidence has been gained concerning satisfactory operation of the whole production line as well as of all its parts.

A safety advisory commission meeting held in March 1994 led to active operation approval by Regulatory Authorities in August 1994.

A step-by-step strategy has been implemented for the commissioning procedure. The objective is to provide the customers (i.e. the fuel vendor, FRAGEMA, a FRAMATOME/COGEMA joint venture, and the final user, EDF) with a gradual, but complete, demonstration that the fabricated products are of required quality. The result is the "process qualification", granted by the customers, which ensures that the product quality will reliably meet his requirements.

As a general rule, back stream units are to be qualified when qualifying one unit. Thus, the assembly unit was commissioned first, using rods coming from the Cadarache plant : as soon as December 1994, authorization was granted by FRAGEMA and EDF to proceed with industrial assemblies production. Then, powder-pellet and rod units can be qualified using the normal process pathway.

For the powder-pellet unit, parametric tests have been performed using exclusively UO2 powders, in order to minimize scrap production during start-up. After the special authorization received in February 1995, the first welded cartridge box containing plutonium was opened on February 7th, allowing operation of the powder-pellet workstations.

A verification phase for the parametral tests was necessary and was performed using little quantities of PuO2 in the laboratory test-lines, sintering being performed in the normal line furnace. The acquired results, involving high and low plutonium contents, were satisfactory and allowed UO2 tests validation.

Full industrial operation of the MOX fabrication line was evidenced, as the first MELOX rods were delivered in July, 1995.

MELOX start-up is being performed according to a very carefully prepared logical course and implemented using a step-by-step program, implying customer's involvement in a detailed and complete qualification of the whole process. Full capacity will be reached progressively by increasing the capacity of each individual piece of equipment. Acquired experience gives confidence that this will be achieved without major difficulty, thus demonstrating the validity of the original options.

PRESENT STATUS AND ACQUIRED RESULTS

Assembly Unit

This unit is fully qualified and is functioning on-line with the actual production of 76 assemblies in 1995, 40 of which have been delivered to the nuclear power plants and loaded in the reactor core. It should be

noted that, at the same time, dispatching and transport activities have been fully qualified as well.

Rod Unit

The first MELOX rods were produced in July. More than 800 rods have been manufactured in 1995. They are a part of the first campaign, launched in order to produce a whole reload (16 assemblies with three plutonium contents).

Powder and Pellet Unit

An industrial demonstration stage is currently in progress, using representative quantities of plutonium (hundreds of kg). This stage involves the whole line. At the end of the year 1995, 9 tHM of Mixed Oxide have been processed, of which about 5 tHM have undergone pelletizing.

Two lines of pelletizing and two furnaces are being, by now, operated with PuO_2.

Waste Conditioning and Incineration Building

This building was officially transferred from the engineering team to the operator in April, 1995. Inactive tests for the incinerator are still in progress and the results are to be submitted to the Safety advisory commission. On the other hand, the waste conditioning and storage part of the building has been granted active operation authorization and has started dealing with wastes coming from the fabrication building.

It should be noted that, as far as no fabrication is involved, customer's qualification is not required in this building.

CONCLUSION

Through a consistent and large-scale program, recycling of plutonium in light water reactors is a mastered and industrial reality. It implements a set of effective operations, the impact on the environment and human health being set well below required limits.

COGEMA effectively possesses a fully coherent facility system for nuclear fuel back-end optimization with MELOX, the Cadarache plant and the reprocessing facilities in La Hague. The maturity of the RCR (Reprocessing, Conditioning and Recycling) strategy in France is being reached.

While the full-scale operation of this strategy will be reached in France before year 2000, COGEMA is able to serve foreign, European and Asian utilities as well. As far as the MOX manufacturing step is concerned, COGEMA considers the possibility of increasing its capacities in order to serve customers who want their MOX fuel to be fabricated in France.

When providing a three-shift operation on some units, MELOX capacity can easily reach a 160 tHM/yr production for standard fuels, which is planned by year 2000. Besides, COGEMA is studying an extension project : adding a fourth sintering furnace and a related independent line could further increase the capacity by 50 tHM/yr with enhanced flexibility concerning BWR MOX fuel fabrication.

COGEMA will adapt to the actual demand and has the references, the experience and the technology to demonstrate the feasibility of a fast, safe and reliable program of MOX recycling in LWRs. While successfully applied to the civilian nuclear industry, such a satisfactory experience could be brought to solve the current concern about weapons plutonium inventory disposition : "moxification of this military plutonium would product electricity while burning it and deteriorating its weapons value.

"THE INEL APPROACH"
ENVIRONMENTAL RESTORATION PROGRAM MANAGEMENT AND IMPLEMENTATION METHODOLOGY

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ABSTRACT

The overall objectives of the INEL Environmental Restoration (ER) Program management approach are to facilitate meeting mission needs through the successful implementation of a sound, and effective project management philosophy. This paper outlines the steps taken to develop the ER program, and explains further the implementing tools and processes used to achieve what can be viewed as fundamental to a successful program. The various examples provided will demonstrate how the strategies for implementing these operating philosophies are actually present and at work throughout the program, in spite of budget drills and organizational changes within DOE and the implementing contractor.

A few of the challenges and successes of the INEL Environmental Restoration Program have included: a) completion of all enforceable milestones to date, b) acceleration of enforceable milestones, c) managing funds to reduce uncosted obligations at year end by utilizing greater than 99% of FY-95 budget, d) an exemplary safety record, e) developing a strategy for partial Delisting of the INEL by the year 2000, f) actively dealing with Natural Resource Damages Assessment issues, g) the achievement of significant project cost reductions, h) and implementation of a partnering charter and application of front end quality principles.

BACKGROUND

The Idaho National Engineering Laboratory (INEL) was established in 1949 by the U. S. Atomic Energy Commission to build, operate, and test various nuclear reactors and fuel processing plants, and to provide and operate various related support facilities. Since that time, 52 reactors have been constructed, 13 of which are still operable. Today, the INEL also supports other government-sponsored projects including energy, defense, environmental, and ecological research.

The INEL is located 42 miles west of Idaho Falls, Idaho and occupies 890 square milers of the northwestern portion of the Eastern Snake River Plain. The INEL is bound on the northwest by three mountain ranges: Lost River, Lemhi, and Bitterroot. The remainder of the INEL is bound by the Eastern Snake River Plain.

The Environmental Restoration (ER) Program was established in 1989, when the INEL was placed on the National Priorities List. Ten waste area groups (WAGs) were created based on geographical boundaries for management of remediation activities. Contamination the ER Program deals with includes radioactive and mixed waste in soils, groundwater and that buried at disposal facilities.

During the first operational years of the ER Program, it became apparent that careful project management would be essential for the milestones negotiated in the Federal Facilities Agreement and Consent Order (FFA/CO) to be met. The approach described here was not conceived on the first day of operations of the program, but rather is the result of several years of evolution that included the pains of learning, as well as the

excitement of success. This approach has helped the INEL ER Program in meeting or beating the enforceable milestones in the FFA/CO.

Environmental restoration programs at other sites can benefit from the INEL ER Program's years of continual learning and improving, by reviewing the approach used at the INEL and incorporating those activities that improve their program. Modeling new processes based on the experience at the INEL will result in reduced schedule durations and related costs for environmental restoration.

AN INTRODUCTION TO THE APPROACH

The Need for Improved Program Management

The overall objective of the INEL ER Program management approach is to facilitate meeting mission needs. This is accomplished through the successful implementation of a sound and effective project management philosophy.

The INEL ER Program management approach assures the application of sound management principles to provide a disciplined, systematic, and coordinated approach to project management. This results in efficient planning, organization, coordination, budgeting, management, review, and control to achieve INEL ER objectives.

Employing a "value added" approach to guidance implementation ensures that DOE Environmental Management (EM) goals, strategies, and budgets are in alignment. Voids in EM guidance are jointly interpreted by DOE-ID led contractor task teams, and modified as necessary to suit INEL ER Program needs and requirements.

The INEL approach can be best described as systemic. Utilizing the focused principles of teamwork, communication, and consensus building provides the program with unity; thereby, enabling it to function as a fully coordinated, cohesive, and mutually supportive organization.

The INEL ER Baseline represents the successful integration of the program's cost, schedule, and technical work scope. It is from this level that all program elements are established and maintained. From the baseline standpoint, the process of maintaining a balance between the functioning elements of the ER Program (cost, schedule, and technical), and EM goals and priorities are established.

What follows provides some background on how we have developed our program, and focuses on the fundamental elements of our management methodology; describing the operating philosophies and the strategies for implementation.

Share Sound Management Philosophies

There is a strong desire by many organizations to learn how to do things more efficiently, to learn techniques that have substance, to learn useable processes that provide tangible results. They want to learn methods that will help them solve chronic, underlying problems and that focus on principles that bring long-term results. Something beyond typical management basics is needed.

INEL ER experience has proven the benefits of increased understanding through the sharing of information. Through this introduction to the INEL approach, there will be a sharing of ER program background and experience, and opportunities to demonstrate how to use the operating philosophies and strategies associated with their application. How these implementing strategies and operating philosophies can be adapted to suit many circumstances will also be demonstrated.

Through the collective experience of its participants, the INEL ER program has come to possess the tools, and lessons learned, and so on,

that has enabled the development of quick, economical, and environmentally sound processes that work, and that can be shared with others. There is also recognition that good communication and feedback are a key element in the continuous improvement process. The interaction and discussion that takes place in sharing information with others is an invaluable means of renewing and validating our operating philosophies. Through sharing of background and experience, the learning curve can be lessened for others, providing something that will save time and money, and ultimately increase functional effectiveness.

Stepping back, we have identified what elements have been present in each of the implementation steps taken in development of the ER program. The result of this retrospective examination is a list of operating philosophies which when properly applied, along with the implementing tools and processes described in this paper, have allowed ER to build a successful program.

INEL Operating Philosophies

Goal oriented approach (this is the key element and is also the first step)

- Clearly defined objectives
- Flexible assumptions
- Finite scope
- Value added concepts
- Cost effective, efficient

Teamwork (partnering)

- Communication
- Consensus building
- Contractor integration
- Unity of cause and purpose

Process oriented

- Systematic
- Coordinated approach
- Work to fill voids in guidance requirements
- Develop processes which are complimentary to existing (successful) processes

Incremental development

- Recognize this is an evolutionary process
- Improve process continuously
- Incorporate flexibility to suit program dynamics

Disciplined approach (commitment)

- Compliance with established processes
- Consistency
- Integrity/quality
- Credibility

Why These Philosophies Work

A sound management methodology, and a strategy to implement that methodology, is essential to ensuring that program management is reliable, well organized, consistent, and successful. A successful program embodies the successful integration of these operating philosophies along with the program's cost, schedule, and technical work scope. Using this approach, all program elements can be established and maintained from a central point of control. From that standpoint, the process of maintaining a balance between the functional elements (cost, schedule, and technical) and program goals and priorities can be easily achieved.

These operating philosophies stimulate a willingness and the capability to deal with exceptions encountered in building and using the various systems and processes, accepting that there are known incongruencies, but not allowing those incongruencies to distract from maintaining discipline with other systems/processes.

The following demonstrates how ER deals with the many influences on our program through the use of the operating philosophies, along with examples of the processes and tools developed to accomplish this. This in turn will help define the goals which make up "What is ER Program." Also included is a description of how the implementing strategies and operating philosophies can be established, or rather instilled within the various existing processes. The approach is a flexible application, and can be applied to a finite program or "fuzzy" program.

INEL OPERATING PHILOSOPHIES

The First Approach: Goal Oriented

The primary step is to be goal oriented. You must have established goals and a strategy to know where you want to go. You will need to develop a finite scope, define knowns, deal with unknowns as changes, and utilize assumptions. All actions taken need to be "value added" actions. Considering the many requirements placed on us from outside, it is important to use what is useful, modify it when necessary, but don't produce just to produce or to fill a square.

When the Federal Facilities Agreement was negotiated, the project managers from each of the stakeholders wanted the quickest and smartest cleanup possible. The negotiating team made of personnel from the U.S. Department of Energy, U.S. Environmental Protection Agency, and the State Environmental Agencies had a common goal. This goal was to ensure that releases or threatened releases were thoroughly investigated in accordance with the National Contingency Plan and that appropriate response actions were undertaken and completed as necessary to protect human health and the environment. Because the agreement is representative of the common goals of the team, the CERCLA process works quickly with regulators and performers communicating and moving forward.

Flexibility

This approach encourages timely remedy selection, flexibility for remedial action, and contingencies to respond to new information discovered during investigations. Through flexibility, implementing strategies and operating philosophies can be adapted to suit many circumstances. Flexibility allows all of the other approaches to work. Flexibility means to be willing to deal with exceptions, and building systems to accept exceptions as if they are inevitable. However, this should not distract from maintaining discipline with systems or processes; if the exception becomes the rule, it may be time to change the system or process.

Defining the technical scope and schedule for remedial action involves the approval of various outside agencies. ER Program management has the administrative flexibility to deal with external regulatory authorities in a timely and responsive manner. Flexibility encourages timely, outcome oriented remedy selection for remedial actions, and an ability to respond to new information discovered during investigations.

Teamwork

The process of identifying who is on the team is critical to your success. Teams should include regulators, government representatives, and

performing organizations as the principle team participants in the development of the scope of work.

It has been our experience that facilitated retreats have assisted in bridging the adversarial attitudes that existed at the beginning of the program. Communication is much easier if the participants can place a face with the name of a regulator.

Facilitated retreats have been used as a tool to develop better communication skills and to exchange information among participants in the decision process. A given benefit is that the parties to establish communication face-to-face in an atmosphere of neutrality; they are no longer talking to a stranger over the phone. This communication also allows members to establish a collective understanding of the program and its long range goals. Participants in the retreats have also been able to jointly define success in the completion of the project.

The generation of the formal ER Partnership Charter is the result of an off-site retreat conducted for all ER Program Waste Area Group Managers and Project Managers. The retreat was successful in developing a greater understanding of the shared values and responsibilities that must exist between DOE-ID and Lockheed Idaho Technologies Company, in order to fulfill the Program's goals.

The elements of the Charter are founded in the development and use of sound communication skills, and the use of same to facilitate completion or achievement of the ER Program goals. We are continually investigating effective ways to further develop and enhance the communication and facilitation skills of all ER Program participants, in order to maximize the progress made in implementing our Charter to date.

Process Oriented

The ER Program has developed strong hierarchy of processes or strategies which allow us to build the ER Baseline and to maintain it. This hierarchy of processes is auditable and consistent (back to the baseline). Each respective process is interrelated and complimentary of the others. Maintaining a "value added" management approach, each process has a specific fit and function. The result is an increased capability to respond to the natural dynamics of the program.

The Federal Facilities Agreement establishes a procedural framework and schedule for developing, prioritizing, implementing, and monitoring appropriate response actions in accordance with CERCLA, RCRA, and Idaho Hazardous Water Waste Management Act.

Preliminary Scoping Guidance documents were developed to facilitate implementation of the specialized sections of the facility agreement. All parties (contractors, DOE, and Regulators) agree to use the appropriate approach and assumptions to evaluate sites and minimize costly sampling and analysis plans and reduce RI/FS requirements. The success of these rapid decision processes has allowed completion of all enforceable milestones on orahead of schedule and on or under budget. ER has also been able to close over half of the sites in just under three years.

Successful Program Management and Implementation Processes

The INEL has focused on the development of a single integrated baseline which can respond to all DOE program and project management control requirements. The INEL ER Program Baseline represents the successful integration of the programs cost, schedule, and technical work scope. Numerous processes or mechanisms have been developed in order to organize the various elements of the ER program into a whole. All program elements are established and maintained at this level. These "baseline tools"

regulate the scope, schedule, and budget for the ER Program, and provide the means for the program to more efficiently address issue and concerns, and to support the requirements and interests of DOE-HQ, regulatory agencies, and various other ER stakeholders.

Hierarchy of Baseline Development Processes

The ER Program has developed a strong hierarchy of processes or baseline tools which allow us to build the ER Baseline and to maintain it. This hierarchy of processes is auditable and consistent (back to the baseline). Each respective process is interrelated, and complimentary of the others. A "value added" management approach is maintained, and each process has a specific fit and function. An increased capability to respond to the natural dynamics of the program is the result.

Life Cycle Planning (LCP) ProcessReplaces annual rebaselining efforts by allowing updating and refining of the baseline only when changes to the program requirements, assumptions, or resource availability are required.

INEL ER Code of AccountsLinks a uniform cost estimating process with accounting system(s) to facilitate cost capture (prior to COA being used, Cost Estimate Basis Sheets are used to document cost basis).

INEL ER Unit Price BookMaximizes the use of historical data and site experience through a unit price guide. The cost estimating focus group concentrates efforts to establish credibility, consistency, duration, and so on, relative to cost estimating system management for the INEL ER Program.

Project Risk/Contingency AnalysisDevelops contingency for the cost baseline to account for uncertainties, risk, and potential growth in scope that may result from unforeseen and unpredictable conditions. This process, which may be iterative, is performed by the Project Manager/CAM level down to the work package/task level, and ultimately establishes the cost baseline for the Control Account.

- Task levels are based on "level of confidence" in scope of worknot dollars
- Applied and controlled at the program level

Configuration Control/Baseline Management

- Uniform application of change control philosophy
- Uniform hierarchy for change control
- Change control scope oriented baseline elementone baseline
- Increase in programmatic understanding re:definition of Work Scope, integrated baseline reconciliation (periodic/all participants)
- Expedite submittal process through increased use of "redlines"
- DOE-ID ownership of all changes and DOE-ID concurrence required to "sponsor" Level 2 changes
- Uniform baseline reports
- Tie traceability back to funding
- Improved BCP process
- Increase quality of Baseline documentation.

Uniform Milestone Guidance/Scheduling ProcessUtilizes pre-negotiated milestone criteria that was determined through discussions with DOE-HQ, EPA, State, DOE-ID, Stakeholders, etc. Specific milestone coding, structure, and nomenclature with standardized milestone configuration management and controls. Utilizes standard logic.

Incremental Development

The first step for ER was to recognize that implementation of the program required an evolutionary process, incorporating flexibility, teamwork,

and continuous self-evaluation to suit program dynamics. This incremental development applies to both technical development and program management. Problems and issues are viewed from a global standpoint, relative to their effect on the program as a whole, and resolved as they appear on the horizonat the lowest level manageable. As the program has evolved, we have applied this approach by creating technical work teams for global issues affecting our program (for example, guidance document development, ATSDR, Cumulative Risk Assessment, BSAF, Eco Risk, Land Use, and so on). In addition, a Program Management Systems Development Initiatives (PMSDI) Committee was created to evaluate and make recommendations for program management improvements. These teams or committees are made up of personnel from each of the affected organizations, and are asked to make recommendations for the program by taking incremental development steps to achieve our end goals.

This collective energy can give an issue the broad analysis necessary to resolve complex problems. Issues are resolved at the appropriate level by developing the initial team communications. When concerns arise, those individuals who are most familiar with the process are involved in the resolution of the issue. This approach results in issues being dealt with more efficiently, with less lost time, as the solutions come from personnel who will be directly involved in the outcome. There is often little need to involve upper management in order to address or resolve an issue.

This approach has also been used for a variety of remediation alternatives, for example, all goals (jointly) set by EPA, the State of Idaho, and DOE were met on or ahead of schedule, two (2) Records of Decision were signed in fiscal year 1995, and the removal and transport of NaK waste to Argonne was accomplished. Some sites have been closed via a preliminary scoping process, two interim actions have been completed, and four removal actions were completed. The "decision process" options (Track I and II, RI/FS, IA, RD/RA, Removal Action) allow a variety of remediation alternatives, dependent upon the type of waste and clarity of the project scope.

The approach can be applied to a well-defined project, or a not-so-well-defined project. For example, through the use of life-cycle baseline maintenance a project's technical, cost, and schedule components are refined as information becomes available, rather than to force-fit an undefined project into parameters which are likely to change on completion of scoping investigations.

Regularly scheduled Plan of the Week (POW) Meetings have been established to provide a forum for identifying and discussing technical and program management issues at a level at which they can be more rapidly and effectively dealt with by senior management.

Similarly, Remediation Project Manager (RPM) weekly POW meetings provide a forum for presenting, discussing, and resolving issues with regulatory management.

As an outcome of these meetings, white papers or guidance papers have been created to establish or clarify the ER position on such technical or programmatic issues as LDRD, land use, Eco Risk, and so on.

Disciplined Approach

The INEL ER Program uses a disciplined approach to manage the program, and each individual part of the program, as a whole. The disciplined approach takes advantage of all of the major approaches in the program:

goal oriented, teamwork, process oriented, flexibility, and incremental development.

The INEL ER Program manages its scope, schedule, and budget to an approved baseline. This baseline can only be changed by the change control process. The costs used in the program baseline and in changes to that baseline are controlled based on bottoms up cost estimates.

BENEFITS OF "THE INEL APPROACH"

Keys to Successful Baseline Development and Improved Program Management

Successful baseline development and management was achieved through the focused principles of teamwork, communication, and consensus building providing the program with unity. The baseline enables the program to function as a fully coordinated, cohesive, and mutually supportive organization. Frequently recognized benefits provided by the baseline are:

- Programmatic control via inter-contractor group(s) constantly working to resolve/eliminate issues; routine application of outcome-based approach/strategy
- "Program Level" Coordination and Control, implementing structured and consistent Configuration Management and Control of all three ER Baseline Elements
- Maximized Team approach - Seek to establish common understanding
- Up-front discussion and resolution/solution to process issues and barriers; i.e., Determination of what needs to happen - what needs to be in place

Tangible Results

A few of the challenges and successes of the INEL Environmental Restoration Program have included: a) completion of all enforceable milestones to date, b) acceleration of enforceable milestones, c) managing funds to reduce uncosted obligations at year end by utilizing greater than 99% of FY-95 budget, d) an exemplary safety record, e) developing a strategy for partial Delisting of the INEL by the year 2000, f) actively dealing with Natural Resource Damages Assessment issues, g) the achievement of significant project cost reductions, h) and implementation of a partnering charter and application of front end quality principles.

CONCLUSIONS

Share in order to experience improvement. Experience has shown us the benefits of increased understanding through the sharing of information. All successful processes and tools, in and of themselves, are valuable lessons to share. Always recognize that good communication and feedback are key elements in the continuous improvement process. The interaction and discussion that takes place in sharing information with others, is an invaluable means of renewing and validating a programs operating philosophies.

A successful program embodies the successful integration of these operating philosophies along with the program's cost, schedule, and technical work scope. Using this approach, all program elements can be established and maintained from a central point of control. From that standpoint, the process of maintaining a balance between the functional elements (cost, schedule and technical) and program goals and priorities is easily achieved.

LIVING WITH THE UK STRATEGY FOR THE DEEP DISPOSAL OF RADIOACTIVE WASTE IN A MORE COMMERCIAL ENVIRONMENT

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ABSTRACT

This paper provides an update on the UK program to dispose of low level waste (LLW) and intermediate level waste (ILW) in a deep facility in light of the outcome of two major Government reviews and a public hearing process considering the program's next phase. The reviews concerned the future of the UK Nuclear Industry and radwaste disposal strategy; the results of both were announced in the Summer of 1995.

The radioactive waste policy review concluded that disposal is the preferred option to indefinite storage, and that the program to implement it should go ahead as soon as practicable, to fulfil the requirement for a policy of sustainable development. The nuclear industry review looked at the structure of the industry, including the proposal that the nuclear generating utilities could be released from Government ownership and become privatized.

Nirex is the UK company responsible for implementing Government policy on LLW and ILW disposal and is owned and financed by the companies of the nuclear industry. Therefore the outcome of both reviews is of particular importance to it as it increasingly faces commercial pressures in providing a disposal facility for the nation's waste.

The paper summarizes the outcome of the reviews on LLW and ILW disposal policy and in particular covers the following points:

Nirex's plans to develop a deep disposal facility at Sellafield. The next stage for this is an underground rock characterization facility. The granting of permission to construct this is on the critical path for a repository and the proposal is currently undergoing a public hearing process;

Nirex's financing and organization structure and how this will operate in the future to be compatible with Government policies on disposal and the environment, in particular ensuring maintenance of the "polluter pays" principle. A description of the repository pre-financing arrangements and the potential tariff structure is given;

how a cost-effective deep repository system is being developed which is "optimized" in terms of:

technical requirements to deal with particular wastes; the cost of disposal;

safety - in light of revised proposals concerning the safety principles by which a future repository should be licensed and authorized to operate.

INTRODUCTION

Nirex, founded in July 1982, is the company in the United Kingdom responsible for providing and managing new national disposal facilities for solid ILW and some LLW. The Company's present shareholders are BNFL and Nuclear Electric plc, each with a 42% share, and Scottish Nuclear Ltd. and the Government Division of the UK Atomic Energy Authority, each with a 7% share. In addition, the Government owns a special 'Golden Share', which safeguards the Company's control of the repository site.

The Government, under independent advice from the Radioactive Waste Management Advisory Committee (RWMAC), has established national strategy

for the development of such facilities, and provided a legal framework within which safety and environmental impact must be assessed and examined in public. Since 1987, Government policy has been that ILW generally should go deep underground, and that the same facility could be extended to take LLW also, as a cost-effective alternative to developing a "greenfield" shallow site.

THE RADIOACTIVE WASTE REVIEW

Between May 1994 and July 1995 the UK Government carried out a review into radioactive waste management policy (the Radwaste Review), undertaken by the Department of Environment (DoE). That review was complementary to the Nuclear Review being carried out at about the same time by the Department of Trade and Industry, which considered the future of the nuclear industry in the UK.

The Radwaste Review encompassed three major aspects:

- a general review of radwaste management policy by DoE;
- a review of the requirements for the authorization of disposal facilities by Her Majesty's Inspectorate of Pollution (HMIP); and,
- a more specific review of the approach to site selection and human health criteria by a Study Group, drawn from the advisory bodies RWMAC and Advisory Committee on the Safety of Nuclear Installation (ACSNI).

The Study Group published its findings in early 1995 (1), the final conclusions of the Nuclear Review were published in May 1995 (2) and those of the Radwaste Review in July 1995 (3).

The outcomes are summarized as follows.

Ownership of Nirex

Under the Nuclear Review, parts of two shareholders (Nuclear Electric and Scottish Nuclear) may become privatized, and will be integrated into the existing Nirex shareholding structure.

This may mean that they will be able to exert "a substantial commercial influence on the affairs of Nirex". In order to ensure that its policies will be carried out, the Government has decided to retain its special share in the Company.

Regulatory System

This will be streamlined and revised guidance will be issued to potential disposers of radioactive waste. In addition, the Government will consider the responsibilities of the different regulatory bodies and other organizations for assessing the doses received by members of the public.

Policy Aims

The Government has concluded that the policy aims for radioactive waste management should be revised and updated to emphasis the respective roles of Government, regulators and producers and owners of waste, and apply the concept of sustainable development and its supporting principles.

Waste Categorization

Further consideration will be given to refining the categories of radioactive wastes in terms of their half-lives and activity, and to the proposal that short-lived ILW might be disposed of to the near-surface disposal facility at Drigg.

Early Applications for Authorization to Dispose

Developers of major projects may submit early applications for disposal authorizations at the same time that they seek full planning permission (under zoning laws) for the project. The regulators will then be able to make decisions on authorizations before major commitments have been made.

Disposal of Solid Waste

The Government believes that both estimates of risk and other factors of a more qualitative nature will need to be considered in determining whether a disposal facility is safe. However, it believes a risk target of 10-6 per year (fatal cancer or a serious hereditary defect) is appropriate. There should be no prescribed cut-off for the period over which risk should be assessed.

The Government welcomed the RWMAC/ACSNI Study Group's report on site selection criteria for radioactive waste repositories, but has reservations about some of the detailed proposals made.

Spent Fuel Management

The question of whether and when to reprocess spent fuel and decisions on the siting of dry stores for spent fuel should be for the commercial judgement of the operators.

High-Level Waste

The Government believes that geological disposal on land is the favored option for vitrified HLW once it has been allowed to cool and is putting in hand development of a research strategy.

Partitioning and Transmutation

The Government will continue to watch with interest the results of international research on partitioning and transmutation, but has no plans to initiate further development of its own.

The Nirex Repository

The Government continues to favor a policy of deep disposal rather than indefinite storage of ILW and considers it appropriate that Nirex should continue with its program to identify a suitable site. There would be no advantage to be gained from delaying the development of the repository itself and it should be constructed as soon as reasonably practicable.

Interim Storage of Intermediate and Low-Level Waste

Waste must be treated as necessary to improve storage conditions.

Decommissioning

Decommissioning nuclear power plants should be undertaken as soon as it is reasonably practical to do so, taking account of all the relevant factors. Segregated funds for decommissioning should be established for those parts of the industry which are privatized.

Waste Substitution

The Government reaffirms its policy that the wastes resulting from the reprocessing of foreign spent fuel should be returned to the country of origin. It accepts that this policy can be implemented by waste substitution arrangements which ensure broad environmental neutrality for the UK.

Import and Export of Waste

Radioactive waste should not be imported to or exported from the UK other than in the specific cases.

Small Users

The Government will not direct small users to particular routes for disposal of their radioactive waste, but will leave them free to make their own arrangements.

Research

Each of the component parts of the industry should continue to be responsible for research and development necessary to support their respective functions.

THE NIREX PROGRAM

Nirex's current focus is on surface-based geological investigations to determine the suitability of a potential deep ILW repository site near Sellafield, Cumbria, in north-west England.

Nirex's next step is to construct a deep underground laboratory (rock characterization facility, or RCF) shown in Fig. 1. This proposal is similar to other countries' deep experimental facilities and along with those being constructed at Yucca Mountain in the US and the Gorleben mine in Germany, is intended to be a site specific research facility. The Company's application for planning permission to construct the RCF, submitted in July 1994 (4), was refused in December 1994 by the Cumbrian Local Government. This decision has forced the Company to take its case to a public inquiry, the outcome of which is expected in 1997.

Fig. 1

When started, the RCF program will last at least 5 years. Subject to a successful outcome from these investigations, Nirex will submit a planning application for the 650m deep (below Ordnance Datum) waste repository (DWR) (Fig. 2), this will be the subject of a further public inquiry. The deep repository (5) will be capable of taking 200-275,000m3 of waste.

Fig. 2

FUNDING REQUIREMENTS

The DWR project is a typical large scale construction project as the bulk of investment is required upfront before any revenues can be earned. However the DWR is a fairly unique project due to the long construction timescales involved, which result in the accumulation of financing costs up to four times the actual cost of the project. The profile of cost and revenue cash flows is illustrated by the graphs Figs. 3 and 4 respectively.

The funding requirements of the DWR project are (1996 prices):

almost 2 bn between 1986/7 and the planned opening date;
between 10m to 20m per annum over an operating period of 45 years;
approximately 50m over a 5 year closure period; and
additional cover for unforeseen expenses in the post closure period.

Fig. 3

Fig. 4

CURRENT FUNDING ARRANGEMENTS

As noted above, Nirex is owned by four shareholders, all of which are currently in the public sector. Their relative shareholdings only broadly reflect their waste emplacement requirements on a volumetric or other attribute basis. The four shareholders currently pay for all Nirex costs on a monthly basis according to their respective shareholdings in the Company.

However, the four shareholders are not the only ones who have a requirement to emplace waste in the DWR, in all Nirex will have seven Customers. This raises an important issue of how to charge all seven Customers for emplacing waste on a fair and efficient basis. This also implies financially compensating those Customers who have been the sole providers of funds to Nirex in the past.

Nirex could recover costs by invoicing seven Customers as opposed to the current four based on the amount of space required in the repository. However, other parameters which reflect the activity or potential harmfulness of waste may be more appropriate.

It is possible to design a comprehensive invoicing system to fund the DWR project, however, this system of funding is difficult to adapt to

changing repository design, cost conditions and emplacement requirements. This had led to Nirex adopting a more flexible tariff mechanism to recover costs which is discussed in a later section.

EMPLACEMENT CONTRACTS

The Emplacement Contracts will form part of the legal framework around which any future form of funding will be based. The Emplacement Contracts will essentially

- secure the long term funding of the DWR project by requiring Customers to sign formal agreements to finance their share of costs over a specified period;
- enhance the economic viability of the project, through the achievement of repository design optimization;
- introduce more market based disciplines to the project; and
- increase accountability to Customers by making the activities and cost base of Nirex more transparent.

The contracts will therefore:

- allow Customers to gain increased influence in the general affairs of Nirex;
- increase certainty from the project's viewpoint by securing funds;
- improve accountability for the Government through greater transparency; and
- impose 'value for money' for Nirex Customers and Shareholders.

These contracts will be essentially on a take-or-pay basis, whereby Customers will still be liable to pay for any space (used or not) for which they are committed. Customers will be given the opportunity to change their initial bids in order to provide a situation whereby all parties are agreed and the DWR costs are optimized - based on the latest technical information, before the contracts are signed. Customers who refuse to sign Emplacement Contracts will not be guaranteed use of any Nirex facilities for disposing of their waste in the future.

Customers requiring more space in the future will not have any guaranteed supply from Nirex, as Nirex does not intend to retain any spare capacity. Instead, the Customer will be obliged to purchase space from others with surplus capacity. If Nirex extends the DWR for any additional requirements; even though the marginal costs of DWR extension are lower than the tariff, the Customer requiring space will be liable to pay the full tariff. Any additional revenue gained by Nirex will be distributed to other Customers in an equitable manner.

THE NIREX TARIFF

The Tariff

A tariff has two major advantages:

i) it is a self balancing funding mechanism which can allocate funds on an equitable basis between initial participants; and

ii) it enables cost savings to be apportioned in a fair way and therefore provides an incentive to achieve cost savings and value for money for Customers through optimizing the DWR project.

The key principles underlying the tariff structure are:

- securing financeability by charging tariffs which recover the full lifetime costs of the project, including financing costs - this is achieved through the application of a discount rate within the tariff calculation;
- to allocate costs amongst customers in a fair and equitable basis;
- to derive tariffs which reflect the cost of emplacing different waste streams and therefore provide efficient pricing signals to Customers; and

designing the tariff to be simple enough to be readily understood by all Customers.

However, it is not practically possible to pursue these objectives in a purist manner as some may conflict. For example, efficiency implies the charging of cost-reflective tariffs which in turn suggests a marginal cost pricing regime. However, in the case of the DWR project, approximately 50% of all costs are currently classed as common and therefore cannot be recovered if the principle of efficiency is pursued to an extreme.

Cost Drivers

The Nirex cost allocation work forms the basis upon which to charge for costs incurred, in an efficient manner. This is achieved by determining the amount each Customer is liable to pay by relating charges to the cost drivers of the project. Based on the current DWR design, Nirex has identified the following five main cost drivers:

- the number of packages to be emplaced;
- the displaced volume of waste to be emplaced in unshielded vaults - this determines the size of vaults required to be excavated;
- the displaced volume of waste to be emplaced in shielded ILW/LLW vaults;
- the amount of long-lived waste to be emplaced; and
- emplacement of specific 'problem' radionuclides - direct charges for any projects which have been conducted to analyze specific radionuclides such as Chlorine 36 and Iodine 129.

Tariff Parameters

The tariff parameters which are based on the above cost drivers are listed below:

a - a general charge per package to be emplaced; plus either:
b - per m3 of vault space for LLW or shielded ILW package; or
c - per m3 of vault for unshielded ILW package.

Plus:

d - per TBq of long-lived radionuclide content;

Plus:

e - per MBq of 'problem' radionuclides

Plus:

any surcharges identified in the future.

In choosing Tariff Parameters, Nirex has recognized that the calculation of tariffs should not impose extra burdens on Customers over and above those already needed by Nirex for technical/safety reasons. For example, there are cases where waste has been stored in such a way that the Customer cannot be exactly sure what a particular package contains. However, at emplacement Nirex must be able to satisfy itself that it has a justifiable record of what each package contains. Whether package contents are known precisely, or whether they are estimated, the payments under the tariff will be based on this emplacement record requiring no additional effort on the part of Nirex Customers.

If at any stage the measurement of any of the parameters included in the tariff ceases to be needed, then it is not necessarily the case that its measurement should continue simply to enable the tariff to be calculated. The tariff is intended to be cost reflective and as such draws on existing technical waste classification measures. If these change it might well be appropriate for the tariff structure to change as well, recognizing the new technical view of which waste attributes are most

important in driving the DWR program. This will be handled by the change control mechanism in the Emplacement Contracts.

With regard to the allocation of costs between long-lived waste packages, Nirex has chosen activity (measured in TBq) as its base measure but alternatives are being reviewed The problem with using activity is that it may complicate negotiations as agreeing the reference time will be key.

The year of maximum post-closure risk may be an alternative method of allocating long-lived waste costs. This will discriminate against those radionuclides which are most influential in terms of success of the safety case. However, peak risk years are likely to be subjective in choice as they are determined by modelling different scenarios in the future. Again this is likely to lead to complex negotiations between Nirex and Customers due to the subjectiveness of the measure.

Common costs of the project may be recovered by pro-ratering them across all or some of the tariff attributes which itself be sensitive to the method of pro-ratering employed.

Surcharges

In addition to the tariff parameters which reflect the DWR cost drivers, a system of penalty payments or surcharges for waste packages which fail to meet Nirex requirements is envisaged. Under this structure a 'standard package' will bear no charge with respect to these attributes but any packages which exceed a given norm for these characteristics will be liable for an additional charge. This is designed to reflect those attributes which do not influence the project program or the base design and therefore should not form parameters in the tariff itself, but would cause operating or safety case difficulties. Imposing a surcharge will discourage such non-conforming packages from being presented for emplacement. Therefore, surcharges are consistent with the tariff principle of efficiency as only the additional costs associated with a 'non standard' package, is charged to that package. An example may be a surcharge on a package with excessive heat output as it may require specialist handling equipment.

The advantage of this structure is that surcharge items need not be resolved at this stage. The tariff will be calculated to recover program costs using the basic parameters: any revenue from surcharges being additional. The structure of surcharges will be developed as more information becomes available, and their level can be determined when more information on the technical implications of non-conforming packages is known.

Advanced Funding

As Customers will only be charged at the time of waste emplacement, Nirex requires advanced funding in the period up opening in order to cover its costs. The amount of advance payment that Customers are allocated will be calculated from certain waste attributes which are present in their waste emplacement schedules. These waste attributes which are identified by the cost drivers and tariff parameters, include the number of TBqs of long-lived waste and the number of packages of shielded and unshielded waste to be emplaced.

Advance payments are calculated in the following manner:

- firstly, each Customer's share of attribute is calculated;
- secondly, Customer shares of each attribute are multiplied by the total cost cash flows of each attribute in order to determine each Customers' cost share for all attributes;

the addition of cost cash flows for each attribute gives the total cost payable by each Customer each year. This is the amount of advance payment that each Customer will be expected to pay in each year to opening;

the proportion of advance payment allocated to each attribute will be simply the proportion of total cost associated with each attribute each year, e.g. if attributes A, B and C account for 60%, 20% and 20% of Customer.

The tariff aims to make Customers indifferent between funding in advance and receiving rebates in the future. The tariff will accrue advanced payments at a pre-determined rate of return that reflects project risk and repay them to appropriate Customers through a rebate at the time of emplacement.

Customers who choose not to pay in advance will be liable to pay the full tariff at the time of emplacement. This tariff will be determined by the rate of return (which will be the same as that for accrued advance payments), as the rate of return is equitable to the cost of borrowing. This break-even implies that the unit price payable for each tariff parameter is set such that the discounted value of the revenue stream associated with that particular attribute equals the discounted value of the cost stream allocated to it. This ensures that financing costs incurred as a result of the mismatch in timing between expenditure and receipt of income are recovered.

Payment Scenarios

The two payment scenarios are summarized below:

advance payment plus emplacement tariff minus rebate - where the rebate equals advance payment adjusted to account for project risk at an appropriate rate of return;

zero advance payment plus emplacement tariff minus zero rebate;

These are consistent with the principles of fairness and simplicity. The overall DWR project share allocated to each Customer may be attained by discounting each Customer's overall tariff payments.

The Rate of Return chosen for the project is key to determining whether Customers elect to pay in advance. If the rate is seen to be too low, then certain Customers may elect to invest in other projects rather than in the DWR project in order to earn a higher rate of return. However, it must be stressed that the project is a joint venture between Customers and if some of the main Customers elect not to pay in advance for financial reasons, then the project may not progress as funding from financial institutions is unlikely to be a feasible alternative.

CONCLUSION

Notwithstanding the requirement to face a public inquiry for the RCF, Nirex is making steady progress on repository development at Sellafield. This is not only in the scientific and technical areas but also in that of financing the project. A successful and early outcome to the next stage of Nirex's program at Sellafield, the RCF, will keep the Company on track for developing a deep repository early next century.

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PENNSYLVANIA'S LAND RECYCLING PROGRAM: IMPLICATIONS FOR CERCLA'S REAUTHORIZATION AND PROGNOSIS FOR SIMILAR PROGRAMS IN OTHER STATES

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ABSTRACT

Two of the biggest problems with the federal Comprehensive Environmental Response, Compensation and Liability Act ("CERCLA") are the policy that every site that is cleaned-up under CERCLA needs to be cleaned up to pristine conditions and the potential for virtually unlimited liability for parties connected with a contaminated site. As a practical matter, parties find that the stakes are often so high as to induce them to choose litigation first instead of going directly to cleanup. In addition to these issues, there is also the problem that CERCLA, being of national scale, is devoted to the cleanup of the worst sites. Many more sites -- the "brownfields" -- do not rise to the level of inclusion on the National Priorities List. The cleanup of these sites remains within the primary province of the states.

A number of states are increasingly recognizing the benefits to be gained from the cleanup of brownfields and putting these idle properties into productive use. One such state is Pennsylvania. With the adoption last summer of progressive legislation for brownfields remediation, Pennsylvania is now on the forefront of this environmental issue. Pennsylvania's land recycling program, as it is called, provides solutions to two of the problems that have hindered the effectiveness of CERCLA. First, Pennsylvania's program deals with the issue of "how-clean-is-clean?" by defining clean as remediation that meets any of three cleanup standards categories -- background, site-specific or state-wide health-based -- and it allows the choice of standard and level of cleanup to be determined by the proposed end use for the property. Second, Pennsylvania's program absolves parties who cleanup according to the program guidelines from further liability

Pennsylvania's handling of the "how-clean-is-clean?" and the liability issues promises to be a great stimulus to brownfields remediation in the Commonwealth and can also provide a model for brownfields remediation programs in other states and for amending CERCLA as well.

INTRODUCTION

Pennsylvania's Act 2 takes a common-sense approach to brownfield remediation. The approach addresses two of the major disincentives that had existed under Pennsylvania's previous land remediation policies and which also have created problems and delays for cleanups under the Comprehensive Environmental Response, Compensation and Liability Act ("CERCLA"). These two disincentives are: 1) the requirements and policies that would return the land to pristine condition regardless of the end use and 2) potentially never-ending liability for responsible parties. Pennsylvania's program -- informally called the land recycling program -- purports to solve these two problems. Act 2 provides for standards that

by law are protective of human health and the environment and which also allow for the consideration of future uses to which the land would be put. The Act provides owners or developers with cleanup standards based upon risk and end use. In addition, Act 2 provides for an end to further environmental liability when the remediation conducted by the parties attains the agreed-to cleanup standards and is certified by the Pennsylvania Department of Environmental Protection ("DEP").

The full program is actually authorized in three statutes. Act 2 is the land recycling program; Act 3 releases lenders from liability; and Act 4 provides funding.

PENNSYLVANIA'S APPROACH TO BROWNFIELD REMEDIATION

Pennsylvania's approach to recycling brownfields into productive use has four main and several additional elements. The main elements are:

- uniform cleanup standards based on health and environmental risk
- standardized administrative review procedures so that proposals submitted can be better prepared and reviewed and more readily approved
- releases from liability when the cleanup standards are met, and
- financial assistance.

Among the additional elements of the program are two that are designed to further expedite cleanups.

- elimination of the requirements for state and local permits for remediation activities undertaken pursuant to Act 2, and
- provisions for cleanups in "Special Industrial Areas" that limit the cleanup requirements to performing a baseline environmental assessment and remediating only direct and imminent threats.

Uniform Cleanup Standards Act 2 establishes three types of cleanup standards that, when achieved, will allow responsible parties to obtain the release from liability. These are:

- background
- state-wide health-based, or
- site-specific standards based on environmental and human health risks.

Act 2 became effective in Pennsylvania on July 18, 1995. With Act 2's effectiveness, both the opportunity to select the cleanup standards for a contaminated site and a mechanism for absolving the parties from future liability came into existence. The standards available for a site fall into three categories: 1) background, 2) state-wide health-based, or 3) site-specific. The background and the site-specific options became immediately available on the effective date of the Act. The existing state-wide standards in effect for soil (developed under the predecessor Hazardous Sites Cleanup Act) and groundwater (developed under the Clear Streams Law) also became available under Act 2 on an interim basis. The Act, however, directs the Pennsylvania Department of Environmental Protection ("DEP") with assistance from the Cleanup Standards Science Advisory Board ("CSSAB" or "Board") to propose a set of permanent state-wide health-based standards by July 18, 1996, and to adopt the permanent standards by July 18, 1997. Thus, the final, permanent state-wide health-based standards are being developed by regulation over a two-year timeframe. The key to the program in Pennsylvania is that a demonstration by the parties that a site has been remediated according to the pre-selected standard and a certification from the PA DEP will entitle the parties to a release from future liability for the site.

Anyone who wishes or is required to clean up a site and wants to obtain the liability protection must select and obtain compliance with any one

or combination of the background, state-wide health-based, or site-specific standards.

Background. The definition of "background" under Act 2 is the "concentration of a contaminant that is present at the site but is not related to any release of contaminant at the site." Thus, a developer who elects to clean up a site to background must document that the concentration of any contaminants remaining after the cleanup are not related to any releases of contaminants at the site. Act 2 therefore accommodates the situation where contamination could be present because it has been released from an off-site location. Anyone who chooses to use the background standard must meet background for each contaminant in each environmental medium. As mentioned, however, a person could choose to use the background standard for one contaminant in a medium and could also select either the state-wide health-based or the site-specific standard for other contaminants.

Site-specific. Anyone who chooses to use the site-specific standards can develop a cleanup level for the specific site based upon the contaminants, exposures and conditions unique to the site. Under Act 2 for the site-specific standards, certain guidelines must be followed depending on the type of contaminant and the medium involved. For example, for carcinogens, the cleanup level for soil and groundwater are to be established at levels that represent an excess cancer risk between one-in-ten-thousand and one-in-one-million. For toxic chemicals other than carcinogens, the cleanup level must be such as to prevent deleterious effects to the exposed population.

State-wide Health-based. Anyone who chooses to use the state-wide health-based standards can depend, during the interim period (until the permanent state-wide health-based standards are adopted), on the existing groundwater and soil standards. The permanent state-wide regulations under development by the CSSAB and DEP will provide a list of cleanup levels for various contaminants. These "medium-specific concentrations" ("MSCs") are the concentrations of contaminants associated with a specific environmental medium for potential risk exposures. Different exposure potentials such as residential or industrial settings will be reflected in different cleanup concentrations providing for equivalent levels of human health protection.

Thus, what the final state-wide health-based standards will do is round out the statutory requirements for the establishment of three categories of standards governing brownfield remediation in Pennsylvania. The availability of state-wide health standards, together with the opportunity to use background concentrations, site-specific standards based on risk, or a combination of standards, assures optimum flexibility on the part of brownfield site developers for obtaining the release from liability that is the impetus for this program.

Standardized Review Procedures

Legislators in Pennsylvania recognized the benefits to be gained from uniform submission and review procedures and incorporated uniform processes in Act 2. This uniformity makes it easier for site developers or their consultants to prepare submissions and to follow the steps necessary to remediate a site so as to obtain the liability protection. It also provides predictable opportunities for the public, local governments and others to participate in the remediation process.

While there is some variation in the review procedures depending upon the category of cleanup standards, the basic procedures for each of the

categories is uniform. The step that begins the process is the same for each type of standard -- filing a Notice of Intent to Remediate ("NIR") with the Department of Environmental Protection. Thus, the NIR must be sent to the PA DEP regardless of whether the developer is proposing cleanup to background, state-wide health-based, a site-specific standard or any combination thereof. Upon receipt of the NIR, the PA DEP will publish an acknowledgment in the Pennsylvania Bulletin.

The developer must also send a copy of the NIR to the municipality where the site is located and a summary must be published in a local newspaper.

The next step in the process is where the procedures differ depending on the standard selected. Except for the site-specific standard, there is no provision in Act 2 that requires DEP's prior approval for a remediation plan. However, prudent developers who have selected background or state-wide health standards will obtain DEP's concurrence before beginning the actual remediation since DEP's certification at the end of the process is required for the parties to obtain the liability protection.

For remediation designs based on site-specific standards, the developer must prepare and submit site assessment and remediation proposal documents which are subject to public input and DEP approval. These documents must be sent to the municipality and notice of availability must be published in a local newspaper and in the Pennsylvania Bulletin.

The additional steps begin with a thirty-day public comment period which also gives the municipality the opportunity to be involved in the development of the remediation and reuse plans for the site. Moreover, if the municipality requests a public involvement plan, the parties must develop and implement such a public involvement program. The requirement for a public involvement program carries with it additional obligations. That is, each time a remedial investigation report, risk assessment report, cleanup plan or final report is prepared, notice of its submission to DEP must be provided to the municipality, a notice summarizing its findings and recommendations must be published in a local newspaper, and the reports must also include the comments submitted by the municipality or the public and the developer's responses to those comments.

As described above, the Notice of Submission of the Final Report must be sent to the municipality, published in a local newspaper and published in the Pennsylvania Bulletin. Finally, where the background or statewide health-based standards are used, the PA DEP has sixty days to review the final report and either approve the report or respond with a letter of deficiency. If a site-specific standard is involved, the PA DEP has an additional thirty days -- for a total of ninety days -- to act.

These uniform procedures are designed to add to the administrative aspects of the program a degree of certainty that is intended to expedite review and approval and complement the certainty on the technical side that comes from having uniformity in the technical standards.

Limitation or Release of Liability

The key provision in Pennsylvania's land recycling laws is the opportunity for owners and developers to obtain a release from future liability. When the DEP approves the final report, i.e., determines that the site has been remediated according to the standards and the procedures in Act 2, the release from liability takes effect. A companion law, Act 3, extends the liability protection to project financiers and fiduciaries.

The requirement for final certification by DEP that a site meets the standard depends on the cleanup standard(s) chosen and carries with it certain restrictions. For all of the standards, attainment must be demonstrated by the collection and analysis of representative samples for the environmental media of concern viz., soils and groundwater. Moreover, the vehicle for obtaining the release from liability, common for all standards, is the final report as described above. There are important differences, however, and these are as follows. When background is the cleanup standard, institutional controls, such as fencing or land use restrictions, may not be used and a deed notice is not necessary. When a state-wide health-based standard is the chosen standard, institutional controls may not be used and whether deed notice is required depends upon the projected end use of the site -- deed notice is not necessary for sites meeting the residential health standards but is required where non-residential health standards (exposure factors) were used to comply. Finally, when the site-specific standard is used, deed notice is required and this notice must indicate whether residential or non-residential exposure assumptions were used in developing the site-specific cleanup standards.

Financial Assistance

The program has two avenues for financial assistance. These are funds established under Act 2 and Act 4 and administered by the Pennsylvania Department of Commerce.

Act 2 establishes the Industrial Sites Cleanup Fund. This fund is available to help innocent persons conduct voluntary cleanups. Through the fund, grants or low-interest loans are available for 75 percent of the cost of completing an environmental study and implementing a cleanup plan. Act 4 provides a funding mechanism through the creation of the Industrial Sites Environmental Assessment Fund.

Additional Program Elements

Among the additional components that enhance the workability of the program are the provisions that state and local permits are not required for remediation work under Act 2 and the provisions for sites in "Special Industrial Areas." Special Industrial Areas are either orphan sites or sites in "Enterprise Zones." For these sites, developers need comply only with the requirement to remove the imminent threat to obtain the liability protection. In other words, they need not comply with the three types of standards. For example, a developer need only remove leaking drums of hazardous material if these drums constitute the imminent threat.

ISSUES AND PROGNOSIS FOR OTHER STATES

The promise of Act 2 is that by providing a mechanism for parties to limit their liability it will provide the necessary encouragement to stimulate a high degree of brownfield remediation in Pennsylvania. The program provides a realistic approach to the remediation of sites that are too contaminated for further development but not enough to warrant a place on the National Priorities List.

As with any such statute, Act 2 does not provide answers to all of the implementation questions. Therefore, a number of important questions remain to be answered in order for the Act to be implemented successfully. These remaining questions not only produce challenges for the implementers in Pennsylvania but also serve as an indicator to legislators and implementers in other states of the types of issues with which they will likely be confronted.

Accordingly, this section of the paper is devoted to a discussion of several of the most important implementation issues that have been spawned by Act 2 in Pennsylvania. These issues are broad, if not universal, in their application as well as being difficult to resolve. First, however, it is important to present a brief description of the implementation of Act 2 in Pennsylvania.

It was discussed above that a key provision of Act 2 is the establishment of three categories of remediation standards -- background, site-specific and state-wide health-based. In the implementation phase, use of the background standard is a matter of demonstrating what background is (not necessarily an easy determination) and then demonstrating after remediation that the standard has been met on the target site. The site-specific approach, in a similar fashion, requires the development of an individual cleanup standard for the site based on risk assessment and the demonstrating that it has been achieved from the cleanup. At this writing, these two categories of standards are fully available for implementation -- in other words, the liability protection can now be obtained through either one or both of these two categories of standards. The full availability of the state-wide health-based standards, however, must await the development and adoption of the permanent standards by the Pennsylvania DEP with the participation of the Cleanup Standards Science Advisory Board. As mentioned previously, the Board's role is to assist the DEP in developing the state-wide health standards. Moreover, the Board's charge is to propose the standards within one year and adopt the standards within two years of the effective date of Act 2.

The current work of the CSSAB is particularly instructive because the Board is confronted with implementation issues that are universal in their application as well as being timely and difficult to resolve. By examining the workings of the CSSAB it is possible to gain an understanding of implementation issues that are not unique to Pennsylvania's program. How the Board and the DEP resolve these issues can provide very useful information in other venues. Thus, the following list of issues has been drawn from the list of major issues that has been developed by the CSSAB in Pennsylvania.

These issues were selected for discussion in this paper because of their importance to the successful implementation of Act 2 in Pennsylvania as well as their potential for broad-based application to other states' programs. The issues are:

1. What is a carcinogen?
2. Does a paucity of toxicological data for a particular contaminant absolve the regulators from the requirement to establish a standard.
3. Does the statute require and, if it does, how can ecological risk be incorporated into the development of remediation standards?
4. What is an aquifer?
5. How does a party demonstrate attainment of a numerical standard?

Taken in the numbered order, these issues are discussed below.

What is a Carcinogen?

Act 2 in Pennsylvania defines a carcinogen as a "chemical, biological or physical agent defined by the Environmental Protection Agency as a human carcinogen." The problem is that this reference to "human carcinogen" does not make it clear whether Act 2 is referring only to the U.S. EPA's Class A human carcinogens or to the suspected and possible human carcinogens in Classes B and C. This issue is not mere semantics; it has added significance because it is related to the second issue.

While it seems likely that the Pennsylvania legislature meant to include all of the chemicals in EPA's Classes A, B and C, this is not clear. Consequently, a reasonable interpretation of the definition in Act 2 would indicate that "human carcinogen" pursuant to Act 2 means the known carcinogens. To date, much discussion among the members of the CSSAB has been devoted to this issue.

It is clear that other states confronting this issue need to provide for the clarity that is lacking in Act 2 in Pennsylvania for the definition of human carcinogen.

What is the Effect of a Lack of Toxicological Data for a Contaminant?

For many chemical contaminants there is a lack of toxicological data. This is a serious problem for the implementers who are charged with developing the state-wide health standards. Simply stated, the issue is that the lack of a standard for any particular chemical contaminant means that there would be no mechanism for that contaminant in Pennsylvania by which a party can obtain the release from liability that is key to the effectiveness of the program. Having no mechanism for a full limitation of liability is clearly not in keeping with the spirit of the Act.

The lack of toxicological data for chemical contaminants is a universal problem. In Pennsylvania, the problem is intensified because of the way that Pennsylvania has established the mechanism for obtaining the release from liability. Any other state that would take a similar approach would be confronted with the same problem.

Does the Requirement to Incorporate Risk Include Ecological Risk?

As discussed above, Pennsylvania's approach to establishing the state-wide health-based standards is through an advisory board charged with providing assistance to the DEP. The essential guideline in Act 2 for developing the state-wide standards is to determine "the appropriate risk factors as needed to implement the provisions of [the] Act."

Thus, an issue that rises immediately to the fore is whether "appropriate risk factors" includes ecological risk. One of the policy declarations of Act 2 states that "cleanup plans should be based on the actual risk that contamination on the site may pose to public health and the environment." It should be clear to other states from Pennsylvania's experience that they may want to assure that their legislation clarify the ecological risk component. Moreover, if ecological risk is to be included in the development of the state-wide standards, the next issue is how to do it.

Groundwater Versus Groundwater-in-Aquifers

While this issue arises as a consequence of the wording of Pennsylvania's statutory definition of aquifer, it is not an issue of mere semantics.The legislative history of Act 2 clearly contemplates, as well as remediation of groundwater in aquifers, the remediation of contaminated shallow groundwater -- that is, groundwater that is by definition not in aquifers. Act 2 defines "groundwater" as "water below the land surface in a zone of saturation." On the other hand, the Act defines "aquifer" as a "geologic formation, group of formations or part of a formation capable of a sustainable yield of significant amount of water to a well or spring."

A major problem arises because of the way the statute requires the consideration of the inhalation factor in the establishment of state-wide remediation standards for "groundwater". For groundwater -- that is, groundwater not in an aquifer -- the development of an exposure factor that accommodates exposure through inhalation would incorporate inhalation factors much longer in duration than those for contaminated

groundwater in aquifers. The reason is that exposure through inhalation of vapors from aquifer waters would be expected to occur when the water is actually being used -- drinking, showering, cooking, etc. -- while exposure to inhalation from groundwater-not-in-aquifers can be expected to occur any time. The likely anomalous result is that the state-wide standard for groundwater-not-in-aquifers could be more stringent than the standard for water in aquifers.

One possible solution to this problem, discussed by the CSSAB, is to develop dual standards for each contaminant in groundwater -- one for groundwater-not-in-aquifers and a separate standard for groundwater in aquifers. The prospect of the Board's having to develop dual standards, particularly within the one-year time period allowed by Act 2, is daunting. It is clear, however, that this is a problem that other states would want to avoid.

How Should Attainment of a Numerical Standard be Demonstrated?

This issue is universal. In Pennsylvania, Act 2 provides that attainment (regardless of the standard) "shall be demonstrated ... through the application of statistical tests ..."

The problem confronting the CSSAB is that while its charge is to develop statistical tests only for the state-wide health-based standards, its work is actually the vanguard for demonstrating compliance with any of the other standards pursuant to Act 2. Whatever statistical methods the Board develops, they must be consistent with accepted statistical methods. In addition, it is to be expected that the Board's work will be applied across-the-board to demonstrate compliance.

As of this writing, the CSSAB has identified a range of options for an overall approach to specifying statistical demonstration methods. This range is from complete reliance on available guidance to developing very prescriptive statistical demonstration methods. Also as of this writing, the CSSAB has opted for a middle-of-the-road approach that involves a combination of developing anew a statistical model for the use of those who want to use it and also adopting existing guidelines developed to implement the predecessor to Act 2, the Hazardous Sites Cleanup Act.

IMPLICATIONS FOR THE REAUTHORIZATION OF CERCLA

The anticipated economic and environmental benefits from programs that encourage brownfield remediation are clear. A key question is whether the approach to brownfields remediation in Pennsylvania or any other state can have implications for the reauthorization of CERCLA. While CERCLA was up for reauthorization in 1994, the U.S. Congress was unsuccessful in reaching agreement to amend those aspects of CERCLA that have been widely criticized -- the provisions for joint and several liability, the provisions for retroactive liability, and the lack of a mechanism to tie the level of remediation to the end use of the property. The land recycling program as enacted in Pennsylvania and similar programs enacted and being considered in other states can provide valuable information for Congress to deliberate. Of course, the key provisions that promise to make Pennsylvania's experience a success are the opportunity for the parties to absolve themselves from liability and the mechanisms -- i.e., the three sets of standards -- established for doing so. Whether Congress is ready to adopt such radical changes to CERCLA remains to be seen.

SUMMARY AND CONCLUSION

In Pennsylvania, Governor Tom Ridge and his administration have enthusiastically backed Act 2 and have placed a considerable amount of stock in the implementation of the program. The CSSAB, a board of

thirteen members, has been aggressive in trying to meet its July 1996 deadline for producing the proposed state-wide health-based standards. By these measures, the program promises to be successful.

In reality, the success of the program will also depend upon the work of many sectors -- developers, municipalities, lenders, regulators and the public. Perhaps the most important factor for the effective implementation of the program is that each of these sectors has a vested interest in its success. Everyone stands to gain from a successful program. Moreover, as the program results in the cleanup of contaminated sites across Pennsylvania, both the economic sector and the environment stand to benefit. Thus, it can truly be said that Pennsylvania's land recycling program promises a win-win situation for the Commonwealth.

10-7

FINAL DISPOSAL OF LOW AND INTERMEDIATE LEVEL WASTE THE NORWEGIAN CONCEPT: A COMBINED STORAGE AND DISPOSAL FACILITY

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ABSTRACT

There are no nuclear power plants in Norway, only two research reactors. The Institute for Energy Technology (IFE) is operating these reactors. One is in Kjeller, 25 km east of Oslo, the other is in Halden, 110 km south-east of Oslo. IFE is responsible for all waste management actions in Norway and operates a waste conditioning facility at the Kjeller site, where all the conditioned waste is stored. Almost all conditioned waste in Norway is packed in 210 l steel drums.

The process to select a site for the disposal of low and intermediate level waste in Norway has been under way since 1989, when a committee was appointed for this purpose by a royal decree. The committee focused principally on disposal in existing abandoned mines and railway tunnels, and in March 1991 recommended that the abandoned Killingdal mine, in mid-Norway, should be used for disposal. A second alternative was to construct a new facility close to the waste conditioning facility which is located in Kjeller, outside Oslo.

In 1992, The Directorate of Public Construction and Property (Statsbygg) prepared its impact assessment for a repository for Norway's low and intermediate level waste in accordance with the Planning and Building Act. Three sites, the Killingdal mine together with Kukollen and Himdalen, both located close to the Kjeller area, were evaluated. The steering committee recommended Himdalen as the preferred site. The decision was made by Parliament in April 1994 that a combined storage and disposal facility should be built in Himdalen.

It is planned to build the facility in a hard rock formation about 50 m below the surface. The facility will be accessible through a slightly declining tunnel. Some plutonium bearing waste will be stored in a separate room in the facility. Before the year 2030, based on the knowledge and experience gained during the operational phase, a decision will be made whether the storage facility should be transformed into a repository or the plutonium bearing waste should be retrieved.

The operation is planned to start in 1996/97 and closure is foreseen for the year 2030.

INTRODUCTION

After the merging of the National Institute of Radiation Hygiene and the Norwegian Nuclear Energy Safety Authority in April 1993, the Norwegian Radiation Protection Authority (NRPA) was formed as regulatory body in Norway for both radiation protection and nuclear safety. NRPA is divided into four departments, Nuclear Safety Department, Environmental Protection Department, Health Physics Department and Radiation Medicine Department. The Nuclear Safety Department is, among other things, responsible for the supervision concerning the safety and the radiation protection at the Norwegian research reactors, including the waste management and transportation of nuclear material.

In 1989, a committee was appointed by the government to investigate possible solutions for final disposal of all the Norwegian LLW and ILW (1).

In 1992, an Impact assessment was performed including three possible sites, with a recommendation for one of them. This was an engineered rock cavity facility, 25 km from the Kjeller waste conditioning plant (2).

During the Parliamentary committee work it was decided that it should be a combined storage and disposal facility, with storage of the plutonium bearing waste and disposal of the short lived waste.

THE NORWEGIAN NUCLEAR PROGRAM AND THE ORIGIN OF THE NORWEGIAN RADIOACTIVE WASTE

Norway does not have nuclear power plants, although there are two research reactors.

The Institute for Energy Technology (IFE) operates these two reactors located in Halden, 110 km south-east of Oslo and in Kjeller, 25 km east of Oslo. Figure 1 shows the sites on a map of southern Norway. IFE is an industrial foundation funded from the national budget and from commercial research programs. The main data of the two reactors are:

1. Halden Boiling Heavy Water Reactor (HBWR):

Max. Thermal output: 20 MW
Coolant: Heavy water
Moderator: Heavy water
Fuel: Enriched uranium dioxide
Cladding: Zirconium
Test fuel: Both uranium dioxide and MOX fuel is used
Started: 1959

Main research areas at the Halden reactor are reactor safety, technological research and development. Fuel testing and research on man-machine interactions are two important tasks.

The waste is mainly ion exchange resins. Tested fuel elements are returned to their owners.

2. JEEP II, at Kjeller:

Max. Thermal output: 2 MW
Coolant: Heavy water
Moderator: Heavy water
Fuel: Slightly enriched uranium dioxide
Cladding: Aluminum
Started: 1967

JEEP II is used to produce pharmaceutical products and irradiation services for medicine, industry and research. Neutron beams from the reactor are used to study the fundamental physical characteristics of solids and liquids.

Fig. 1

The origin of the Norwegian waste is summarized below:

Laboratory scale fuel reprocessing plant (operation 1961 - 1968)
liquid fission products and decommissioning waste

Examinations of irradiated fuel in the metallurgical laboratory II
cutting and grinding waste

Halden reactor
ion exchange resin

Isotope (pharmaceutical) production
all kinds of laboratory waste

Research
waste from IFE and other research institutions

Medicine
tracer examinations

Industry
solid sources from industrial applications

Scale deposits on production equipment from offshore industry in the North sea,
low specific natural activity.

Exit signs and ionic smoke detectors

Decommissioning
when the two research reactors are going to be closed down, this work will result in various types of LLW and ILW.

WASTE AMOUNT AND RADIOACTIVITY CONTENT

The low and intermediate level waste in Norway is presently conditioned and stored at IFE, Kjeller. Most of the waste is packed in 210 l steel drums, but also 800 l boxes are in use. An equivalent of about 2000 drums is in store.

In 1970 it was decided to bury the waste drums (210 l) that had been generated in Norway until then at the Kjeller site. 1013 drums and some other large waste components were disposed of in a 4 m deep trench, stacked in two layers and overfilled with 2 m of clay. There are no engineered barriers. About 230 of these drums contains solidified liquid waste from the laboratory scale fuel reprocessing plant at Kjeller. The liquid was mixed with concrete and placed into the steel drums with an inner coat of polyethylene (3). This waste will be retrieved, reconditioned and transferred to the new storage and disposal facility when it is available.

Between 50 and 100 drums are produced each year. About 80 per cent of the waste volume originates from the activities connected to the research reactor program, while industry and medicine produce about 10 per cent each.

Up to the year 2030 it is estimated that an equivalent of about 10 000 drums of low and intermediate level waste with a total activity of approximately 200 TBq including about 50 g of plutonium will have been generated.

SITING AND LICENSING PROCEDURE FOR A REPOSITORY

The siting process for a repository started in 1989 when a committee was appointed by the government. This committee prepared a report that described the waste and discussed safety criteria and principles for the disposal of all the LLW and ILW in Norway (1).

In its search for possible solutions for the disposal of Norwegian waste, the committee assessed existing rock cavities (mines, road tunnels, railway tunnels a.s.o.). The construction of a new installation specially designed for the purpose of disposal was also considered. The main

conclusion was to construct a repository in the abandoned mines at Killingdal near Rros, 430 km from Kjeller, as the primary solution. Secondary, the committee recommended a further investigation of the possibility to establish an engineered rock cavity repository near Kjeller.

Based on this document a more detailed siting procedure was started in an area within a distance of 25 km from the waste conditioning facility at Kjeller. A systematic site screening process was carried out, in which 52 possible locations were identified and then reduced to 13 technically comparable sites. Further meetings with municipals and local authorities to discuss future land use plans and potential environmental impacts, reduced the alternatives to two sites (Kukollen, Himdalen) (4). An impact assessment was performed for these two sites and the closed Killingdal mine covering aspects such as area planning, industry, natural resources, historical sites, recreational areas and wildlife (2). A number of public meetings were held in potentially affected municipalities, and an official hearing was conducted.

After this procedure one of the sites, Himdalen, was recommended to the Government. The Government sent the recommendation to the Parliament where it was treated by the Energy and Environmental committee. The committee forwarded an adjusted proposal to the Parliament, in which the plutonium bearing waste, was suggested to be stored instead of being disposed of. It would then be possible to monitor and, if necessary, remove or recondition it at a later date. All short lived waste was to be disposed of. On 28 April 1994 the Parliament decided that the facility should be a combined storage and disposal facility and that the investigations at the Himdalen site should continue.

The builder and owner of the facility is the Directorate of Public Building and Property (Statsbygg). They build and own most of the public buildings in Norway. Statsbygg has to apply for a building license, make a detailed description of the building together with a safety report. According to the Atomic Energy Act, the Norwegian Radiation Protection Authority (NRPA) will review the safety report and provide advice to the Government on whether to give a license to start building of the combined storage and disposal facility.

IFE will be the operator of the facility. They will have to make a safety report for the operation and apply for an operating license. The Government will issue the license to the operator based on recommendations from the NRPA.

IAEA-WATRP REVIEW

NRPA requested an international peer review through the IAEA Waste Management Assessment and Technical Review Program (WATRP). The scope of the WATRP review is; the approach to the selection of site, the technical concept (combined storage/repository in a rock cavern), the long term safety. The members of the review team are from Switzerland (chairman), Canada, France, Germany and USA. The report of the WATRP review team was published in December 1995 (5). The main conclusions are that based on the existing information the team believes that the Himdalen site, in combination with the engineering concept, can be suitable for the storage and disposal of the relatively small amount of Norwegian LLW and ILW. Comments from this review will be taken into account when Statsbygg is finalizing the design and in the safety report. Comments will also be taken into account when the license application is treated.

DESCRIPTION OF THE COMBINED STORAGE AND DISPOSAL FACILITY

The combined storage and disposal facility will be built into a hillside in crystalline bedrock. The rock formation on the Himdalen site is Precambrian Mylonitised Gneiss.
The facility will have 4 caverns for the waste packages and one slightly declining 150 m long access tunnel for vehicles and persons. A service and control room, with certain crevice functions for the personnel, will be built along the tunnel and there will be a visitors room inside the facility. The layout is shown in Fig. 2. The rock caverns will be excavated with at least 50 meters of rock covering. The geological covering is for protection against intruders, plane crashes etc. and it is not intended to be taken into account as a barrier in the long term safety calculations.
Fig. 2
In the storage and disposal caverns, a granular base will first be installed as an underdrain for the facility, and a concrete floor will then be constructed on top of the granular base. Gaps will be left between the concrete floor and the walls of the caverns to allow ground water inflows to enter the underdrain.
Walls will be constructed on top of the concrete floor and the short lived waste packages will be stacked in 4 layers within the walls. At various stages the voids will be filled with concrete to create a solid sarcophagus incorporating the concrete floor and walls. The roof of the sarcophagus will be shaped to shed infiltrating ground water and a waterproof membrane will be fixed to the concrete roof. Void spaces will be left between the sarcophagus and the cavern walls and ceiling. This would create a hydraulic cage around the sarcophagus. Infiltrating ground water would tend to flow around the sarcophagus rather then through it. A similar construction is planned to be created for the plutonium bearing waste if it is decided in the year 2030 not to retrieve it.
There will be two separate drainage systems. a) All water entering the cavern shall be drained into a tank. This will be analyzed for radionuclides before it is released. b) One system of separate channels will be built under each cavern. This drainage system should always be dry. If water is detected here it has been in contact with the waste. It should then be possible to repair the damage, at least during operation. The facility will be designed taking seismic activity in the area into account (6).
Operation is planned to start in late 1996, the operational period is expected to last until the year 2030, this would take care of all Norwegian low and intermediate level waste, including decommissioning of the research reactors and laboratories at IFE.
Based on the experience during the operational period and safety reports for the closure, it will be decided whether to retrieve or dispose of the waste in the storage part by encasing it with concrete. During operation it is not intended to retrieve any of the waste that has been placed into the storage facility.
The facility will be closed by backfilling the disposal caverns in such a way as to permit functioning of the drainage for a very long time because it is anticipated that the caverns will be maintained in a drained condition after closure of the repository. After closure there will be an institutional control period of 300 -500 years with restrictions on land use and monitoring the discharges. Inspections or maintenance activities inside the underground excavations are not planned. Remedial actions

would be possible, if such actions appear to be necessary on the basis of the radionuclide content in the drainage waters.

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Session 11 -- UTILITY SPENT FUEL STORAGE/LLW VOLUME MINIMIZATION

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THE LEGACY OF THREE MILE ISLAND, IMPLICATIONS FOR TODAY'S U.S. DEPARTMENT OF ENERGY CHALLENGES

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ABSTRACT

Over the course of the 16 year period following the accident at THREE MILE ISLAND - UNIT-2, much has been learned and volumes have been written regarding the cause and massive cleanup activities of the incident. Because of these "Lessons Learned", important changes have been made and the U.S. commercial nuclear industry is safer and more reliable as a result.

It is important to recognize that two major sources of information emerged from this event. First and foremost were the important safety issues that require immediate answers and the addition of the modifications to plants that these answers generated. Second and of considerable significance to the U.S. Department of Energy (U.S. DOE) in today's post-cold war environment are the frequently hard-won lessons involved with the recovery, clean-up, and defueling of TMI-2 and it's unprecedented transition into long-term, monitored storage.

While the production of nuclear weapons and the use of nuclear energy to produce electricity involve very different applications of nuclear technology, a comparison of the TMI-2 Recovery Project with the present DOE Weapons Complex activities identifies several strikingly similar challenges.

The cleanup of Three Mile Island, Unit-2 has been the most arduous and expensive program ever conducted by the U.S. commercial nuclear industry. Similarly, the defense weapons complex cleanup has been characterized as and is the most arduous program conducted by the DOE.

Perhaps the single most vexing area of similarity between the two projects is that, there was no one to turn to who faced a similar problem

set of such scope and magnitude. All ground was new ground. It is understood that the magnitude of the DOE situation is far greater then TMI-2 but the approach methodologies, programs and some of the resolutions can be directly applied to DOE activities. GPU Nuclear successfully managed the TMI-2 recovery program, is currently decommissioning its Saxton facility and is the successful operator of both a Pressurized Water Reactor (PWR) and a Boiling Water Reactor (BWR) commercial nuclear generating facilities. GPU Nuclear has maintained the capability to provide a complete computer based monitoring systems and simplify necessary systems to enhance safety for a long term storage configuration. This includes capability to evaluate the risks posed by the facility to workers and the public, while greatly reducing facility maintenance costs.

This GPUN experience is directly applicable to the massive DOE challenge for which long-term storage and monitoring of its facilities could be a cost effective alternative to costly decommissioning programs.

U.S. DOE/TMI-2 HISTORICAL PARALLELS

When reviewing the lessons learned from the TMI-2 Recovery Program, it is interesting to note the historical parallels which serve as a "common denominator" between the TMI-2 recovery and the DOE complex cleanup.

The U.S. DOE had a mission to produce nuclear weapons for the defense of the Nation and the Free World. The "Cold War" ended, the world changed and the mission was shifted from weapons production to facility stabilization and cleanup.

Three Mile Island, Unit-2 had a mission to generate electricity. The "Accident" occurred, the world changed and the mission was shifted from electricity production to facility stabilization and cleanup.

Both the U.S. DOE and TMI-2 were organized to efficiently fulfill the obligations of their respective missions. Virtually overnight, the missions of both organizations dramatically changed. With this shift came unique requirements and challenges which placed heavy demands on the existing organizational structures. Immediate requirements to stabilize facilities, protect worker and public health, develop new program plans and methods and do it with an immediate culture change have been thrust on both organizations.

The regulatory environment which governed the normal missions of U.S. DOE and TMI-2 was, in many cases, not well defined during unique time. This created a burdensome process of fitting existing rules and requirements into the context of the new mission objectives. In many cases it also meant the implementation of new regulations or rules. These were not only hard to learn and interpret but, in some cases, seemingly impossible to implement and still accomplish work.

SHARED CHALLENGES

While the production of nuclear weapons and the use of nuclear energy to produce electricity involve very different applications of nuclear technology, a comparison of the TMI-2 Recovery Project with the present DOE Weapons Complex activities identifies several strikingly similar challenges.

The cleanup of Three Mile Island, Unit-2 has been the most arduous and expensive program ever conducted by the U.S. commercial nuclear industry. Similarly, the defense weapons complex cleanup has been characterized as and is the most arduous program conducted by the DOE.

Perhaps the single most vexing area of similarity between the two projects is that there was no one to turn to who faced a similar problem

set of such scope and magnitude. All ground was new ground. It is understood that the magnitude of the DOE situation is far greater then TMI-2 but the approach methodologies, programs and some of the resolutions can be directly applied to DOE activities.

TMI-2 RECOVERY CHALLENGES

Decontamination and Dose-Reduction

The accident produced radiation and contamination levels that were unprecedented in the commercial nuclear industry. In the plant Auxiliary and Fuel Handling Buildings, radiation levels ranged from 50 to 5000 millirem per hour, with local "hot spots" up to 125,000 millirem per hour. Certain areas, such as the reactor coolant bleed tank rooms, contained radiation levels which exceeded 1,000,000 millirem per hour. Surface contamination levels, in some plant areas, ranged from 100,000 to 1,000,000 times greater than today's release limit standards. In every case, the Reactor Building radiological conditions were far worse. These harsh radiological conditions greatly exacerbated early plant stabilization and recovery operations because operator access to many plant areas was either extremely limited or denied altogether.

Due to the uniqueness and severity of the post-accident radiological conditions, GPUN developed a substantial degree of knowledge and skill with regard to large scale decontamination and dose-rate reduction activities. The following are highlights of the essential elements of this process:

A highly specialized and sophisticated Radiological Field Operations and Engineering Department was established in order to assure the health and safety of the work force and the public. This important function was essentially re-engineered from the ground up and the total work force exposure of less than 6500 person-rem, over the entire 16 year recovery period, is a testament to the success of the program.

The substantial utilization of remote technology was essential. This equipment was innovatively adapted to efficiently gather radiological and physical data from inaccessible plant locations, thus providing planners with vital information about plant conditions. Additionally, remote platforms were modified to accomplish the decontamination of these areas. The remote platforms literally saved thousands of person-rem.

The severe post-accident radiological conditions consisted of both very strong radiation fields and very high contamination levels. Effective remediation required the development of specialized equipment capable of "breaking down" the high radiation fields into several contributor sources. The more severe radiation levels were eliminated first. This type of equipment was particularly useful during system flushing because radiation levels could be monitored in real time and excessive flushing operations could be avoided.

Decontamination and dose-rate reduction activities are not series events. They had to be accomplished in a defined pattern in order to achieve optimum and lasting results. Considerations such as personnel safety, waste form, waste minimization, ventilation flow patterns, occupancy requirements, impact on adjacent areas, fire prevention, etc., played an important role in the planning and execution of this type of activity.

Due to the presence of high surface contamination levels and widely distributed high-energy beta radiation fields, work force personnel conducted their activities in a substantial array of protective clothing. While this clothing provided protection from contamination and beta

radiation, it also produced high levels of "Heat Stress" and dramatically reduced the amount of time that workers could safely remain in the work areas. To overcome this problem, the project team pioneered the development of an extremely effective and efficient "Heat Stress Control Program". The program and its supporting equipment provided the capability to significantly extend worker stay time and work efficiency while reducing heat stress, radwaste and radiation exposures.

All work in the TMI-2 Reactor Building involved the potential to interface with very high radiation fields. In order to accomplish efficient work in this environment, a "Command Center" system was developed. Using this system, the Reactor Building work force personnel were constantly monitored via closed-circuit video equipment and two-way radio communications. Supervisors, managers, safety & health specialists and radiological control personnel were able to constantly monitor workers and provide instantaneous resolution to questions and problems. This system was extremely effective in monitoring work and minimized worker exposures to both radiological and industrial safety hazards.

The capability to expeditiously and effectively get personnel into and out of, harsh radiological areas was essential. During the TMI-2 recovery, GPUN developed and implemented a highly specialized work center which was responsible for preparing personnel for entry into the Reactor Building, providing logistical support for entry teams and assisting in personnel egress. This work center, in close cooperation with the "Command Center", significantly enhanced the safety and efficiency of operations. At the same time, it greatly minimized worker contaminations, increased organization to contaminated laundry collection, radwaste collection and controlled the spread of contamination and hot particles. This method is still in use in other GPUN plants.

RADIOACTIVE WATER CLEANUP

The accident initially produced hundreds of thousands of gallons of contaminated water. The ensuing recovery and decontamination programs produced millions of gallons of highly contaminated water. All this created several , large-scale and difficult radioactive water clean-up challenges. The project team developed and implemented numerous, highly specialized, water processing systems which were successfully utilized to safely clean-up and dispose of all radioactive water resulting from the accident.

The accident-generated radioactive water varied considerably depending on its source, transportation path, final storage location and dilution factors. Reactor coolant samples obtained on the day of the accident were in the milli-curie per milli-liter range. Water in the Auxiliary Building sumps and tanks was generally below 100 micro-curies per milli-liter while water from the Reactor Building exceeded that figure.

A number of focused radioactive water processing campaigns were conducted in order to satisfy recovery program objectives. These campaigns included the following:

Processing of the Auxiliary Building water inventory.

Processing of the Reactor Building Basement water inventory.

Processing of the highly contaminated Reactor Coolant water inventory.

On-going processing of "Previously Processed" water which was continually recycled in support of decontamination operations.o On-going processing of Reactor Coolant water to maintain water clarity during defueling operations. Maintaining water clarity for defueling operations became incredibly complex due to very small fuel debris and the

proliferation of organic materials which became established in the reactor coolant system. Methods of organic control debris removal were developed and successfully utilized.

Specialized programs were developed for processing contaminated resins and sludge effluents from plant demineralizers and sludge removal operations from sumps and tanks.

Final processing and evaporization of all accident-generated water was 2.3 million gallons of water from TMI-2.

Full and complete training in all aspects of system operation were developed and implemented. Training was also included in all aspects of expected system performance and radiological concerns.

RADIOACTIVE WASTE MANAGEMENT

The TMI-2 accident created enormous quantities of widely varying radioactive waste forms which were well outside the normal capabilities of a commercial nuclear power plant. This waste generally resulted from water cleanup operations, decontamination and dose reduction activities and defueling. As a result of the accident, the project team developed substantial expertise involving all aspects of waste management. GPUN found that the following elements were crucial in the safe and proper control and minimization of radwastes:

Waste Characterization - Radwaste characteristics can vary substantially even though the waste originates from the same general area or was caused by a similar process. A complete understanding of the radiological and chemical constituents, as well as the physical processes and regulatory requirements, essential in developing the simplest organized approach. Analysis of huge banks of data was then formatted to support efficient processing, packaging, and disposal procedures and methodologies.

Process Control Program - The Process Control Program varied from a single document to a collection of many documents which provide the basic procedures for determining cleanup processes, methodologies, and specific end point or acceptance criteria. The Process Control Program was essential to the operation of a processing system so that optimum performance is achieved while assuring overall program goals were met. Well structured, simple procedures were essential to support the safe and reliable performance of a system and the proper handling of the resultant waste.

Training - The best equipment and procedures were of limited value if the workforce did not comprehend their significance and recognize the importance of compliance. Because of the variety of waste forms resulting from the accident, it was imperative that each and every member of the team receive a full and complete understanding of the waste streams within the plant and the programmatic and safety significance associated with proper handling and control. A series of specialized training programs were developed. To reinforce the importance of the training, sessions were attended by, and frequently presented by, senior management responsible for the different project teams.

Chemical Control - TMI-2 instituted an extensive chemical control program in order to prevent cross-contamination of radiological wastes with chemical wastes. This was again supported by meaningful training programs.

Packaging & Handling Radwaste Containers - The project team developed substantial experience in the safe and proper packaging, handling, storage, and shipment of hundreds of highly radioactive waste containers.

This also required the requisite knowledge of regulatory requirements and the interpretive ability to specify whether lease, purchase or construction of containers would meet the regulatory requirements for transportation and disposal.

Waste Container Radioactivity Content Estimation and Prediction - Highly restrictive state and federal transportation and burial regulations dictate strict limitations on radioactive material content within waste containers destined for disposal. It was equally important that radioactive material loading within process containers be maximized to achieve the greatest economy. This apparent dichotomy required the ability to accurately predict the amount for radioactivity within a container during loading operations. The production, shipment, and disposal of several hundred containers was successfully accomplished.

Explosive Gas Generation Determination - The energy deposition from radiation interacting with hydrogenous material can cause the liberation of hydrogen gas. This process can create potentially explosive concentrations of hydrogen gas within sealed waste containers. Federal transportation regulations prohibit the shipment of this type of explosive mixture. Proper processing controls, container selection, handling, preparation for shipment and prediction of gas concentrations were required to mitigate potentially dangerous situations.

Waste Disposition - Simply put, the TMI-2 goal was to find the best method for waste removal. Each specific waste type was evaluated and prepared for disposal after considering regulatory requirements, scrap value, processing and disposal costs. Construction of an "On-Site" Waste Handling and Packaging Facility with all of the associated controls and staffing costs was determined to be a cost effective investment. The facility paid back its construction and operating costs over a very short period of time.

Throughout the entire recovery process there were several notable and unique examples of radioactive waste handling, packaging, and shipment.

Underwater Loading of Shipping Casks - A special underwater processing system was constructed and placed into operation. The excellent shielding provided by this configuration permitted extremely high levels of radioactive materials to be captured by the system process vessels. The resultant cleanup vessel radiation levels were dangerously high (the highest loaded vessel contained approximately 158,000 curies). Methods and techniques were developed to load the processing vessels into a heavily shielded shipping cask under water. This unprecedented process was safely conducted throughout the operating cycle.

Specialized Transloading Shipment to Disposal - Disposal of several processing vessels required confinement in a specialized disposal container. Facilities to allow radiologically safe conduct of this process were not available at TMI-2. Working with a national laboratory, GPUN coordinated and directed shipment of 14 highly radioactive vessels to the laboratory facility. These vessels were then transloaded into the GPUN special disposal containers and shipped. This operation was unique to both the national laboratory system and the commercial industry.

TMI-2 Fuel Debris Shipment - During the defueling phase of recovery, the severely damaged fuel was placed into specially designed containers for shipment and storage. Due to the damaged nature of the fuel, a new generation of shipping cask had to be used to ship the fuel safely. This task required close interface between DOE, NRC, and GPUN in order to

successfully design, license, fabricate, test and operate a new generation cask.

TMI-2 DEFUELING

The most technically challenging aspect of the TMI-2 recovery program was the removal of the 100 plus ton volume of damaged fuel from the Reactor Vessel. Many concepts were considered to perform this operation most of which required considerable development. The developmental nature of otherwise attractive proposals raised many substantial questions and concerns. This led GPUN to develop a defueling concept which utilized operators on a rotating work platform above the Reactor Vessel using long-handled tools and underwater cameras, to accomplish the defueling operations. This approach also allowed for rapid solutions to unexpected problems without having to make extensive modifications to elaborate systems.

A full scale mock-up of the TMI-2 Reactor Vessel internals provided defueling team personnel with an excellent facility for training and tool/method evaluation. By training and refining techniques and equipment, enormous radiation exposure savings were realized. Additionally, this approach allowed the development of incremental improvements in tool design as operators became more familiar with the tools and could make suggestions for enhancements.

The fuel removal operation could best be characterized as "one gigantic technical hurdle after another". No sooner would the team solve a particularly vexing problem then the next one showed up. One central theme rang clear, "Keep it Simple". Generally, the less complicated the approach, the better. The following involved simple solutions to difficult problems:

Following completion of the removal of the loose debris which covered the top of the post-accident core, a very large, hard mass of resolidified core was exposed. This material proved impervious to the long handled tools developed to break it apart. A small commercial drilling rig that had been originally used to take bore samples of the damaged core was fitted with a solid-faced bit, allowing the team to break up the resolidified mass by drilling approximately 400 holes into the core mass. This process, which came to be called "Swiss Cheesing", was very effective.

Removal of small pieces of core debris was a problem. The original vacuum system did not operate satisfactorily. An air lift system (suction dredge) was fabricated and employed to perform vacuuming operations. This simple system was responsible for the most productive months of defueling during the entire cleanup program.

GPUN also designed a family of specialized canisters into which the damaged core debris was placed for storage and shipment. These canisters were able to accommodate the full range of core conditions, from intact assemblies to chunks, to extremely finely divided particles which resulted from the destructive defueling methods.

GPUN had extensive pre-accident experience in handling new and spent low enrichment nuclear fuel. This experience along with the DOE experience and others led the group to decide that canisters and a new generation shipping cask were required to safely ship the core.

GPUN loaded the specialized canisters with core material then loaded the canisters into the DOE owned shipping cask. GPUN then prepared the cask documentation and the fuel was shipped to the Test Area North (TAN) facility at INEL for R&D, storage, and eventual disposal.

The core was shipped in 342 canisters in 49 cask loads requiring 22 trains - all without any regulatory violations or major incidents over the four year defueling period.

POST-DEFUELING MONITORED STORAGE

With the recovery program of TMI-2 complete and successful, GPU Nuclear has since moved this facility to long-term monitored storage.

A remote monitoring system allows the owner to detect changes in facility conditions, particularly those that can lead to unsafe conditions for workers or the public. Since the TMI-2 nuclear plant was to be placed in a long-term monitored storage condition, installing such a system was important to minimizing facility costs.

GPU Nuclear successfully managed the TMI-2 recovery program, is currently decommissioning its Saxton facility and is the successful operator of a Pressurized Water Reactor (PWR) and a Boiling Water Reactor (BWR) commercial nuclear generating facility. GPUN has maintained the capability to provide a complete computer-based monitoring system and simplify necessary systems to enhance safety for a long-term storage configuration. This includes capability to evaluate the risks posed by the facility to workers and the public.

This GPUN experience is directly applicable to other industrial or laboratory situations for which long-term storage and monitoring of the facility is required or could be a cost effective alternative to costly decommissioning programs.

11-2

SPENT FUEL MANAGEMENT AT NORTHERN STATES POWER COMPANY

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ABSTRACT

The Northern States Power Company (NSP) has developed an extensive spent fuel management program to resolve storage limitations at the NSP nuclear plants. Amidst great political opposition and legal restraint, the program has successfully provided for the continuation of nuclear power at NSP.

INTRODUCTION

The disposal of nuclear waste has become a critical factor in the continued operation of several commercial nuclear power plants in the United States. Disputes over federal responsibility in accepting spent nuclear fuel have forced some utilities, including NSP, to seek interim storage solutions. As the litigation over storage and disposal alternatives wages on, spent fuel pools continue to fill and plants edge closer to premature shutdown.

This paper discusses the spent fuel management program at NSP, including the Independent Spent Fuel Storage Installation (ISFSI) project at the Prairie Island Nuclear Plant. It provides a historical overview of the program, discusses recent developments, and concludes with a prospectus of NSP plans for resolving the nuclear waste dilemma.

NSP Spent Fuel Management Program

NSP owns and operates two nuclear power plants: Monticello, a single unit boiling water reactor (BWR), and Prairie Island, a dual unit pressurized water reactor (PWR). Like many reactors built in the 1960s and 1970s, the NSP units were designed for limited and temporary spent fuel storage, with the expectation the fuel would later be removed and reprocessed. In 1977, a federal moratorium against reprocessing forced utilities to retain spent fuel at the plants. To accommodate the additional storage requirements, NSP attempted to increase the storage capacity of the spent fuel pools through fuel consolidation and redesign of the holding racks. The enhancements, however, were insufficient to allow the accumulation of fuel for an extended period of time.

The Monticello nuclear plant does not currently present a spent fuel storage problem. During the initial cycles of operation, fuel bundles were leased from the General Electric Company (GE). The spent fuel bundles were temporarily stored in the plant holding pool until storage capacity became scarce, at which point the fuel was returned to GE. In all later cycles, GE sold fuel bundles directly to NSP, which assumed responsibility for the storage and disposal of the spent fuel.

At Prairie Island, storage capacity was nearly exhausted, despite reracking the spent fuel pool twice. The redesigned storage pool, shared between the two units, could not support operating cycles beyond 1995. To sustain plant operations, NSP initiated a highly controversial effort to obtain approval for an ISFSI, a temporary onsite dry cask storage facility.

ISFSI Project

In the early 1980's, NSP anticipated a fuel storage dilemma and began evaluating options. In May, 1989, NSP announced its decision to temporarily store spent fuel in dry storage casks. The announcement initiated a long and exhaustive review process (Table I). After unanimous approval from the Public Utilities Commission (PUC) and the Nuclear Regulatory Commission (NRC), the construction of the ISFSI site began in September, 1992.

In June, 1993, court appeals by opponent groups forced a suspension of the preparatory work on the ISFSI site. At that time, the on-site construction was essentially complete and the production of the casks was progressing successfully. The Minnesota Court of Appeals determined the duration of storage would be in excess of eight years, which it ruled as permanent storage. Because Minnesota statute requires legislative approval for "permanent" high level and low level waste sites, further progress on the project required approval by the Minnesota Legislature.

In January, 1994, NSP submitted testimony before a number of Minnesota House and Senate committees requesting the legislature's ratification of the spent fuel storage facility at Prairie Island. Representatives attested to the safety of the ISFSI technology and the necessity of the facility in providing for the electrical, economic, and environmental needs of the communities serviced by NSP.

Minnesota Legislation Results

On May 6, 1994, the Minnesota Legislature enacted bill S.F. No. 1706: "relating to public utilities; providing legislative authorization of the construction of a facility for the temporary dry cask storage of spent nuclear fuel at Prairie Island nuclear generating plant; providing conditions for any future expansion of storage capacity; approving the continued operation of pool storage at Monticello and Prairie Island

nuclear generating plants; requiring development of wind power; regulating nuclear power plants; requiring increased conservation investments; providing low-income discounted electric rates; regulating certain advertising expenses related to nuclear power; creating a legislative electric energy task force; appropriating money; ..."

In essence, the bill provided for the continued operation of the NSP nuclear plants in the near term. However, it set forth the eventual abandonment of nuclear power in the state of Minnesota through mandates on alternative energy sources, economic regulations, and miscellaneous rate ordinances.

A total of 17 Transnuclear TN-40 storage casks, of the original 48 requested, were authorized for Prairie Island. The casks will be dispensed to the site in three intervals pending NSP's compliance with the legislative mandates. The elements of the bill regarding dry cask authorization are:

1. Five casks are authorized for immediate use.
2. Four more casks will be authorized if, by December 31, 1996:
   NSP has filed a license application with the NRC for an alternate ISFSI site off of Prairie Island in Goodhue County.
   NSP continues to make a good faith effort to implement the alternate site.
   NSP has constructed, contracted for construction and operation, or purchased an additional installed capacity of 100 MWe of generation from windpower.
3. Final eight casks will be authorized if, by December 31, 1998:
   NSP has constructed and operates, purchased, or contracted to construct and operate a total additional installed capacity of 225 MWe of generation from windpower.
   NSP has constructed and operates, purchased, or contracted to construct and operate an additional installed capacity of 50 MWe of generation from farm grown closed-loop biomass.
   An alternative ISFSI site in Goodhue County, Minnesota, is operational or under construction.

If NSP fails to satisfy the mandates for December 31, 1998, the Minnesota Legislature may revoke the authorization for the final eight casks, if a law ordering the revocation is enacted before June 1, 1999.

The bill also specifies that NSP and the Governor, acting on behalf of the state of Minnesota, must reach a contractual agreement on the cask plan. The provisions of the contract, as later established in the Energy Policy and Storage of Spent Nuclear Fuel Act, include:

1. Considerations for the siting of an alternative ISFSI site.
2. Additional renewable energy alternative requirements to be implemented, without regard to cost, by December 31, 2002:
   NSP must construct and operate, purchase, or contract to construct and operate a total additional installed capacity of 425 MWe of generation from windpower.
   NSP must construct and operate, purchase, or contract to construct and operate a total additional installed capacity of 125 MWe of generation from farm grown closed-loop biomass.
3. Designation of the Mdewakanton Tribal Council as the third-party beneficiary of the contract, with standing to enforce the agreement.

The legislative act further mandates:

4. Additional renewable energy alternative requirement to be implemented, under the resource planning process, by December 31, 2004:

NSP must utilize 400 MWe, in addition to the required 425 MWe, of generation from windpower if the energy need exists and the cost of windpower is the same or lower than non-renewable alternatives.

A large emphasis has been placed on the "temporary" designation of the casks. The spent fuel stored in the casks must be moved to an alternative storage site as soon as a site becomes available. For any casks remaining at Prairie Island after January 1, 1999, NSP must transfer $500,000 per cask, per year to a "Renewable Development" account.

In addition to the cask limitations, NSP must accept certain responsibilities for the newly established Legislative Electric Energy Task Force, assigned to study and make recommendations on future energy options. The bill specifies that NSP must provide:

1. "Dry Cask Alternatives Study" reevaluating alternative combinations to dry cask storage.
2. "Worker Transition Plan", issued to the Department of Jobs and Training, in the event the Prairie Island nuclear generating station is shutdown for more than six months.
3. "Nuclear Power Phase-out Plan" for the entire NSP nuclear generation sector.
4. "Decommissioning Plan" for the TN-40 casks once they are emptied of spent fuel.

Though the Minnesota Legislature has not ordered the immediate closure of operating nuclear plants, nuclear power will be under close scrutiny in the state of Minnesota. Except to accommodate the decommissioning of a nuclear power plant, the current storage capacity for high-level nuclear waste will not be increased. The construction of a new nuclear-powered electric generating station is strictly prohibited. The bill also states that any nuclear reactor unit which has an annual capacity factor of less than 55% for three consecutive years will be required to shut down and cease operations.

Status of NSP Compliance

Wind:

In accordance with the first legislative mandate for December 31, 1996, NSP must acquire an additional installed capacity of 100 MWe of generation from windpower produced within the state. NSP intends to develop a wind energy conversion system to accommodate this requirement. A Certificate of Need (CON), issued by the PUC, and a siting certificate, issued by the Environmental Quality Board (EQB), were filed in late September, 1994. NSP has issued a Request for Proposal (RFP) and is evaluating bids.

The wind conversion system will be implemented in phases. Phase 1, which has been completed, involved a contract for an existing 25 MWe from the Kenetech Company, which owns and operates 73 wind turbines in Lake Benton, Minnesota. Phase 2 entails the development of a 100 MWe plant, which has been contracted to Zond Systems Incorporated. Phase 3 will include an additional 100 MWe plant, for which the CON is currently before the PUC. The EQB will manage the siting of windpower plants and develop Environmental Impact Statements (EIS). After public review, the EQB will designate sites.

Alternative Site:

Another requirement of the first legislative mandate is Nsp must file a comprehensive license application with the NRC for an alternate ISFSI site in Goodhue County, Minnesota, and make a "good faith" effort to

implement the site. For the initial identification and evaluation process, NSP assembled a Public Advisory Committee (PAC) to serve as a liaison and represent diverse interest groups in the county. With the assistance of the Stone & Webster Engineering Corporation, NSP has identified two candidate sites in southeastern Goodhue County and has filed a Site Certificate Application to the EQB for further consideration. The EQB has begun an extensive siting process while NSP assembles information required for an NRC license. The application documents will be submitted to the NRC in 1996. Pending public intervention, the review process can require up to three years to complete.

Biomass:

In accordance with the second legislative mandate for December 31, 1998, NSP must acquire an additional installed capacity of 50 MWe of generation from farm grown closed-loop biomass. NSP has submitted an RFP to the PUC. An independent evaluator is evaluating bids and will make a recommendation to NSP.

RECENT DEVELOPMENTS

ISFSI Site Implementation

On May 23, 1995, NSP placed the first loaded TN-40 dry storage cask on the Prairie Island ISFSI site, which allowed a refueling outage of Unit 2 to proceed as scheduled. The second and third casks have also been loaded and placed on the holding pads to allow for refueling of both units. The remaining two authorized casks will be delivered to the plant at monthly increments. Each cask is meticulously inspected upon arrival then immediately loaded and placed inside the heavily secured facility.

Utility Lawsuit

A group of 14 utilities, led by NSP, are pursuing a lawsuit against the U.S. Department of Energy (DOE). The suit was filed on June 20, 1994, in the U.S. Court of Appeals in an attempt to clarify the DOE's obligation regarding spent nuclear fuel. NSP is joined by several other utilities, public agencies, and state governments nationwide which believe the federal government has a contractual responsibility to accept commercial spent nuclear fuel beginning in 1998. The lawsuit has been admitted for judicial hearing and awaits final resolution.

Private Spent Fuel Storage Initiative

On March 9, 1995, the Mescelaro Apache Tribe passed a referendum to establish a temporary storage facility on the Mescelaro Reservation in New Mexico. A consortium of utilities and contractors have committed to developing a privately operated spent fuel storage facility for interim use until a federal repository becomes available. In addition to providing a temporary solution to the spent fuel storage dilemma, the facility would contribute to the self-sufficiency and economic diversity of the Tribe and the surrounding communities. Development of the Mescelaro facility would not relieve the federal government of its obligation to begin taking fuel from utilities in 1998.

In the near term, NSP will still require all 17 allotted dry storage casks at Prairie Island in order to maintain continued operation of the plant. Even if the interim storage project is successful, a storage facility in New Mexico would not be available until 2002. A permanent underground repository, under study at Yucca Mountain, Nevada, is not expected to be ready until 2010 at the earliest.

Dry Fuel Storage Conference

On July 25, 1995, NSP, in coordination with the Nuclear Energy Institute (NEI), hosted a industry workshop on dry fuel storage. Representatives from nearly two dozen utilities attended the forum to discuss and resolve problems observed in utility dry fuel storage projects.

The predominant issue discussed was inadequate quality assurance in the manufacturing and procurement of storage containers. Industry experience has demonstrated the process is remarkably complicated and susceptible to poor oversight. Additional concerns involved project licensing and field loading of the casks.

As a result of the NSP/NEI conference, the industry can better prepare for the challenges of dry fuel storage. The knowledge gained by the utilities and regulators may allow for improved management of spent nuclear fuel, increasing safety while decreasing costs.

NSP PROJECTED PLANS

The primary objective for the NSP spent fuel management program will be ensuring the uninterrupted operation of the nuclear power plants, particularly Prairie Island. Of immediate concern will be the implementation of the remaining two dry storage casks to provide for the next refueling outage. Prior to utilizing the casks, NSP is required to perform a substantial amount of training, rehearsal, and system checks in accordance with the NRC license for the ISFSI.

To prepare for future refueling outages, NSP will continue to implement the legislative mandates which provide for additional storage casks at Prairie Island. The next deadline of December 31, 1996, requires that NSP makes progress in establishing an alternate ISFSI site in the Goodhue County and installs an additional capacity of 100 MWe of generation from windpower. Plans to meet these requirements have been initiated and are progressing on schedule.

NSP will maintain close liaison with the DOE to accept spent fuel in 1998. There are a large number of utilities nationwide which produce a significant portion of their power from nuclear reactors. Many of these reactors, such as Prairie Island, were designed for only limited and temporary storage of spent nuclear fuel. The federal government accepted responsibility for the removal and disposal of spent fuel. In compensation, it collected funds, totaling over $11 billion, from the utilities and their customers. The Department of Energy, under the direction of Congress, has a moral and legal obligation to the utilities to uphold its commitment.

CONCLUSIONS

In a determined effort to obtain political and public acceptance of an ISFSI at the Prairie Island nuclear plant, NSP has developed an extensive and successful spent fuel management program. The program has revealed some of the areas of contention and concession in approving on-site dry cask storage, which may prove beneficial to other utilities requiring interim storage solutions. To provide for the continuation of nuclear power, NSP will persist in seeking solutions to the nuclear waste dilemma.

11-3

STATUS OF MULTI-PURPOSE CANISTER (MPC) PROJECT

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ABSTRACT

The Multi-Purpose Canister (MPC) project represents a cornerstone of the current DOE Office of Civilian Radioactive Waste Management (OCRWM) program for handling spent nuclear fuel. The MPC and associated support equipment is being designed to accommodate the requirements for not only storage and transport but also for the specified disposal requirements of the Mined Geologic Repository System. The Phase 1 design effort for the MPC system, being performed by the Westinghouse Team on behalf of the OCRWM Management & Operating (M&O), is on schedule for delivery of completed Safety Analysis Reports (SARs) in April 1996.

INTRODUCTION

This paper addresses the current status of the Multi-Purpose Canister (MPC) Project. The MPC Project was awarded to Westinghouse by TRW Environmental Safety Systems, Inc. (TESS) on behalf of the Department of Energy. TESS is the Management and Operating Contractor (M&O) for the DOE Office of Civilian Radioactive Waste Management (OCRWM). Westinghouse, as the lead subcontractor, is performing the design, licensing, testing and fabrication of a complete storage, transport and disposal system for handling spent nuclear fuel in a three phase program. The MPC Project represents a cornerstone of the current OCRWM program for handling spent nuclear fuel. The MPC, in conjunction with the Mined Geologic Repository System, represents the solution to the nation's spent fuel disposal dilemma.

The status of the design, Safety Analysis Report submittal to the NRC, preparations for fabrication of prototypes and out year plans are discussed.

BACKGROUND

The U.S. Department of Energy recognized the importance of dry spent nuclear fuel storage and took a critical step toward addressing this need by awarding the MPC contract in April 1995.

The Westinghouse Team includes Packaging Technology, Inc. of Tacoma, Washington, Chem-Nuclear Systems, Inc. of Columbia, South Carolina, Westinghouse divisions in California, New Mexico, Tennessee, and Pennsylvania and E.J. Bentz and Associates in Springfield, Virginia. Select domestic utilities are also supporting the Phase 1 system design and development as utility advisors to the design team.

The design activity and Safety Analysis Reports (SARs) will be complete by 26 April 1996, marking the conclusion of Phase 1 of the program. Phase 2 consists of the USNRC certification process for the 10 CFR 71 and 10 CFR 72 licensed components of the MPC System, associated testing and the fabrication of prototype hardware for system level demonstration and ultimate qualification of the MPC system for use by utilities and site operations. Phase 3 consists of the fabrication and delivery of the initial complement of approximately 150 Multi-Purpose Canisters and automatic welding/drying equipment. Both Phase 2 and 3 are contract options to be exercised as appropriate by the M&O.

Additionally, key reports and studies have been accomplished and submitted to TESS as required. These include the System Safety Report, Human Factors Engineering Report, Critical Operating Times (COT) and Reliability, Availability and Maintainability (RAM) Report, MPC System/Bare Spent Nuclear Fuel Transfer (BST) Interface Report, SAR Alternative-Enhanced Fuel Study, SAR Alternative-Stainless Steel Fuel Clad SNF Study, and the Depleted Uranium Shield Plug Study. Meetings with the utility advisors have provided valuable feedback used to influence these studies and the overall design process, from the ultimate users

standpoint, leading to the Preliminary Design Reports (PDRs) and Safety Analysis Reports (SARs).

MPC SYSTEM DESCRIPTIONS

The MPC system is being designed per requirements set forth in specifications provided by the M&O to contain SNF from commercial boiling water and pressurized water reactors (BWR, PWR respectively). The overall MPC system design concept is shown in Fig. 1. The MPC is designed in two (2) sizes: 125 ton and 75 ton. Because of the weight, the MPC is intended for rail shipment, however, in cases where the nuclear power plant or a potential interim storage site lacks rail access, the casks and system components are to include heavy haul means.

Fig. 1

The MPC has been designed incorporating highly corrosive resistant materials. Inside the canister is a metal frame or "basket" that has three functions: 1) structural support and criticality control for the SNF assemblies, 2) transfer of heat from the SNF to the canister walls, and 3) maintenance of the fuel in a totally safe condition during any normal, and postulated accident scenarios. Once loaded with spent fuel assemblies, the MPC will be permanently sealed by automatic welding, dried and backfilled with an inert atmosphere and then loaded into a cask for each phase of storage (whether it be at a nuclear power plant or interim storage facility), transport and disposal. Other MPC system elements that will be developed include an On-site transfer and Storage System for loading and on-site storage of MPCs, and for transferring MPCs into transport casks for MPC transport to interim storage or disposal sites.

MPCs are being designed with broad fuel acceptance parameters to accommodate all PWR and BWR assembly types and classes, specifically all Zircalloy clad fuels (except South Texas and CE 16x16 with control components). Stainless steel clad fuels can physically be accommodated by the design. The MPC design essentially handles all of the fuel in the projected inventory.

CURRENT PROJECT STATUS

Approach

The initial effort on the MPC project involved the preparation of the basic program guidance documents. These prescriptive documents (see Fig. 2) were submitted to and approved by TESS and form the basis for managing the Project.

Fig. 2

The Phase 1 MPC system design process was structured in a three step approach 1) concept optimization, 2) detailed design, and 3) SAR preparation. Concept optimization was scheduled for the first three (3) months of the project and consists of establishing the baseline configuration for an integrated MPC system. Once the baseline configuration was fixed, the second step was to complete the detailed calculations, perform thermal, shielding and structural analyses and prepare the design drawings required to fully define and specify the components. The final stage is to prepare the Preliminary Design Reports (PDRs) and SAR chapters in preparation for submittal of the SARs for USNRC certification review.

The concept optimization was completed on schedule and culminated in a two day formal Design Concept Review (DCR) and technical exchange. The DCR was attended by the DOE, the M&O and a number of utility advisors.

Feedback and comments from the DCR were factored into the baseline design process during the detailed preliminary design.

The key driver for the MPC system integrated design is the basket configuration and source terms. The four basic basket designs (12 assembly PWR, 21 assembly PWR, 24 assembly BWR and 44 assembly BWR) have been completed and are illustrated in Table I. Structural, thermal, criticality and shielding analyses are underway to facilitate the final detailed design of the remaining system components.

Table I

All the designs are based on the Design Procurement Specifications (DPSs) issued by the M&O, Based on the primary design parameters given in the DPSs, an MPC Functional Specification was prepared and issued for design team use. The MPC Functional Specification provides the extrapolation of the DPS to a system level design basis and provides additional guidance to the designers. For example, the MPC Functional Specification allocates weight parameters between system level components. System Specification Documents (SSD's) were then prepared to define the detailed specifications applicable to each configuration item being designed (see Fig. 3). All SSDs have been completed as well as the interfacing control points for each piece of hardware. Hardware interfaces are defined and controlled using an Interface Control Sheet mutually generated by each involved design agency and approved by the Chief Design Engineer.

Fig. 3

One unique feature of the Westinghouse Team design effort is the "concurrent engineering" approach being utilized. Concurrent engineering has been successfully used to integrate the disciplines of manufacturing engineering with the design engineering to enhance component fabricability and perform value engineering. This integration was implemented at the beginning of the project, has resulted in very active interchanges among the design team and prospective material suppliers, and has resulted in a design that can be built cost effectively. Design options are discussed with the fabricators to generate the most cost effective project and to help ensure a smooth transition from design drawing to the shop floor. Additionally, the goal is to maximize the use of standard processes and tooling, further reducing the cost to produce MPCs.

The MPC design process has also taken the approach to utilize many previously NRC accepted materials, analytical methods, acceptance criteria, etc., to maximize the potential for successful licensing of the Part 71 and 72 components. Four meetings have been held with the NRC to discuss the MPC design details and approaches as well as potential certification issues that have surfaced during the design process. This proactive approach has been beneficial in the timely resolution and selection of design options that have the best chance to be licensed within the constraints of the program schedule; i.e., deployment of the MPC by 1998.

The Preliminary Design Reports (PDRs) are segregated into three phases. The first PDR grouping covering auxiliary equipment has been completed and submitted to TESS. The remaining two (2) PDRs covering the storage unit and the MPC assemblies will be completed in the Spring of 1996 to complete Phase 1 of the project.

QUALITY ASSURANCE

All hardware items requiring USNRC certification under 10 CFR Part 71 and 10 CFR Part 72 are being designed by teammates with USNRC approved

quality assurance programs. The suitability of each teammates' QA program was determined at the beginning of the contract. QA audits and surveillances (internal, by the M&O and by OCRWM) have been conducted frequently throughout the performance of the design effort and compliance with QA requirements has been achieved. Corrective actions have been initiated promptly as required.

A Quality Items List (Q-List) has been generated as a part of the Phase 1 design effort. Each hardware configuration item was categorized according to NRC guidelines to designate items required for protection of public health and safety and for occupational radiological exposure control.

PHASE 2 AND PHASE 3 FORECAST

Phase 2 consists of the USNRC certification process for the MPC System, associated testing and the fabrication of a number of prototype hardware items. Phase 3 consists of the fabrication and delivery of the initial complement of approximately 150 Multi-Purpose Canisters and welding equipment. Both Phase 2 and 3 are contract options to be exercised as appropriate by the M&O.

In the Fall of 1995, after OCRWM took action to reduce the activity in the OCRWM program from the FY95 level of $512 M to $400 M, consistent with the continuing resolution, Congress appropriate $400 M for the program and reserved $85 M from use pending enactment of separate legislation. The level of funding available to the program is $315 M, which required further reductions in activity. Phase 1 will be completed, but the work scope has been reduced to eliminate activities in preparation for the second and third phases. Therefore, funding for Phase 2 and 3 is postponed until an authorization bill is passed. The program does not anticipate proceeding to the next phase, consisting of NRC certification and prototype fabrication, or to the third phase of MPC fabrication and deployment beginning in 1998.

SUMMARY

The Westinghouse MPC program is proceeding on schedule for the design of MPC system components and submittal of the SARs on 26 April 1996. Due to funding constraints placed on the program by Congress, OCRWM has chosen not to exercise the Phase 2 MPC certification and prototype fabrication option as originally planned, pending legislative actions that would free up the $85 M of funds set aside late last year.

As Ivan Selin, Chairman of the NRC, has stated in testimony before the Senate Committee on Energy and Natural Resources, the Multi-Purpose Canister represents the only way to link all phases of spent fuel handling (that is storage, transportation and disposal) in one integrated approach. We believe the MPC maximizes the degree of standardization in equipment and infrastructure needed to operate a cost-effective OCRWMS and reduces the degree of specialized equipment, etc., that would be required to handle SNF if the MPC system were not the baseline.

The DOE and the NRC have recognized that utility customers deserve a cost-effective, integrated and standardized system for the billions of dollars they have contributed under the Nuclear Waste Policy Act to support the management of civilian radioactive waste. Spent nuclear fuel needs to be managed safely, cost-effectively and within the best technology available. The MPC represents both a critical first step, and long-term standardized solution to the way this nation should approach the issue of spent nuclear fuel management.

RECYCLING OF LOW-LEVEL RADIOACTIVE ION EXCHANGE RESINS: A CASE STUDY

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ABSTRACT

In the past, ion exchange resins used to clean up radioactive process fluids have been disposed of as radioactive waste. In 1994, Chem-Nuclear began designing a pilot program, RECYGENTM, which targeted the centralized recycling of these radioactive ion exchange resins for reuse at the nuclear utilities from which they were generated. The technology for regenerating organic ion exchange resins is widely used in traditional water treatment but the use of this process presents difficulties when dealing with radioactive materials. The RECYGENTM process was developed with these potential problems in mind. In 1995, the first centralized pilot system for regenerating radioactive ion exchange resins was built and operated.

This paper will detail the operational aspects of RECYGENTM to reduce utility radwaste volumes. An evaluation of the concept, design, and operational aspects of regenerating resin for return to the utility will be discussed. The difficulties of the process will be analyzed such as mixed waste issues, waste by-products, and material handling. Performance data will be presented on the conditions of ion exchange resins received both chemical and physical, and their potential for regeneration, reuse and life expectancy. The utilities interface with the system and the limitations for resin receipt will also be discussed.

The Chem-Nuclear proprietary RECYGENTM process can provide utilities with a alternate method of handling spent radioactive resins. This system provides the utility significant advantages of radwaste volume reduction, cost saving for disposal, cost savings for new resin purchase, and an environmentally beneficial way of handling these spent resins.

INTRODUCTION

A new innovative process to treat radioactively contaminated ion exchange resins has been developed which adds a new aspect to an already existing technology. This process represents a hybrid of recycling and regeneration of radioactively contaminated resin used for water purification at Nuclear Utilities. These resin are processed off-site and returned for reuse at the utility. With ever-increasing disposal prices and uncertainties surrounding disposal site access, utilities continue to seeks ways to reduce the volumes of solid radioactive wastes generated. Organic ion exchange resins are used in a variety of water purification applications throughout nuclear plant systems. Traditionally, these resins which have contacted radioactive water have been used once and sent for disposal. The new process called RECYGENTM was developed to regenerate radioactive ion exchange resins for reuse in nuclear plant applications.Regenerating ion exchange resins has been a common practice for years in non-radioactive applications, but operational costs and waste by-product handling have prevented broad application of such technology to the nuclear industry. Using nuclear industry experience and knowledge in radioactive waste processing coupled with expertise from the commercial water treatment industry, the RECYGEN process was designed and constructed at Chem-Nuclear's consolidation facility in Barnwell, SC.

Using a process of regenerating radioactive contaminated resins for reuse, the utility can save on disposal costs of the resin, purchase costs of new resin, and can reduce solid radwaste volumes.

BACKGROUND

In developing a process to regenerated radioactivity contaminated resin, various aspects of the process and their potential problems had to be studied. The first step was to identify whether radwaste resins could be regenerated given the chemical nature of many radwaste systems which contain high suspended solids, high dissolved solids, detergents, and oils and greases. The regeneration of such resins was not a common practice. To test this potential laboratory bench top regeneration was performed. The results showed a definite ability for radwaste resins to be regenerated.

With this verified, a detailed study needed to be conducted to determine amounts and types of regenerate chemicals needed, the equipment required, and the amounts of waste generated. Most of this data was readily available from conventional water treatment sources. Detailed information was gathered on the types and weight amounts of regenerate chemicals required. Based on this information sodium hydroxide was chosen to regenerate strong base anion resins due to its availability and ease of handling. Hydrochloric acid was chosen to regenerate strong acid resin due to its availability and the high solubility of most chloride salts. The regeneration equipment evaluated was very simple requiring separation/regeneration vessels, a source of chemicals, and a source of demineralized water. In order to enhance the operability efficiency, it was decided to decouple the clean and separation portion of system from the regeneration portion of the system. This would allow the simultaneous operation of clean and separating and regeneration of resins. The study revealed the largest obstacle of regenerating ion exchange resins in a radioactive waste application as being the generation of large volumes of liquid wastes with high dissolved solid content. To address this problem, the study evaluated methods of volume reduction for the waste water problem.

Conventional regeneration systems generate between 50 and 100 gallons of waste water for every cubic foot of resin regenerated. This range is far from acceptable in a radwaste application. A system had to be designed which would reduce this range by at least half and the developed Recygen system would generate between 25 and 40 gallons of waste/cubic foot of resin regenerated. Several technologies were evaluated, including reverse osmosis, thermal concentration, and diffusion dialysis, to reduce the volumes of liquid needed for the process. The answer appeared to be a combination of recycling and thermal concentration.

Based on the study results and the technology evaluation, a radioactive regeneration facility was designed and constructed. The system process was composed of 6 major parts: 1) cleaning, 2) separation, 3) regeneration, 4) rinsing, 5) storage/packaging and 6) neutralization/thermal concentration. Each batch of resin undergoes processing through each phase as illustrated in Fig. 1.

Fig. 1

1. Resin received from the customer is sluiced from the receipt container to the cleaning vessel. During the sluice, a sample of the resin is taken for analysis of physical and chemical characteristics. A reverse flow rinse is introduced into the vessel expanding the resin and rinsing particulate from the resin. The rinse flow is returned through a filter

and back to the rinse water source where it is recycled back thus conserving water generation.
2. The mixed bed resin is then separated. The separation process is a combination of physical separation and chemically enhanced separation. This process produces almost complete separation with little or no mixed interface between the anion and cation resins.
3. The separated resin is then sluiced to the regeneration vessels where the resin is dewatered and rinsed. The resin is then filled with regenerate and undergoes several regeneration steps. The regeneration steps are integrated such that the regeneration chemicals can be utilized on multiple regenerations. This saves on chemical usage and reduces waste generation.
4. Once the resins have been regenerated they are again drained and rinsed. The rinsing at this stage is extensive and the resultant effluent from the resin must meet the set criteria. The first portion of the rinse water is discarded. The vast majority of the rinse is recycled back to the rinse water source. At the source, the rinse water is cleaned and returned back for rinsing. Recycling rinse water eliminates a significant volume of what otherwise would be waste.
5. After rinsing the resin is sluiced to a storage container where the resin is analyzed for physical and chemical characteristics. When the regenerated resin quality is verified, the resin is sluiced into packaged for return and reuse.
6. The final step in the process is the neutralization of regenerate chemical solutions. This neutralization is accomplished by controlled mixing of the hydrochloric acid and sodium hydroxide solutions. Once these solutions have been neutralized, they are pumped to the Thermex thermal concentrator where the liquid is reduced to a salt cake.

PROCESSING

To initiate the system, verify assumptions, and verify system procedures, clean non-radioactive resin processing was begun in August, 1995. After working out the procedural issues, several technical questions had to be addressed through the clean processing. The potential for mixed waste generation using acid and caustic, the regeneration potential, and waste generation volumes had to be solved. The mixed waste potential was easily solved as part of the process where the regenerate acid and base are combined and neutralized. The first process batch verified that the calculated regeneration chemical dosages would achieve the anticipated regeneration efficiency solving the second issue. The primary problem with initial assumptions was identified after the first non-radioactive regeneration batch. The production of waste water was higher than had been anticipated. Initially, all generated rinse water was to be recycled for reuse. It was found that higher chemical concentrations in portions of the waste made it impossible to recover all the waste waters. The minimization of these wastes became the focus of the clean testing.

A detailed chemical characterization was performed on rinse water batches as clean resin was processed. In the process, after resin is regenerated the resin is rinsed to a conductivity specification prior to return for reuse. The rinse water removes residual regenerate chemical to clean the resin. The intent of the rinse characterization was to identify at what point the rinse water could be recovered for reuse. Figure 2 and Fig. 3 reveals the chemical concentrations as the rinse procedure is conducted.

Fig. 2

As the characterization shows, the regenerate chemical concentrations in the rinse drop dramatically during the first part of the rinse. After the first portion of rinsing, the chemical concentrations level off to a reasonably low level. It was concluded that all rinse water, after the first portion, could be recovered for reuse. This conclusion would reduce the rinse water waste generation by 80%. With the rinse water waste volumes under control, the go ahead was given to proceed with radioactive resin processing.

Fig. 3

Low activity resins were acquired to provide an accurate radiation worker dose assessment. A contract was entered with a PWR station who would provide low activity resins for regeneration. These resins would be limited to <50 mrem/hr on contact. The resin provided was mixed bed, composed of 2 parts macroreticular strong acid cation and 1 part gel strong base anion. The goal in processing these resins would be to regenerate back to as close as a new resin specification as possible. The following in Table I are parameters similar to those used in a new resin specification.

Table I

The regenerating of resin to a new resin specification is impractical. As resin is used, physical and chemical degradation occurs. Physical stresses on the resin during use also causes cracking of some resin beads. The trimethyl functional group on the anion resin may slowly decompose leaving weak base sights or no exchange site thus reducing capacity. The polymer chains making up the resins undergo oxidation over time which can also reduce capacity.

For these reasons, used resin must be evaluated to determine the maximum practical extent to which resin may be regenerated with respect to new resin. A representative sample of the resin received is laboratory tested prior to processing through the Recygen system. These tests evaluate the physical condition of the resin and the potential to which the resin can be regenerated. Table II illustrates the analytical results for the initial condition of the low activity resin received.

Table II

The "As Received" resin results confirm the degree of degradation and/or irreversible fouling which has occurred to the resin. The cation resin analysis revealed little change over the new condition of the resin. The anion resin analysis, however, indicated a significant drop in capacity potential. The loss was 10-20% of new anion resin capacity. This could affect the number of regenerations which could be performed if this trend continued. This parameter will be monitored as this resin is received back for a second regeneration.

After analysis, the resin was cleaned, separated, and regenerated per the developed procedures. The regenerate chemical amounts used were in accordance with published literature which would achieve the highest practical regenerated resin capacities. The results of the initial processing is recorded in Table III.

Table III

RESULTS ANALYSIS

The results from the radioactive resin regenerations were very positive. The regenerated resin capacities reached 90% of potential and greater which was the target of the system. The amount of regenerate chemical to achieve these results were very close to the anticipated values. In an effort to investigate the amount of chemical needed to achieve higher

capacity regeneration, a laboratory regeneration was conducted. The capacity increase was plotted against the amount of regenerate chemical used. The results showed that 25% more regenerate chemical was need to achieve a 2% increase to 97% of the potential capacity. This would also require the elimination of recycling the regenerate chemical which would add dramatically to the waste volumes generated. Other alternatives such as varied flow rates and regeneration recycling did prove to increase the capacities without increasing regenerated chemical quantities.The capacities achieved in the process will be more than adequate for reuse in PWR radwaste treatment systems for which the Recygen system was designed. There is some question as to whether the regenerated resin could be placed back into a pure water application such as for secondary system use. The question can only be answered by an in plant test which has not yet been conducted.

The waste produced in these initial runs was slightly higher than anticipated. Approximately 40% more waste generated was contributed by the regeneration of a batch of resin than was anticipated. As processing continues the liquid waste generation will be reduced to the original assumptions. The liquid waste from the process is to be taken to a thermal concentrator where a volume reduction of 7 to 12 will be achieved. The thermal concentration of the waste has only been conducted on a single batch of waste where a volume reduction of 11 was achieved. As regenerating radioactively contaminated resins continues, the regenerated product can only get better. Familiarization with the process and small modifications to the process will provide increases in efficiency and reductions in wastes generated.

CONCLUSIONS

The goal of the CNSI RECYGEN process is to produce quality regenerated resins for reuse at the nuclear facilities. Initial regenerations of radioactively contaminated resins have proven the technology can achieve such quality of resins. These resin will have application in PWR and BWR radwaste waste water treatment systems and possibly in pure water systems. The use of the regeneration system can save the generator of these resins: replacement costs, and disposal costs while minimizing radwaste volumes. The CNSI RECYGEN system is an environmentally sensitive technology which will save the utility money.

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RADWASTE VOLUME MINIMIZATION - DEVELOPMENT AND IMPLEMENTATION OF A PLAN FOR SUCCESS AT TVA NUCLEAR PLANTS

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ABSTRACT

In the early 1980s, Tennessee Valley Authority's Browns Ferry and Sequoyah Nuclear Plants were among the highest low-level radwaste producing plants in the United States. Because of volume restrictions at the Barnwell disposal site, most of TVA's low-level radwaste was transported over 3,200 kilometers (2,000 miles) to the disposal facility at Richland, Washington. In order to reduce the transportation and economic impacts resulting from this large radwaste volume, TVA developed

a plan to minimize the generation of low-level radwaste and reduce the amount sent to disposal facilities. This plan involved benchmarking a number of "good" performers with a team of corporate and site radwaste professionals, determining which methods would work at TVA, and implementing the best volume minimization practices. Implementation involved up-front costs for new products and services and culture changes by the nuclear sites involved in the changes. Results sometimes took years to realize. Almost six years after initiating the plan, the results are staggering. Processing and disposal costs are down dramatically. Performance at TVA plants is now in the first quartile of radwaste generation for their reactor types.

Tennessee Valley Authority has three nuclear plants located in Tennessee and Alabama. Browns Ferry Nuclear Plant is located near Athens, Alabama and has three 1,100-megawatt General Electric BWRs. Currently two of those units are in operation. Sequoyah Nuclear Plant is located near Chattanooga, Tennessee and has two 1100-megawatt Westinghouse ice-condenser PWRs. Both units are currently in operation. Watts Bar Nuclear Plant has one completed 1100-megawatt Westinghouse ice-condenser PWR with no current plans to finish the second unit. The completed unit has a full-power license and is expected to be producing power soon. TVA has another two-unit nuclear plant near Scottsboro, Alabama (Bellefonte) but does not plan to complete these units at this time.TVA shipped its first low-level radwaste in October 1973 from Browns Ferry to Chem-Nuclear at Barnwell, South Carolina. Radwaste burial prices were cheap (about $0.02 per cubic meter or $0.75 per cubic foot), and there was not enough volume to cause concern. As all three units at Browns Ferry were started up and operated, it became apparent that at least one of the units would always be in a refueling outage with the associated large volume of dry active waste (DAW) that is generated. Operation of the three units put a strain on the common radwaste system and on the operating and support personnel in charge of keeping the system operating and the waste shipments traveling between the plant and the Barnwell disposal facility. One challenge was to dispose of the massive amounts of ion-exchange resin produced by the condensate demineralizer system. This powdered resin was difficult to dewater to meet disposal facility requirements. The need to make three or more shipments per week to keep up with the volume resulted in inadequate dewatering of the resin and subsequent violations from NRC and the State of South Carolina.

In 1979, the State of South Carolina restricted the amount of radwaste that TVA and others could send to the Barnwell disposal site to one-half of the previous year's volume. TVA was in the process of starting up the Sequoyah Nuclear Plant with the associated radwaste volumes usually encountered during start-up. Initially, both TVA plants had to store radwaste onsite because the TVA Board of Directors would not allow the nuclear sites to use the Beatty, Nevada or Richland, Washington disposal sites. Without large-volume storage facilities at Browns Ferry, areas in the Turbine Building, the stack, and other outside areas were used for storage. After permission was received to ship radwaste to the Richland site, the storage backlog at the sites was eventually reduced. For some time, over 75 percent of TVA's low-level radwaste was transported over 3,200 kilometers (2,000 miles) to the US Ecology disposal facility at Richland, Washington. Even though TVA then had a place to bury all of the radwaste that was generated at the two sites, we realized that we were

generating much more than we should in comparison to other similar nuclear plants.

In the early to mid 1980s, Tennessee Valley Authority's Browns Ferry and Sequoyah Nuclear Plants were at the top of the list of the highest low-level radwaste producing plants in the United States. Browns Ferry disposed of more than 3,800 cubic meters (134,000 cubic feet) of radwaste in 1983 and 3,600 cubic meters (128,000 cubic feet) in 1984 (see Fig. 1). Both of these numbers were more than 50 percent higher than the INPO median for three BWR units in those years. Sequoyah had both units in operation by 1984 and disposed of more than 1,048 cubic meters (37,000 cubic feet) of radwaste (see Fig. 2). Here again, this was more than 50 percent higher than the median two-unit PWR plant. At that time, radwaste disposal costs were relatively inexpensive and not a major concern for TVA. More emphasis was placed on power generation. High radwaste generation was considered an unfortunate byproduct of electricity production. At this time, Browns Ferry was making five or more large-volume radwaste shipments per week to the disposal facilities.

Fig. 1

Fig. 2

In early 1985 TVA voluntarily shutdown all of its operating units, three at Browns Ferry and two at Sequoyah. This shutdown was unrelated to its radwaste generation problem, although the radwaste problem was a symptom of the bigger management problems faced by TVA in the operation and maintenance of the large nuclear system. Many of the next several years were spent working to resolve the overall problem, and the radwaste problem was not specifically addressed. As can be seen from Figs. 1 and 2, the radwaste volumes at both Browns Ferry and Sequoyah decreased drastically beginning in 1985 as a result of the unit shutdowns.

As part of the overall "get-well" plan for plant recovery following the voluntary shutdowns, TVA began to focus on individual operating and maintenance problems. We realized that in order to have a healthy nuclear program, we would need to get our radwaste generation under control. In 1988, the volumes at Browns Ferry were well under the industry median. This was because Browns Ferry had no units in operation. At Sequoyah, the annual radwaste volume was more than twice the volume of a median PWR plant. As can be seen from Fig. 2, the radwaste volume at Sequoyah remained at or over the PWR median even though no units were in operation. In order for TVA to start-up any of the units and keep them operating properly, something would have to be done to reduce radwaste generation and the associated cost.

From April 1980 until October 1987, Browns Ferry received 24 NRC and state violations for radwaste packaging and shipment problems. Sequoyah received 6 NRC and state violations in radwaste during this same time period. Many of these violations were repeats of previous violations with root causes ranging from personnel errors to inadequate equipment to procedure problems. Several of these violations were as a result of inadequate dewatering of powdered condensate demineralizer ion-exchange resin. Many of these problems could be traced to the plants trying to make too many radwaste shipments too quickly because of the large volume generated. Between 1985 and 1988, a liquid radwaste and resin management plan was put into place in conjunction with Chem-Nuclear Systems to help resolve these problems.

In order to get a handle on run-away radwaste volumes and costs, TVA developed a plan to minimize the generation of low-level radwaste and to

reduce the amount sent to disposal facilities. The first step was to assemble a team of TVA radwaste professionals from the three nuclear sites and the corporate office in Chattanooga. This team would first determine the problems at each of the TVA nuclear plants. Then they would determine "good performers" in the nuclear industry. These plants would be benchmarked to determine if any radwaste minimization activities could also work at TVA facilities. The workable ideas would be tried to determine feasibility. A final report would be written to document the findings of the team and the potential benefit to TVA. It was realized at the start that the fix to TVA's radwaste problems would be costly and take many years. Many of the radwaste minimization problems were ingrained in the system to the point that only a culture change could correct them.

INPO and EPRI were contacted to determine which nuclear plants were considered good performers for benchmarking purposes. EPRI recommended Brunswick Nuclear Station as having a program which had made significant strides in turning around a program which "made electricity as a sideline to creating radwaste". INPO provided a list of BWRs and PWRs which had low radwaste volumes for the year 1988 or with a low three-year average from 1986 to 1988. From that list, Grand Gulf was selected for a benchmarking visit. Riverbend was also selected because of their use of oil decontamination methods. These three sites were visited in early 1990.

In addition to the benchmarking visits, Corporate Radwaste contacted five PWRs and seven BWRs from the INPO "good guy" list by telephone to determine their methods of minimization and volume reduction, staffing levels, methods of processing oily waste, and experience with vendors that contract for radwaste processing services. Based on the onsite visits, telephone calls, and a review of TVA training, site procedures for waste handling, and other plants' experience with offsite vendor processing, the TVA team put together a Volume Minimization and Reduction Plan (VMRP) for the nuclear plant sites in mid 1990. This plan centered around the reduction of DAW waste, since earlier contracts with Chem-Nuclear had started the process of reducing resin volumes through liquid and resin radwaste processing. The following actions were immediately implemented from the Plan:

1. A contract for the burning of contaminated oil was set up with Scientific Ecology Group (SEG).
2. Signs were placed at the entrance to regulated areas regarding the disposal of clean packing materials and other extraneous materials prior to entering the area. Disposal receptacles were placed near the signs.
3. The sites published awareness bulletins concerning the use of incinerable materials and the potential effect of offsite incineration on disposal costs and volumes.
4. Corporate Radwaste obtained a multisite contract for supercompaction which was awarded to SEG. This contract replaced a similar contract with Quadrex. A new multisite contract for DAW incineration was also awarded to SEG.
5. The sites began procurement of incinerable materials and reusable supplies (non-PVC) and would continue replacing non-incinerable and disposable stock items with these items.
6. The sites began more decontamination of floor and wall areas instead of exclusively utilizing disposable plastic coverings. When floor and wall coverings were used, certified incinerable plastic would be used.

7. The amount of contaminated area in the sites was greatly reduced. This, in turn, resulted in less radwaste since work in clean areas did not produce as much contaminated waste.
8. Use of the site hot tool room increased. This resulted in fewer tools to decontaminate and fewer tools and other equipment shipped as radwaste.

In addition, the following recommendations were made in the Plan:

1. The Technical Training staff should modify Level II GET training lesson plans to incorporate identified improvements to the radwaste minimization portion of the course.
2. All initial and retraining GET courses should adopt the REDUCES concept for teaching radwaste minimization at the sites.
3. All upper-tier site minimization procedures should be updated to include the findings of the TVA team.
4. Site management should incorporate radwaste goals into each section manager's goals program for sections that generate radwaste.
5. Site management should appoint a radwaste minimization liaison to aid in the preplanning of major tasks that generate radwaste and routinely monitor work practices for impact to radwaste volumes and cost.
6. Site management should implement methods of penalizing willful violations of minimization practices.
7. The nuclear sites should implement a liquid segregation program which would provide a mechanism for the sampling and proper disposition of oil and other liquids.
8. The sites should continue to provide visual aids for employee education depicting the volume of radwaste shipped, processing and disposal cost, and violations incurred.
9. Sections which generate radwaste at each site should be charged for disposal of their waste directly rather than having the site radwaste organization budget for the entire site. This would put the emphasis for volume minimization directly on the generators of the radwaste.
10. Front-end source term minimization and reduction techniques (such as stellite control, chemical decontamination, constant elevated pH, and improved filtration) should be evaluated by each site and implemented as needed.
11. Work on contaminated equipment at the sites should utilize containments instead of setting up contamination zones.
12. Technological enhancements for equipment or methodologies should continue to be explored by the nuclear sites and corporate staff.
13. The sites should continue with the successful reduction of resin volumes through liquid and resin processing conducted by Chem-Nuclear Systems.

Many of the easy recommendations were implemented quickly and began to show positive results. However, the TVA team quickly realized that TVA plants were in a "Catch-22" situation. The largest problem was the high volume of dry active waste. Until TVA could come to grips with the onsite minimization problem, the obvious answer was either offsite incineration, offsite supercompaction, or a combination of the two. But much of this waste contained PVC (plastic sheeting, bags, and other disposable items) which could not be sent to the incinerator because of resultant hydrochloric acid production in the incinerator's scrubber system. The use of offsite incineration depended upon removing all sources of PVC from the plants and substituting equivalent non-PVC products. This substitution took years to accomplish since many of the warehouses were full of PVC items and workers were use to using the old PVC items. In

some cases, non-PVC substitutes were not available for purchase or were not as reliable as the well-known PVC products. The use of offsite incineration at SEG was delayed for years until certified incinerable items were in complete use at the two sites.

In 1990, the Senior Vice President of Nuclear Power set a Level 1 goal requiring TVA nuclear plants to be at or below the INPO average for radwaste volume. TVA was to maintain this goal every year regardless of plant evolutions. Increased emphasis from site management to reduce radwaste-related costs was a large factor in changing the culture and gaining support from site workers. Other factors included:

1. The use of bulk material permits before large quantities of plastic, wood, or other bulk materials would be allowed into the regulated area.
2. The introduction of reusable or recyclable materials for previously disposed items.
3. Use of onsite and offsite decontamination of tools and equipment . Much of this material was disposed of as contaminated in the past.
4. Improvement in worker education and awareness of radwaste costs and volume impacts.

Onsite segregation of contaminated from noncontaminated waste was tried for some time at Browns Ferry. Plant laborers and RADCON technicians were used to sort through yellow bag trash to determine which items were contaminated. Noncontaminated items were disposed of with the green bag trash. Much of the pay-back from this practice came from the location and reuse of tools and equipment found in the radwaste. This practice was discontinued at Browns Ferry because it was labor intensive and costly.

One of the biggest impacts on DAW generation was the reduction in the amount of contaminated area in the plant. At Browns Ferry, the percentage of contaminated area in the plant decreased from a high of 14.7 percent in 1989 to the present value of 0.75 percent. At Sequoyah, contaminated area has been reduced from a high of 13.7 percent in 1990 to the present value under 3 percent (at the end of the unit 1 refueling outage). This means that more of the plant can be accessed and worked performed in street clothes without the volume impacts associated with contaminated zones. It also means that equipment leaks that cause many of the zones have been reduced as well as the associated introduction of contaminated water that must be processed by plant and vendor systems. Reduction of contaminated plant areas is one of the first steps to getting control of the radwaste problem at a plant.

Although the use of offsite compaction and incineration systems is very important, the real indicator of improvement is a reduction in the amount of radwaste generated. At Browns Ferry. total DAW generation has reduced from 2,400 cubic meters (85,000 cubic feet) per year in 1993 to less than 1,700 cubic meters (60,000 cubic feet) at the present time. At Sequoyah, the amount of DAW generated has decreased from 2,800 cubic meters (100,000 cubic feet) per year in 1991 to 566 cubic meters (20,000 cubic feet) at the present time. During this time, unit refueling outages at these sites have produced less DAW than the previous outages. This indicates that we are getting better at doing the same work while making less waste each time.

Dealing with the ion-exchange resin generation problem at the plants was somewhat easier than the DAW problem. At Sequoyah, annual resin generation volume has now dropped from a high of 227 cubic meters (8,000 cubic feet) in 1985 to the present value of less than 28 cubic meters (1,000 cubic feet). This decrease can be attributed to better plant

chemistry, the efforts of an excellent site radwaste staff, and a dedicated vendor crew utilizing Chem-Nuclear's ALPS water processing unit. Payment for water processing at the site is based on the number of gallons of water processed rather than the number of hours worked or the volume of resin generated. This provides an incentive for Chem-Nuclear to optimize the throughput of the resin to produce the quality of water required by TVA. It has worked to the advantage of both Chem-Nuclear and TVA to minimize the resin volume.

At Browns Ferry, the amount of ion-exchange resin generated is below the best quartile value for a two-unit BWR. Most of this success is due to changes in the condensate demineralizer precoat system and new technology in the use of filtration at the plant. Credit also goes to the use of the Chem-Nuclear RDS-1000 resin drying system run by a Chem-Nuclear operator at the site. This system is responsible for reducing the amount of resin volume sent for disposal through its drying system and for eliminating the potential for excess water in Browns Ferry's resin shipments to Barnwell. Browns Ferry received three violations and fines for having excess water in its dewatered resin from 1980 to 1987. There have been no violations since use of the RDS-1000 began in 1988.

In July 1993, TVA and Chem-Nuclear Systems entered into a long-term Radwaste Services Partnership to provide a full scope of radwaste services and equipment to TVA nuclear plants and other facilities. The goals of this partnership were to keep radwaste volumes and costs as low as possible while providing quality processing and disposal services to TVA plants. In the first 2 years of the Partnership, millions of dollars of processing and packaging savings have been realized by TVA. As a result, Chem-Nuclear recently received the 1995 TVA Suppliers Excellence Performance Award for their innovations and cost reduction achievements as a supplier and Partner to TVA.

Almost six years after initiating the Volume Minimization and Reduction Plan, the results are staggering. After receiving 30 NRC and state violations from 1980 to 1987, neither Browns Ferry or Sequoyah has received a radwaste violation since October 1987. Corporate and site radwaste goals are now used to determine employee and management incentive bonuses. Processing and disposal costs are down dramatically. TVA plants are now in the first quartile of radwaste generation for their reactor types. Sequoyah has been at or within the best quartile since 1992. This turnaround in the radwaste management problems at TVA nuclear plants required much work by the site and corporate staffs and a commitment at all levels to make it succeed. However, the results have been worth the hard work, and the program is healthier than it has been in years.

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RETRIEVAL, PROCESSING AND PACKAGING OF RADIOACTIVE WASTES

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ABSTRACT

Within the UK Electricity Supply Industry in England and Wales there are six operating Magnox and five Advanced Gas Cooled Reactor (AGR) stations,

a PWR at Sizewell B and two Magnox stations undergoing decommissioning. The Sizewell B PWR has recently started commercial operation, and is equipped with a plant for the routine encapsulation of Intermediate Level Wastes (ILW). An extensive program of research and development has been carried out over a number of years, augmented by on-going sampling and analysis, in order to formulate a cement based encapsulating medium for all wastes.

Over the 30 year period of Magnox reactor operation a number waste materials have been produced, and those classified as ILW are retained in storage facilities at the stations. Some of the materials are no longer commercially available, and a suitable simulant must be identified before any formulation development work can be carried out. In addition, the design of the early Magnox reactors did not permit a high degree of waste segregation, and consequently some waste storage facilities contain a number of different materials which could have undergone physical/chemical change. The situation at the AGR stations is characterized by lower waste arisings and improved segregation. In addition, most of the materials which make-up the waste are still commercially available which simplifies the selection of a simulant for development work.

Successful formulation is measured against a number of criteria, all directed at producing a satisfactory wasteform which will have acceptable physical and chemical properties in the post-closure repository environment. Properties measured include dimensional stability, weight change, tensile strength, elastic modulus, pore water chemistry, the effect of radiation, exposure to water and freeze/thaw conditions. This is complimented by an extensive program over a number of years to monitor the potential for corrosion of the waste container. It is also essential to take account at the formulation stage of aspects such as heat evolution, rheology of the cement mix, retrieval of the waste and application to an encapsulation plant.

Materials which would be expected to have an affinity with a cement based medium have in general represented no particular problem with high waste loadings readily achieved. Other materials, such as phenol formaldehyde based resins, have shown a propensity for expansive phase formation under irradiation resulting in physical failure of the wasteform. Some ion exchange resin wastes, in a formulation with a high sodium hydroxide content to offset the inhibition to setting caused by the presence of boron, exhibited failure under water immersion. A number of wasteforms have appeared to fail under radiation exposure, although modelling suggests that the mechanism is a combination of radiolytic gas generation and the limiting diffusion rate through the cement matrix. These issues have now been resolved and the emphasis has moved to the development of a methodology for the encapsulation of the retrieved wastes, which in a number of cases will consist of an homogenized mixture of materials. There is a significant generic element to the methodology database, particularly where the technology has been underwritten by more fundamental studies on the underlying physics and chemistry. Experience has been obtained with techniques to improve the strength of the wasteform, and the potential effect on the wasteform of a variety of low concentration materials such as oils, de-greasing agents and detergents, decontaminants and superplasticisers. This body of information, which includes some novel formulations, therefore represents a valuable contribution to the requirements of utilities with all types of waste.

The overall conclusion is that sufficient experience has now been accumulated to enable real retrieved wastes to be encapsulated in a cement based media which is highly compatible with the required repository environment. This can usually be achieved at a high, and therefore economic, waste loading. The formulations have also shown themselves amenable to modification in order to accommodate a particular feature of a waste or property required of the wasteform.

BACKGROUND

In the privatization of the UK Electricity Supply Industry in 1990 the nuclear component in England and Wales was retained in the public sector under the name Nuclear Electric plc (NE). NE assumed responsibility for the operation of the Magnox stations (two of which are now undergoing decommissioning), five Advanced Gas Cooled Reactor (AGR) stations and completion of the Sizewell B PWR. Since 1990 NE has continued to develop a comprehensive strategy for the management of radioactive waste arisings from the operation and decommissioning of nuclear power stations (1). The nuclear component of the UK Electricity Supply Industry is now being further privatized, with the Magnox reactors remaining in the public sector.

This paper is a logical follow up to the paper given to the WM'95 Conference (2), in which a broad overview was given covering waste types, work on encapsulation techniques, the intentions with respect to encapsulation plant and some of the novel technologies which were being developed. The intention here is to update the position for Magnox, AGR and PWR intermediate level wastes (ILW)a, and focus on some of the more intractable areas where satisfactory encapsulation formulations have now been developed. It should be noted that low level waste (LLW) is routinely disposed of to a shallow land burial site in the UK operated by British Nuclear Fuels plc.

ENCAPSULATION OF SOLID ILW IN CEMENT FORMULATIONS

At present operational radioactive ILW arisings at power stations are held in special accumulation facilities pending eventual retrieval and disposal. It is both national and company policy that disposal will be by deep land burial in the proposed NIREX Deep Waste Repository (DWR). To ensure the safe handling and transport of the wastes between retrieval and emplacement in the repository, it is intended that they will be encapsulated in a suitable matrix contained within a NIREX approved container. The encapsulation matrix will be cement based, as this material has a number of advantages, although the option to use an organic polymer is still retained and may be used as a one-off to encapsulate some ion exchange resins at Trawsfynydd power station which is undergoing decommissioning. Although cement is a cheap, well tried and tested material such that its handling and general mechanical and chemical properties are well known with a wealth of practical experience on which to draw, there are other factors which need to be borne in mind concerning how cement based formulations are predicted to behave within the repository.

In developing this cement based approach to the encapsulation of ILW the following issues have been continuously reviewed, as it is important to the long term integrity of the product that the optimum cement formulation is chosen for each waste, whether a single material or a mixture.

- availability of novel cement based formulations to offset a particular problem.

modification of the cement set process and the wasteform properties by waste/cement interactions.

achievement of a particular cemented wasteform characteristic with chemicals and other additives.

formulation development compatible with a full scale operational and active encapsulation plant.

Nuclear Electric's Intermediate Level Wastes

The ILW which NE will have to encapsulate can be summarized as follows, noting that there are also minor amounts of a number of other materials present.

Magnox Stations:

Magnox fuel element debris

Magnox sludges

Active Effluent Treatment Plant (AETP) sludges

Oily wastes (especially oily sludges)

Inorganic ion exchange materials

Organic ion exchange resins (2 main types)

AGR Stations

Borated ion exchange resins

Active Effluent Treatment Plant (AETP) sludges

Filter pre-coat materials

PWR Stations:

Boric acid concentrates

Borated organic ion exchange resins (similar to those at AGR stations)

Filter elements

The amounts of ILW currently held at NE's stations are comparatively modest and are typically a few tens of m3 for any individual waste (except at Trawsfynydd where the holdings of spent ion exchange resin are a few hundreds of m3) (3).

REQUIREMENTS OF THE CEMENTATION PROCESS

Retrieval

It is assumed that all ion exchange resin and sludge type wastes held in storage will be retrieved by homogenizing the contents of a tank and hydraulically transporting the mixture to a secondary vessel or the encapsulation plant. This is particularly the case with the accumulation facilities on the earlier Magnox stations which may contain a number of waste types. In such instances the cement formulation must be able to encapsulate the mixture of wastes and, allowing for the possibility that the wastes may not be homogeneously mixed when retrieved, the formulation should ideally be able to accept the extreme of 100% of any one component. This is potentially quite a severe constraint, but as yet has not proved insurmountable, and in most cases represents a very unlikely situation. There is the additional point concerning the water content of the retrieved waste and whether de-watering is required before encapsulation.

Mixing

In the application of a cement encapsulation process, where waste and other ingredients such as Ordinary Portland Cement (OPC) and Blast Furnace Slag (BFS) are dispensed into the container and mechanically mixed with water to form an homogeneous mixture, heat is evolved which will assist the hydration process and the mixture will become more viscous over time until the initial set occurs. It is usual to measure the torque required by the motor turning the mixing paddle, and for two

speeds to be used for the mixing process, a lower speed while the ingredients are being dispensed into the container to minimize splashing, and a faster speed to ensure homogeneity. The composition of the mixture, and the rate at which initial set occurs must therefore be such that an homogeneous mixture is formed before the viscosity starts to increase significantly. This represents another area of potential constraint on the cement formulation.

Heat Evolution

In the encapsulation of radioactive waste in cement based formulations, some account must be taken of the heat evolved. In terms of the specific heat of hydration the reaction of the aluminate phase is most exothermic, followed by alite, the ferrite phase and belite. The bulk of the heat, however, arises from the alite C3S component, because it is the major constituent. One simple way of reducing the heat generated is to dilute the C3S component with other cementitious materials which contain small percentages of C3S. Two such materials are pulverized fuel ash (PFA) and BFS and both have found use in formulations for NE's wastes. The temperatures developed in some typical laboratory experiments for various ratios of OPC to BFS are shown in Fig. 1. It is worth noting how the heat is released earlier in the OPC-rich mixes than in the BFS-rich ones, due to acceleration of the rate of the hydration reaction by heat. For practical purposes it is found necessary to have a least 7:3 BFS/OPC and in general 9:1 BFS/OPC is specified for use with NE's wet slurry wastes. One consequence of the use of high slag or BFS cements is that the range of available water/cement ratios is rather less than would be the case with pure OPC. This is illustrated in Fig. 2 where the range of acceptability in terms of the ability to mix and the presence of bleed water or failure to set narrows increasingly as the %BFS increases. This also restricts the range of the water/cement ratio.

Fig. 1

Fig. 2

NE's mobile encapsulation plant will utilize containers ("liners") with a capacity of 2.6 m3. Such a large volume of cement can lead to problems with excessive heat generation as noted above and to ensure that this does not happen the cement formulation must be selected carefully.

Application to an Encapsulation Plant

A cement formulation suitable for a particular waste will have been developed on a small scale initially and then taken to the 10-20 litre scale for the product evaluation work and the production of specimens for testing. If a particular problem concerning handling or mixing is envisaged some testing may be carried out at full scale.

Eventually the cement formulation, in terms of weight of cement and BFS and volumes of waste slurry, mix water and other ingredients, will be programmed into the control system of an encapsulation plant for automatic dispensing during a production run on active waste. Is essential to know the expected level of plant error bands, the repeatability of the system in terms of valve opening times and pump delivery quantities per stroke or revolution. Also the volumes of all vessels used to store or dispense materials, the volume of any spaces where "hold-up" can occur and the implications of all flushing operations. In this way the parameters of the plant can be taken into account and the formulation adjusted if necessary to ensure that the containers do not over or under fill and that the wasteform is satisfactory when set. This is another area of potential constraint on

the cement formulation, as it must be flexible enough to accommodate such modifications.

The plant throughput must also be such that containers are not moved until after the initial set has occurred, in order to avoid the possibility of the formation of cracks or fissures. Although the wasteform is not fully hydrated by this time it is sufficiently free of liquid water that the crack or fissure will not self-heal, and the wasteform is therefore cracked and in a failed condition from the start.

Due to the quantities of waste involved it is not considered cost effective to construct and retro-fit a waste encapsulation plant at each existing station and it is intended that a mobile encapsulation unit will be available. Sizewell B has a dedicated encapsulation plant and this is the intention with any future stations. In order to achieve the required throughput it is important that the cement hardens sufficiently quickly, as storage capacity for drums set aside to harden is limited and a maximum set time of 16-24 hours is envisaged. There is also a requirement that there be no standing or bleed water on the product after set has been achieved, and this places an upper limit on the acceptable water/cement ratio. The process will be in-drum mixing with a sacrificial paddle, and as the mixing torque requirement is a function of the viscosity of the mixture, it imposes an additional constraint which sets the lower limit to the water/cement ratio.

REQUIREMENTS OF THE CEMENT FORMULATION DEVELOPMENT

Cement Chemistry

The chemistry of cement both during set, curing and subsequent longer term hydration is complex but fairly well understood in terms of the main physical phases and the overall micro-structure. The important issue in the cementation of a variety of radioactive wastes is the potential for the formation of unstable cement monoliths. This can range from spalling, which may be at a tolerable level, to fracture and crumbling of the monolith which is not acceptable. This in turn can be due to physical/chemical mechanisms such as radiolytic gas generation or the formation of an expansive phase.

Sampling/Analysis and Simulants

There is some physical and chemical data available on most of the wastes, however this is not always comprehensive enough and there is an on-going program of sampling and analysis. The identity of the waste can often be obtained from station records, although one difficulty not apparent in the list of waste types given earlier is that several different materials may occur together in the same waste accumulation facility.

Development work is not usually carried out on active waste and it is necessary to identify a number of simulants. If the waste material is still available it would be used in the "as received" condition, unless there was specific knowledge of physical or chemical degradation in storage, in which case this would be simulated. An example is particle size reduction of a material during use. If the material is not available commercially, as with granular Lewatit DN ion exchange resin, then either remaining sources must be located or a simulant material identified. The simulant must be chosen carefully in order not have any significant deviations from the expected behavior of the actual waste during cement encapsulation and the product evaluation testing.

A category of waste for which simulation is the only way to proceed is sludge from the Active Effluent Treatment Plant (AETP). Based upon the origin of the AETP input flows, including filter backwash, dirty drains,

washrooms and laundry, up to twenty materials are required to simulate a sample of AETP sludge. One of these is oil which acts as a binder for the sludge.

Wasteform Properties

Adequate mechanical strength: great strength is not required but it must be adequate for handling and transport, which means that the container/wasteform must pass the appropriate drop tests.

Dimensional stability: some cements expand as they set, imposing stresses on the container which may effect its integrity. Excessive contraction is also undesirable as this potentially allows water access to the container which could lead to enhanced corrosion.

Monolithic block: the wasteform must set to give a monolithic block free of cracks or loose material.

Minimal inside-out corrosion of drum: whilst the cement remains in contact with the drum, corrosion is likely to be minimal (the cement maintains a high pH environment) unless excessive quantities of corrosive anions (such as chloride) are present in the waste. Outside-in corrosion is a function of the storage conditions and has no bearing on the wasteform properties.

Fire resistance: must be able to withstand a standard fire test.

Low leachability: if the wasteform comes into contact with water, for instance, in an accident whilst being transported to the repository, or ultimately, in the repository itself, radionuclides must have a low leach rate. This condition is usually easily achieved if the wasteform maintains its integrity (cracking/fissuring/crumbling etc. increases the surface area, and may release active particulate).

Minimal gas evolution: gas may be evolved either through radiolysis of the water in the cement or due to corrosion of metallic materials such as Magnox. Gas evolution under such circumstances at a realistic repository dose rate is unlikely to cause cracking of the cement (due to the build-up of gas pressure in the micropores of the cement). However, since one of gases evolved is hydrogen, this may represent a safety hazard in addition to leading to drum pressurization. Fortunately at the doses likely to be received from NE's wastes these problems are of little importance, and there is always the design option of a vented drum.

Resistance to radiation: the cemented wasteform will be exposed to a predicted dose rate of about 20 Gy h-1 in the repository from self-irradiation and irradiation from adjacent wastes. The whole life repository dose is estimated to be 9 MGy. In order to achieve the whole life dose, irradiation tests have to be accelerated and as will be explained this can lead to experimental problems.

SUPPORTING DEVELOPMENT AREAS

Long Term Monitoring of Cemented Wastes

A large number of samples are included in this long term monitoring program, and dimensional stability and other physical properties are updated annually for the following wasteforms:

Fine and coarse Magnox sludge
Bead Lewatit DN ion exchange resin
Ammoniated IR-120 ion exchange resin
Decalso Y
Granular carbon types 203C and 207C
Dicalite Speedplus
Cemented AW500 inorganic ion exchange material
Magnox splitters

Borated wastes such as ion exchange resins
PWR primary ion exchange resin
AGR silica gel resin
PWR wastes such as boric acid concentrate

Corrosion of Waste Containers

A number of 50 litre containers have been instrumented to monitor the in-drum environment in terms of the corrosion potential, oxidation potential, galvanic current, relative humidity, temperature and corrosion rate. The containers are stainless steel or carbon steel and are either vented or sealed. Each container is filled with a simulated waste in the appropriate cement formulation, followed by a capping grout and a lid. Following a period of monitoring the containers are sectioned, the instruments recovered and the internal surface of the container subjected to detailed examination. Table I lists the containers monitored to date. The sealed containers, with the exception of number 1 (boric acid evaporator concentrate simulant in a stainless steel container), are continuing to show anaerobic potentials. The move to aerobic potentials in the stainless steel container (number 1), is associated with micro-cracking of the wasteform allowing air penetration. The destructive examination of container number 4 (boric acid evaporator concentrate simulant in a vented carbon steel container) confirmed that conditions within the drum were reducing and not corrosive.

Indicated corrosion rates are between 0.02 and 10 mm year-1. Times to penetration assuming uniform corrosion from inside only are 280 years for the mild steel container and over 15,000 years for the stainless steel container. However localized corrosion will reduce this time considerably. Corrosion rate measurements will continue on the remaining six containers in order to underwrite the envisaged minimum lifetime of 50 years.

Table I

Magnox/graphite/alloy Combinations

This work was undertaken to identify any particular combinations of materials that could give rise to enhanced rates of corrosion when electrically/chemically coupled. The materials used to set-up galvanic couples with Magnox were graphite, cast iron, zirconium, 316 stainless steel and Nimonic springs as shown in Table II, all with appropriate degreasing and surface treatment. In addition two 50 litre stainless steel containers were instrumented to monitor the galvanic corrosion current between Magnox and graphite, corrosion and oxidation potential, relative humidity of the air in the container void, temperature within the wasteform and on the container surface, and the corrosion rate of a single Magnox electrode. In all cases the containers were filled with 3:1 BFS/OPC with a w/c ratio of 0.39.

Table II

It was concluded that galvanic coupling of Magnox to graphite is unlikely to result in significantly higher rates of hydrogen gas generation, the long term corrosion rate for Magnox/graphite couples is <0.01 m year-1 and that the oxidation potential of the wasteform indicated that conditions in the cement were oxidizing throughout the test duration. Physical examination showed that exposure of Magnox to a cement environment resulted in a protective surface film of Mg(OH)2 and the inside walls of the stainless steel container showed no signs of corrosion after greater than 6000 hours exposure to a 3:1 BFS/OPC (w/c =

0.39) grout. Over the test period of up to 250 days, mixed graphite and Magnox in a cemented matrix showed no propensity for enhanced corrosion.

Assessment of Superplasticisers

The literature suggests that the correct use of these materials will give increased fluidity without any adverse affects on the long term wasteform properties. Only sulphonated naphthalene formaldehyde (SNF) type superplasticisers are assessed as possessing negligible complexing ability and are therefore preferred as they would not enhance radionuclide mobility in the repository post-closure situation. Experiment has shown that it is essential to use these materials within the manufacturers recommended concentrations, otherwise poor process and product properties result. The conclusion of the tests was that the use of SNF type superplasticisers within prescribed limits, will not result in any deleterious long-term wasteform properties.

Assessment of Detergents, Decontaminants and Other Agents

Power stations use a number of materials for cleaning and decontamination processes, some of which will enter radioactive waste streams and could result in low concentrations in the retrieved waste for encapsulation. The objective of this and other studies is to prove that credible concentrations of such materials in typical waste streams will not adversely effect the encapsulation process or the long term stability of the wasteform.

A typical decontaminant liquor would contain sodium citrate, EDTA and active detergent in a sodium sulphate carrier. The detergent would typically consist of sodium alkyl aryl sulphonate, ethoxylate non-ionic alcohol, sulphate, chloride and other salts in a water based carrier. Nitric acid decontamination liquor neutralized with sodium hydroxide was also included, in addition to other decontamination solutions containing potassium permanganate, nitric, citric and oxalic acid. The composition of some AETP simulant sludges includes commercially available detergents and decontaminants. Some detergents are based upon mixtures of chemicals such as nonyl phenol ethoxylate, sodium salts of EDTA, sodium tripolyphosphate, sodium carboxy methyl cellulose and sodium sulphate, while others are alkaline and based upon biologically soft surface active agents and sequestering agents. Other commercially available materials consist of sodium metasilicate, sodium carbonate, trisodium phosphate, low foaming surfactants, anti-foaming agents and a sequestrant. Specific anti-foaming agents have been examined separately. The overall conclusion from this comprehensive body of information is that there will not be a problem with the encapsulation in cement of radioactive ILW containing such materials at anticipated concentrations. However, long term degradation of organic species may need further consideration. The high pH of the cement environment and the low concentration of these organic materials (and their degradation products) probably represents a satisfactory situation with respect to the repository post-closure safety case.

This work was directed at commercially available detergents/decontaminants which will be used at nuclear power stations normally in a form diluted with water to 10 vol% although more concentrated solutions may have particular applications. The conclusion from the small scale testing and product evaluation studies is that these materials can be encapsulated satisfactorily in a 9:1 BFS/OPC cement blend with up to 100 volume per cent of the water volume replaced by the detergent/decontaminant, and with no adverse effect on the waste product

properties up to 90 days. Properties of these cemented wasteforms will be monitored long term.

WASTE-CEMENT INTERACTIONS

Having reviewed briefly the major constraints on the encapsulation process and factors influencing the selection of cement formulations, there remains the subject of waste-cement interactions. Some wastes can be incorporated into cement at acceptable loadings without difficulty, but others require special pre-treatment or the addition of other chemicals to modify the chemistry during or after setting. This will be illustrated by reference to two specific waste streams, boric acid and Lewatit DN ion exchange resin. Boric acid concentrates will arise at the Sizewell B PWR, and an understanding of how boric acid interacts with cement is essential to the treatment of this waste and the successful incorporation of borated resins (which arise at both AGR and PWR stations) into cement. Lewatit DN is a phenol formaldehyde type ion exchange resin which has been used on Magnox stations, and which has proved difficult to successfully encapsulate in a cement based formulation due to its unique properties.

Encapsulation of Boric Acid and Borated Resins

The well known effect of boron (and other inorganic salts) in retarding the process of cement set is illustrated in Figs. 3 and 4, where quite modest amounts have a considerable effect on the time to set as measured by the exotherm and the cement strength as measured by the ultrasonic pulse velocity. A typical PWR boric acid concentrate will be 12% by weight (though concentrations up to 24% may be produced) and with such a concentration cement does not, for all practical purposes, set. The borated resins will be of the divinylbenzene (DVB) type and will arise at all power stations from pond water ion exchange plant, although the most active will be from the PWR CVCS system. The formulation must be able to encapsulate all extremes from 100% anionic to 100% cationic resin.

Fig. 3

Fig. 4

The mechanism of delayed setting is the formation of an amorphous calcium borate phase of composition CaO.B2O3.6H2O around the grains of cement, hindering normal access of the porewater. This effect may be overcome by preventing the formation of the solid calcium borate phase by either keeping the boron in the aqueous phase or removing the boron into a solid phase other than calcium borate.

Initial work was directed at keeping the boron in the aqueous phase, and study of the [Ca(BO2)2.2H2O-Ca(OH)2-H2O] system indicates that a high (OH)- concentration increases the solubility of Ca(BO2)2.2H2O. There is also an increased rate of hydration in the presence of NaOH as indicated by the evolved heat. Therefore the use of sodium hydroxide additive would keep the B(OH)-4 monoborate ion in solution and replace the calcium borate with calcium hydroxide through which the porewater could diffuse. For the PWR boric acid concentrate, experiments have shown that in 9:1 BFS/OPC mixtures the optimum NaOH:boric acid ratio is 2.5:1 to achieve set in 16 hours, and this simple approach works well in overcoming the retarding effects of boric acid. This type of wasteform was satisfactory in all respects except resistance to irradiation.

The problem with radiation testing is that in order to achieve the estimated whole life repository dose of 9 MGy it is necessary to use accelerated irradiation testing, and this means using a dose rate of 2,000 Gy h-1 or greater in order to complete a total dose of 9 MGy in a

suitable time period such as 3 or 6 months. Radiolytically generated gas in the pore structure of the set product was unable to escape at the required rate and caused pressurization of the internals leading to fracture. The gas generation rate is a function of the dose rate (about 2x104 Gy h-1) and this is much greater than the predicted dose rate in the repository (20 Gy h-1). Subsequent work established that by limiting the boric acid loading to 12 wt%, allowing the product to cure for over 100 days and reducing the dose rate for the accelerated irradiation tests, a satisfactory resistance to irradiation up to the whole life repository dose of 9 MGy was achieved. Therefore, the product obtained by incorporating 12% boric acid in 9:1 BFS/OPC is able to meet all targets.
With borated resins, the first formulations contained up to 21.5 wt% borated IRN 150L resin in a 9:1 BFS/OPC (w/c = 0.3) mixture with 47 wt% NaOH (approximately 13 Molar). The wasteform was acceptable in all respects except underwater stability, where degradation of the wasteform was caused by resin bead swelling due to the presence of large NaOH concentration gradients. Subsequent work demonstrated that with a reduced waste loading of 10 wt% in 9:1 BFS/OPC (w/c = 0.33), the NaOH concentration could be reduced to 5.5 wt% (approximately 2.1 Molar mix water) and the wasteform was stable to underwater immersion.
It was therefore necessary to develop an alternative system which could return to waste loadings greater than 10 wt% while still meeting all the requirements for the properties of the wasteform. Further development work identified the mix consisting of 20 wt% borated IRN-420 or IR-120(Na+) resin in 9:1 BFS/OPC with 6 wt% (of the cement powder) of high alumina cement (HAC) (w/c = 0.37 and 0.50) using 1.5 Molar NaOH mix water. The role of the HAC, as for the NaOH, is to prevent retardation of the OPC hydration by boron. The calcium aluminate phases from the HAC react with B(OH)4- to form a solid boron containing phase which does not hinder access of porewater to the cement particles.
However, this formulation showed unacceptably poor compressive strength development at 2 days for 25 wt% borated IRA-420 in a cement formulation of 9:1 BFS/OPC + 6 wt% high alumina cement (HAC) + 1.5 Molar NaOH mix water over the range of water/cement ratios. This would give problems in moving the cemented waste without causing fracture. Although beyond 2 days and up to 90 days strength development, dimensional stability and underwater stability were acceptable, this formulation was unacceptable. The maximum waste loading for 100% borated resin is therefore limited to 20 wt% until the level within the range 20 to 25 wt% is located in terms of the 2 day compressive strength. A 25 wt% resin loading is suitable for IR-120 (Na+), primary circuit resin and secondary circuit resins, in terms of mixing characteristics, set, 24 hour bleed, compressive strength, dimensional stability and underwater stability to 90 days.
Irradiation tests used 20 wt% IRA-420 or IR-120 (Na+) in a 9:1 BFS/OPC mix with 6 wt% HAC and 1.5 Molar NaOH mix water, and as with the boric acid concentrate wasteforms, samples split into several pieces after a dose rate of 10,000 Gy hr-1 to a total dose of 6 MGy. The mode of failure (several large pieces) was not the same as the observed failure mode for cemented Lewatit DN under irradiation (which was more of a disintegration into a coarse powder), and there was no evidence of the expansive phase formation for the borated resins, noting that the cationic resins contain a sulfonic acid group. It was therefore suspected that the wasteform failure mechanism was, as with the boric acid samples, the radiolytic generation of gas within the wasteform at a rate too great for normal

dissipation by diffusion through the pore structure. In fact some of the pore structure may be sealed. The resulting pressurization of the wasteform both in local near-surface areas and deep within the structure caused the observed spalling and complete fracture. Also, if this mechanism is correct, the effect should be less and eventually not occur at all at lower dose rates.

Using this approach, samples of 9:1 BFS/OPC (w/c=0.37) with 1.5 Molar NaOH mix water cured for 120 days prior to irradiation at 2,000 Gy h-1 were found to be damaged at between 3 and 6 MGy, and samples of 9:1 BFS/OPC (w/c=0.37) + 6 wt% HAC with 1.5 Molar NaOH mix water cured for 120 days prior to irradiation at 2000 Gy h-1 were found to be undamaged after 9 MGy. However, samples of 20 wt% IRA-420 resin in 9:1 BFS/OPC (w/c=0.37) + 6 wt% HAC with 1.5 Molar NaOH mix water cured for 120 days prior to irradiation at 2000 Gy h-1 were found to be undamaged at 2 MGy and severely damaged with surface chipping and sample splitting after 3 MGy. Samples of 20 wt% IRA-420 resin in 9:1 BFS/OPC (w/c=0.37) + 6 wt% HAC with 1.5 Molar NaOH mix water cured for greater than 120 days prior to irradiation at 200 Gy h-1 were found to be undamaged after 0.6 MGy and the irradiation will continue.

Encapsulation of Lewatit DN

Lewatit DN is a phenol formaldehyde resin used at Magnox stations for caesium removal from pond water and liquid effluents. It was one of the first waste types to be investigated and these early studies were concerned with its incorporation into pure OPC. At waste loadings of about 30% by weight (a typical target loading) the wasteform performance was satisfactory at first but over a period of several weeks the wasteform gradually expanded, and eventually became little more than a pile of crumbly lumps. A solution to this problem was found by loading the resin beads with calcium ions prior to encapsulation. However, although this worked well in the laboratory it proved difficult to make into a process because the rate at which the resin beads take up calcium ions is very slow. The move to high slag (BFS) cements led to alternative formulations which appeared to incorporate Lewatit DN without problems, although care is required not to use excessively high loadings and a relatively low water/cement ratio is essential. Such formulations have been tested over periods of up to a year and appear to give satisfactory products with respect to the expansive phase failure mechanism mentioned above.

In the expansive phase failure mechanism in pure OPC it is proposed that oxalic acid or another organic acid resulting from breakdown of the resin, leaches from the beads and interferes with the cement chemistry in a number of ways:

Enhanced precipitation of calcium hydroxide (Portlandite) in the vicinity of the resin beads.

Formation of mal-formed Portlandite crystals (massive plates rather than needle-like crystals) which are responsible for the expansive failure reaction.

Formation of crystals of other calcium salts including calcium oxalate which may be the cause of the mal-formation of the Portlandite.

Substitution of BFS for OPC alleviates this problem, and the key lies in the nature of the hydration reactions and the relative amounts of Portlandite formed. The initial products of OPC hydration are Portlandite and CaO-SiO2-H2O or C-S-H gel (which ultimately gives the product its

strength), whereas with BFS some Portlandite is consumed to generate more C-S-H gel.

Although the BFS/OPC based formulation was extensively tested with Lewatit DN and normal product evaluation data on dimensional stability, strength and other parameters obtained, there remained the question of resistance to radiation. Due to the presence of a sulfonic acid group, phenol formaldehyde type resins can give rise to the formation of expansive phases when encapsulated in some cements and exposed to g-irradiation. The mechanism is radiolytic scission of the sulfonic acid group releasing $(SO4)2-$ which then reacts with $Al2O3$ to form an expansive phase called ettringite with a solid volume increase by a factor of two. This can be overcome by reducing the $Al2O3$ content of the cement.

These difficulties with Lewatit DN resin have been addressed by using a developed formulation based upon Condensed Silica Fume (CSF) and Sulphate Resisting Portland Cement (SRPC). The $Ca(OH)2$ is removed by pozzolanic reaction with the $SiO2$ from the CSF, and both CSF and SRPC are low in $Al2O3$ thereby reducing the formation of ettringite with $(SO4)2-$ released after irradiation. However, expansive phase formation was observed when the SRPC was greater than 50 weight percent.

A 1:1 CSF/SRPC (w/c = 1) control and a 20 wt% bead Lewatit DN in 1:1 CSF/SRPC (w/c = 1.40) formulation were shown to be acceptable in terms of compressive strength, dimensional stability and underwater stability after 90 days curing. The major effect of the Lewatit is to reduce the 90 day compressive strength from 20.5 to 5.8 N mm-2. Further small scale trials have identified a maximum waste loading of between 22.5 and 25 wt% bead Lewatit DN in 1:1 CSF/SRPC, as giving acceptable values for viscosity, set time, bleed at 24 hours and underwater stability after 28 days. Reaction between the calcium and hydroxide ions and the silica framework of the CSF, results in a significant reduction in the pore water pH from 13.2 to 9.9 after 2 days and the implications of this are being addressed.

In terms of radiation stability a 1:1 CSF/SRPC control formulation was stable at a gamma dose of 2000 Gy h-1 to a total dose of 7.67 MGy, although pore water analysis has shown increased $(SO4)2-$ levels in the irradiated system. Overall the 20 wt% bead Lewatit DN in 1:1 CSF/SRPC (w/c=1.40) is found to be acceptable in terms of viscosity, set time, 24 hour bleed, compressive strength, dimensional stability, underwater stability to at least 90 days and a gamma dose of 2x103 Gy h-1 to a total dose of 9 MGy.

METHODOLOGY

Based upon the data accumulated from the above development program, some aspects going back several years, the intention now is to develop a methodology which can apply the formulations for a number of individual wastes and predict a satisfactory formulation for mixtures of wastes. Only those materials which can be anticipated from records or sampling to comprise mixtures are considered. A number of mixtures have been analyzed in this way and current work will test the resultant formulations with the simulated mixtures of wastes in the usual way.

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Session 12 -- MIXED WASTE PROGRAMS AND TECHNOLOGY INTEGRATION
Co-chairs: Leon C. Borduin, LANL
Lloyd McClure, INEL
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THE STARTUP AND STATUS OF THE DOE MIXED WASTE FOCUS AREA*

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ABSTRACT
The Mixed Waste Focus Area began operations in February 1995. Its mission is to provide technology to support design, construction and operation of implementable treatment systems. These treatment systems shall be capable of treating DOE's mixed waste, and shall be developed in partnership with end-users and with continual input from stakeholders, tribal governments, and regulators. This paper describes the program's status at the end of its first year.
Systems Engineering has been implemented. A Complex Needs Study and the Technical Baseline, both based on site visits and the site treatment plans, have been completed. Requirements have been formalized and a program strategy has been developed. The structure and status of the program are discussed in terms of the program strategy.

INTRODUCTION

The Department of Energy (DOE) established the Mixed Waste Characterization, Treatment, and Disposal Focus Area to develop and facilitate implementation of technologies required to meet the Department's commitments for treatment of mixed low-level and transuranic wastes. The Idaho National Engineering Laboratory (INEL) has been identified as the Lead Organization for the technical management of the Mixed Waste Focus Area. A DOE Idaho Operations Office manager has overall responsibility for performance and accountability of all organization elements. Lockheed Martin Idaho Technologies is accountable to DOE-ID for the performance of each functional element. To successfully carry out the DOE's planned approach, the Mixed Waste Focus Area uses capabilities and expertise from across the DOE complex.
The mission of the Mixed Waste Focus Area is to provide technology to support design, construction and operation of implementable treatment systems. These treatment systems shall be capable of treating DOE's mixed waste, and shall be developed in partnership with end-users and with continual input from stakeholders, tribal governments, and regulators. This mission arose from the Resources Conservation and Recovery Act (RCRA) as amended by the Federal Facility Compliance Act. Each DOE site facility was required to prepare a plan for developing treatment capacities and technologies for each facility that generates or stores

mixed waste. Consent Orders have been issued for the implementation of the treatment technologies detailed in the Site Treatment Plans.

The Mixed Waste Focus Area has taken the initial approach of dividing all mixed waste streams into five waste types, based on treatability considerations: 1) waste water, 2) combustible organic, 3) homogeneous solids and soil, 4) debris, and 5) unique waste. These five waste types are used to allocate the mixed waste technology development activities within the Mixed Waste Focus Area.

REQUIREMENTS

The Mixed Waste Focus Area Proposal, written in November 1995, contains the primary set of commitments that drive the Mixed Waste Focus Area. It was analyzed for the top-level requirements that define the Mixed Waste Focus Area deliverables. These requirements are shown in Table I, and can be succinctly stated as: The Mixed Waste Focus Area shall deliver sufficient implementable technology that is correct, complete, acceptable, sponsored, permittable, safe, cost-effective, and timely such that the end-users have sufficient treatment systems to treat essentially all of the DOE Complex's mixed low-level and transuranic wastes.

Table I

STRATEGY

Top-level requirements and two higher level DOE strategy documents, the Environmental Management Strategic Plan and the EM Action Plan, were analyzed. This resulted in the identification of the following seven elements of a strategy with which to structure the MWFA Program.

1. Use a systems engineering approach,
2. Assure technical integration and implementation through Waste Type Managers,
3. Achieve program integration with a central program staff,
4. Prioritize all technology development and demonstration,
5. Assure an end-user for all technology developed or demonstrated,
6. Base technology needs on FFCA Site Treatment Plans, and,
7. Team with stakeholders, industry, and universities.

STATUS

The Status of each element of the Mixed Waste Focus Area strategy will be addressed.

Systems Engineering

Lockheed Martin Idaho Technologies has implemented systems engineering on all major programs at the Idaho National Engineering Laboratory. The systems engineering approach is evident in the development of the top-level program requirements and other aspects of the program, such as the development of the program's technical baseline.

The Systems Engineering element provides systems engineering support at the program level and through representatives on each Waste Type Team. A typical systems engineering effort consists of six basic activities, each of which are performed to support the Waste Type Teams. The six activities are requirements definition and functional analysis, alternatives development, alternatives analyses, verification and test planning, and system integration and control.

Systems Engineering personnel support each Waste Type Team in defining its systems requirements, selection criteria, and performance measurements to help ensure that all activities support the Waste Type Teams defined objectives. In addition, Systems Engineering provides systems analysis tools and techniques as needed, assist in integrating

between the Waste Type Teams, and the conduct of design reviews as appropriate.

Waste Type Managers

Five Waste Type Managers were selected from experienced leaders, proposed by the sites in the DOE Complex, through an open nomination process. The Waste Type Managers are typically from the dominant DOE site compared with needs and inventory of that specific mixed waste type. Waste Type Managers have used systems engineering processes to develop the technical baseline (Published January 1996, and discussed in more detail below, under "Prioritize all Development and Demonstration Tasks.")

Each Waste Type Manager leads a Waste Type Team dedicated to one of the five mixed waste types. The Waste Type Team consists of personnel from multiple organizations across the DOE Complex. Composition of the team is based directly on the skills needed, including technical, legal, regulatory, stakeholder, and systems engineering. The team makeup is dynamic, while the team leadership is maintained constant for consistency and stability.

Members of a Waste Type Team are expected collectively to: a) be fully knowledgeable of waste, including inventories, within the specific waste type, b) understand the customer (EM-30, EM-40, and EM-60) plans for managing that waste, c) be knowledgeable of ongoing and proposed technology development activities that could apply to the specific waste type, and, d) understand the risks associated with their target wastes.

DOE customer needs are a necessary input to the work of the Waste Type Teams. To define the deficiencies or needs of the DOE Environmental Management customers, the Mixed Waste Focus Area analyzed Proposed Site Treatment Plans, and other applicable documents, and conducted site visits throughout the summer of 1995. Representatives from the Office of Waste Management (EM-30), the Office of Environmental Restoration (EM-40), and the Office of Facility Transition and Management (EM-60) at each site visited were consulted to collaboratively define their technology needs. These needs were categorized by the five mixed waste types.

The site visits had several specific purposes: 1) to identify the technology development needs for managing mixed wastes at the DOE sites; 2) to understand the regulatory status/situation at the sites; 3) to status the technology transfer and other privatization efforts at the sites; 4) to identify completed, ongoing, and planned technology development work being conducted by EM-30, 40, 50, and 60 at the sites; and, 5) to identify potential matches between current capabilities and defined site technology needs or "quick wins."

Prioritize all Development and Demonstration Tasks

One of the first products of the Waste Type Teams was a prioritized, defensible technical baseline. The technical baseline is based on the Complex needs for waste treatment systems, and the technologies needed to make those systems implementable. It forms the basis for deciding which technology development activities will be supported by the Mixed Waste Focus Area. The technical baseline forces a strong technical justification for funding technology development. Technology development activities are tied to customer needs and systematically integrated with other development activities. Development of technologies is supported only when they are not commercially available. Development of the technical baseline is further described in more detail in a companion paper.

Program Organization

Organizationally, the Mixed Waste Focus Area includes the following functional units: Technology Coordination, Technical Resource Team, Waste Type Team, Waste Type Manager, Regulatory and External Liaison Program, Program Integration, and Systems Engineering. Waste Type Team members report to a Waste Type Manager. Waste Type Managers programmatically report to the Manager of Technology Coordination as does the Technical Resource Team, completing the integration cycle.

Regulatory and External Liaison, Waste Type Teams, Waste Type Mangers, and Systems Engineering are discussed separately in this paper.

The primary function of Technology Coordination is to participate on the Waste Type Team to coordinate technology development and to facilitate implementation of technologies for mixed waste.

The Technical Resource Team consists of science and engineering experts in waste management disciplines including technology development, characterization, thermal treatment, solidification and stabilization, decontamination, waste handling, and disposal. The Technical Resource Team supports the Waste Type Teams by: 1) evaluating existing, new and proposed technology development activities; 2) providing the technical input to directed calls for proposals to fill technology deficiencies; 3) evaluating proposals and recommending preferred technology solutions; 4) coordinating technical support that crosscuts the Waste Type Teams; and, 5) assisting in the selection of a preferred waste type technical strategy.

The Program Integration and Control element implements and maintains a sound program management process within the Mixed Waste Focus Area. This element provides project cost and schedule planning and control, and document management to the Mixed Waste Focus Area. Program Integration coordinates activities and interests with the other Focus Areas, the Crosscutting Technology Programs, and the Technology Task Plan Principal Investigators. The Mixed Waste Focus Area Program Integration Function handles all formal call letters and conducts the formal transactions leading to approval and changes to the Technology Task Plans with input from the Waste Type Teams.

End-user

The Mixed Waste Focus Area strategy calls for closely working with potential end-users of the technologies being developed. The key objective of this strategy element is to avoid a previous mistake - developing technologies that no one wants. The Waste Type Managers were selected from EM-30 programs specifically to improve the chances that the technology development would "hear the voice" of the end-users. When they came on board, Technology Coordination and the Waste Type Managers conducted site visits to listen to the needs of the end-users. The program is committed to fund only those technology development and demonstration tasks for which an end-user can be identified.

Federal Facility Compliance Act Site Treatment Plan Basis

Federal Facility Compliance Act Site Treatment Plans are key technical drivers of the program. Prioritization is based on waste volumes identified in the Site Treatment Plans. Treatment train and technology deficiency identification begin with a search of the Mixed Waste Inventory Report compiled as a component of the Site Treatment Plans. The achievement of the Level 1 requirements "correct" and "timely" will be based on the waste streams identified, and schedules negotiated and agreed-upon through the Site Treatment Plan process, respectively.

Teaming

In the past four years there has been increasingly greater emphasis placed by DOE on mutually beneficial relationships with industries, universities, tribal governments and the public. The Mixed Waste Focus Area has a strong commitment to assure that all possible teaming arrangements are considered and employed where possible.

To support this strategy element, an efficient regulatory and external liaison system has been established. To ensure success, the Mixed Waste Focus Area is initiating a variety of regulatory and external liaison activities. The Regulatory and External Liaison program element helps Waste Type Teams to acquire regulatory, industry, university, and stakeholder input. This input helps define needs, address regulatory options, provide early involvement of stakeholders in product development. It also supports technology selection, prioritization, and evaluation.

CONCLUSION

The Mixed Waste Focus Area has been organized and is operating. Waste Type Managers representing sites with the major mixed waste inventories and/or problems have the charter to provide the primary direction of the technology development and demonstration tasks. Systems engineering methods have played an important part in the preparation of the initial documents and planning processes. A technical baseline was developed and forms the basis of the Fiscal Year 1997 budget submittal. The program has a strong and active Regulatory and External Liaison Unit dedicated to ensuring stakeholder input and ultimately gain their acceptance.

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MIXED WASTE CHARACTERIZATION, TREATMENT, AND DISPOSAL FOCUS AREA TECHNICAL BASELINE DEVELOPMENT PROCESS

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ABSTRACT

The U. S. Department of Energy (DOE) created the Mixed Waste Characterization, Treatment, and Disposal Focus Area (MWFA) to develop and facilitate implementation of technologies required to meet its commitments for treatment of mixed wastes under the Federal Facility Compliance Act (FFCA), and in accordance with the Land Disposal Restrictions (LDR) of the Resource Conservation and Recovery Act (RCRA). Mixed wastes include both mixed low-level waste (MLLW) and mixed transuranic (MTRU) waste.

The goal of the MWFA is to develop mixed waste treatment systems to the point of implementation by the EM customer. To accomplish this goal, the MWFA is utilizing a three step process. First, the treatment system technology deficiencies were identified and categorized. Then these identified needs were prioritized. This resulted in a list of technical deficiencies that will be used to develop a technical baseline. The third step, the Technical Baseline Development Process, is currently ongoing. When finalized, the technical baseline will integrate the requirements associated with the identified needs into the planned and ongoing environmental research and technology development activities supported by the MWFA.

Completion of this three-step process will result in a comprehensive technology development program that addresses customer identified and

prioritized needs. The MWFA technical baseline will be a cost-effective, technically-defensible tool for addressing and resolving DOE's mixed waste problems.

INTRODUCTION

The Assistant Secretary for the Office of Environmental Management (EM) at the United States Department of Energy (DOE) initiated a new approach in August of 1993 to environmental research and technology development (1). The key features of this new approach included establishment of five "focus areas" and three "crosscutting technology" programs, which overlap the boundaries of the focus areas. The five focus areas include the Contaminant Plumes Containment and Remediation; Mixed Waste Characterization, Treatment, and Disposal; High-Level Waste Tank Remediation, Landfill Stabilization, and Decontamination and Decommissioning Focus Areas. The three crosscutting technologies programs include Characterization, Monitoring, and Sensor Technology; Efficient Separations and Processing; and Robotics. In addition, an Industrial Programs group has been established to support all of the focus areas and crosscutting programs. The major characteristic of the new approach is that aggressive teaming with the customers within EM, through the focus areas, is used to identify, develop, and implement needed technologies such that the major environmental management problems can be addressed, while cost-effectively expending the funding resources.

APPROACH OF THE MIXED WASTE FOCUS AREA

The DOE created the Mixed Waste Characterization, Treatment, and Disposal Focus Area (MWFA) to develop and facilitate implementation of technologies required to meet its commitments for treatment of mixed wastes under the Federal Facility Compliance Act (FFCA), and in accordance with the Land Disposal Restrictions (LDR) of the Resource Conservation and Recovery Act (RCRA). Mixed wastes include both mixed low-level waste (MLLW) and mixed transuranic (MTRU) waste. Mixed low-level waste is defined as waste that contains both hazardous constituents, as identified by RCRA, and radioactive constituents, including alpha-emitting radionuclides below concentrations of 100 nanoCuries per gram. Mixed transuranic waste contains RCRA and radioactive contaminants, including alpha-emitting radionuclides with concentrations greater than or equal to 100 nanoCuries per gram. This mixed waste only includes those materials that have an atomic weight greater than 92, or a half-life greater than 20 years.

The goal of the MWFA is to develop mixed waste treatment systems to the point of implementation by the EM customer. To accomplish this goal, the MWFA is utilizing a three step process. First, the treatment system technology deficiencies were identified and categorized. Then these identified needs were prioritized. This resulted in a list of technical deficiencies that will be used to develop a technical baseline. The third step, the Technical Baseline Development Process, is currently ongoing. When finalized, the technical baseline will integrate the requirements associated with the identified needs into the planned and ongoing environmental research and technology development activities supported by the MWFA. The following sections describe these steps.

Needs Identification and Categorization Process

To define the deficiencies or needs of the EM customers, the MWFA used two mechanisms: site visits and process flow/treatment train development. The MWFA completed the site visits throughout the summer of 1995. Prior to conducting each site visit, the MWFA analyzed Proposed Site Treatment

Plans (2) (PSTPs), or similar documents, to perform an initial assessment of the technology deficiencies at that site. Site representatives from the Office of Waste Management (EM-30), the Office of Environmental Restoration (EM-40), and the Office of Facility Transition and Management (EM-60) were requested to consult with the MWFA to further define their technology needs. Personnel from these programs participated in the MWFA site visit as deemed appropriate by the respective sites.

The site visits had several specific purposes

1) to identify the technology development needs for managing mixed wastes at the DOE sites;
2) to understand the regulatory status/situation at the sites;
3) to status the technology transfer and other privatization efforts at the sites;
4) to identify completed, ongoing, and planned technology development work being conducted by the sites;
5) to identify potential matches between current capabilities and defined site technology needs, which could expedite implementation of a technology for treating actual mixed waste (referred to hereafter as "quick wins").

The data provided in the PSTPs was analyzed according to waste type to determine the sites that have the largest volumetric inventory for each category. The waste type categories, namely Wastewater, Combustible Organic, Homogeneous Solids/Sludges/Soil, Debris/Solid, and Unique Wastes, are defined below. As a result of this analysis, the following DOE facilities were identified for site visits:

1) Idaho National Engineering Laboratory (INEL),
2) Savannah River Site (SRS),
3) Hanford Reservation (Hanford),
4) Oak Ridge Reservation (ORR), 5) Ohio Sites (Ohio),
5) Rocky Flats Environmental Technology Site (RFETS),
6) Albuquerque Sites (AL).

The data generated for this evaluation is available upon request. Based on the information reported in the PSTPs, the MWFA effectively addressed the technology needs for over 80% of the DOE mixed waste inventory through the site visits.

As previously mentioned, the needs identified during the site visits were categorized by Wastewater, Combustible Organic, Homogeneous Solids/Sludges/Soil, Debris/Solid, and Unique waste types. These waste types are defined as follows:

Wastewater waste streams, which comprise approximately 3% of the total DOE Complex mixed waste inventory, include liquids and slurries. Slurries are defined as liquids with a total suspended solids (TSS) content greater than 1% and less than 30%. Liquids and slurries defined as Wastewaters contain less than 1% total organic carbon (TOC).

Combustible Organic waste streams, which comprise approximately 1.5% of the total DOE Complex mixed waste inventory, include liquids and slurries containing greater than 1% TOC, and solids with a base structure that is primarily organic such that a maximum of approximately 20% by weight would remain as residue following incineration. Solids are defined, including sludges, as having greater than 30% TSS.

Homogeneous Solids/Sludges/Soil waste streams, which comprise approximately 48% of the total DOE Complex mixed waste inventory, include waste that is at least 50% by volume inorganic sludges, including water content. Sludges are defined as having a TSS greater than 30%. A sludge

may be a mixture with a stabilization agent that has not properly solidified, or may be a mixture with absorbent materials. This category also includes inorganic particulate, paint waste, and salt waste.
Debris/Solid waste streams, which comprise approximately 46% of the total DOE Complex mixed waste inventory, include waste that is at least 50% by volume materials that meet the EPA LDR criteria for classification as debris (..."material exceeding a 60 mm particle size that is intended for disposal...") This category also includes waste that is estimated to be 50% by volume soil, including sand or silt, rock, or gravel which does not meet the EPA LDR criteria for debris.
Unique Wastes, which comprise approximately 1.5% of the total DOE Complex mixed waste inventory, generally include low volume waste streams such as elemental heavy metals, batteries, reactive metals, explosives, compressed gases, lab packs, and other miscellaneous wastes that present unique treatment problems and are not included in the previously defined categories. It also includes the Final Waste Form and Unknown/Other category wastes [Z and U series waste streams identified in the DOE Waste Treatability Group Guidance (3) document, (DOE/LLW-217), January 1995].
The needs identified by the DOE Complex during the site visits were compiled and documented in the Mixed Waste Focus Area Department of Energy Technology Development Needs Report (Needs Report), November 1995 (INEL-95/0555) (4). Several categories of needs and deficiencies are identified in the Needs Report. These deficiencies and technology gaps are broadly categorized as waste type specific and general, non-waste type specific needs. The waste type specific needs are further categorized as general and site specific needs. The non-waste type specific deficiencies and needs are grouped as waste characterization, container integrity, waste handling, treatment system, and programmatic needs. Over 70 specific needs were identified through the site visits and the information obtained from the PSTPs and other applicable documents.
The MWFA selected Waste Type Managers (WTMs) from across the DOE complex to help resolve the site needs related to each of the five waste types. These individuals were selected from experienced leaders, proposed by the sites in the DOE Complex, through an open nomination process. Their primary responsibility is to ensure that the required technologies are available to allow the EM customers to eventually bring the mixed waste inventories associated with each waste type into full regulatory compliance. For this reason, the MWFA chose WTMs for each waste type with a broad experience base in the specific waste category, as well as an EM background. The insight, complex-wide contacts, and knowledge that these individuals provide are necessary resources to ensure that the needs identified by the sites are integrated into the technical baseline and effectively addressed.
In addition to the needs identified during the site visits, other technology deficiencies were defined. These deficiencies were related to the treatment process flows and treatment trains developed for the DOE complex mixed waste inventory. Treatment process flows and associated treatment trains were defined by the MWFA. These treatment process flows address every mixed waste stream identified in the PSTPs. The treatment process flows were generated based on the preferred treatment options identified by the sites in the PSTPs, in conjunction with the process flow diagrams that had been generated during development of the DOE Programmatic Environmental Impact Statement (PEIS). The primary resource documents for this data were the Mixed Waste Treatment Model: Basis and

Analysis (5), September 1995 (LA-13041-MS), and Analysis of Waste Treatment Requirements for DOE Mixed Wastes (6), February 1995 (BCMusgrave Inc., Livermore, CA).

The WTMs, with support from the MWFA Technical Resource Team (TRT), developed an initial list of the technology gaps associated with the process flow diagrams that were defined for the respective waste types. These needs were identified based on the technologies included in the treatment trains associated with the treatment process flow diagrams developed by the WTMs. The technology gaps identified by the WTMs and the TRT were then compared to the needs and deficiencies listed in the Needs Report.

Although over 70 needs were initially identified, only 30 were considered for evaluation in the MWFA Technical Baseline Prioritization Process. This reduction was due to several factors. Some of the needs were identified for more than one waste type, which resulted in duplications. These were combined into one overall need statement. Other needs identified were very site specific, and not generally applicable to a significant volume of waste within a waste type. These deficiencies were removed from consideration in this prioritization process. However, they will be considered on a case-by-case basis as potential quick wins, as discussed below. Some of the remaining deficiencies listed in the Needs Report are clearly more applicable to other focus areas and/or crosscutting programs. The boundaries of responsibility between the five focus areas are presently being defined by the respective program leads. Finally, some of the deficiencies included in the Needs Report were programmatic in nature or related to MWFA policy definition. The prioritization process was not applicable to these types of needs, since it was developed to perform technical evaluation of deficiencies.

Needs Prioritization Process

After the initial evaluation and disposition of the identified technology gaps and deficiencies, the remaining needs were generalized and rolled up into 30 deficiency categories. These 30 "needs" were then prioritized using a three-phase evaluation process. Phase I of the process included prioritization of treatment process flows. Phase II included prioritization of the technology deficiencies and needs associated with these process flows. Phase III integrated the separate prioritization processes completed in the first two phases to develop a final prioritized list of needs associated with the defined treatment process flows.

Both quantitative and qualitative criteria were used to evaluate the treatment process flows and technology deficiencies. The set of criteria used to prioritize process flows was not the same as the criteria used to prioritize the deficiencies. In addition, each criteria was weighted based on its relative importance within the prioritization framework. The criteria and respective weight and scale of merit were developed by the WTMs and TRT. The three phase prioritization process is described in detail, below.

Phase I of the prioritization process used by the MWFA included prioritization of the identified treatment process flows. As previously stated, treatment process flows were developed for every DOE mixed waste stream based on the data contained in the PSTPs and the PEIS. Process flows were included in the prioritization only if a technical deficiency had been identified for that process flow. As a result, 17 treatment process flows were prioritized in Phase I. The evaluation was based on

quantitative and qualitative criteria. Each criteria was assigned an individual weighting, as well as a defined scale of merit. The criteria, weight, and scale of merit used to prioritize the treatment flow processes are presented in Table I.
Table I
The resulting prioritization of the treatment process flows is shown in Table II.
Table II
Phase II of the MWFA prioritization process involved ranking of the identified technical deficiencies These needs had been defined during the site visits and by the WTMs, as related to the treatment process flows. Prioritization of the technical deficiencies was performed based on a different set of criteria than that used to rank the treatment process flows in Phase I. The criteria were both quantitative and qualitative, as before, but the specific criteria were different. These criteria, along with the associated weight and scale of merit, are listed in Table III.
Table III
Phase III was the final, and most difficult, step of the MWFA Needs Prioritization Process. Phase III involved combining the separate prioritization of technology needs and treatment process flows into a single, integrated prioritization. Several methods for combining these independent prioritization were considered. Some techniques were quantitative, such as multiplying technical deficiency scores by the applicable process flow scores. Other methods were qualitative, such as using a graphical approach to determine the highest frequency for combinations of high priority process flows and deficiencies.
The latter method was chosen because a legitimate statistical method for mathematically determining a score could not be identified that did not give double credit for applicability to multiple treatment process flows. Consequently, the final integrated prioritization was accomplished by constructing a matrix with the 17 prioritized process flows listed horizontally across the top and the 30 prioritized deficiencies listed vertically down the left side. The matrix was then quartered, with the upper left portion given highest priority (high-highs), followed by the lower left portion, then the upper right portion, with the lowest priority section being the lower right portion (low-lows). This order was based on the opinion of the WTMs that the ranking of the treatment process flows (Phase I) had priority over the individual technology deficiency ranking (Phase II). The deficiencies were then prioritized according to their frequency within each quarter, and the priority of each quarter. This lead to a final, integrated prioritization for the deficiencies identified for the DOE mixed waste inventory, as shown in Table IV.
Table IVa
Table IV, cont.
Table IV, cont.
Technical Baseline Development Process
The term "technical baseline" refers to the recommended list of technology development activities that the MWFA will support to resolve the needs identified related to DOE Complex mixed waste. The activities included in the technical baseline will address the highest priority deficiencies, based on the results of the prioritization process.
The needs identification, categorization, and prioritization recently completed by the MWFA constitutes the first phase of the Technical

Baseline Development Process. This phase has been documented in detail in the Mixed Waste Focus Area Integrated Technical Baseline Report Phase I - Volume I and II (7), (DOE/ID-10524, January 16, 1996). The next step, which is in progress, involves issuance of directed calls for proposals within DOE and Requests For Proposals (RFPs) to industry and academia. The prioritized needs and technology deficiencies will be used by the WTMs as the framework for developing these directed calls and RFPs.

To make the directed calls for proposals and RFPs as effective as possible, the MWFA has recently issued a Request For Information (RFI) to industry and academia. This RFI will be used to determine the expertise that exists outside of the DOE complex related to the technical deficiencies that have been identified. The content in the subsequent RFPs and directed calls will be based on evaluation of the responses to the RFI. DOE deficiencies related to technology areas that seem more suited to industry and/or academia will be included in the RFPs. Other technical deficiencies will be addressed to the DOE complex through directed calls. The calls and RFPs issued for FY-1997 will be focused on the top priority deficiencies identified for the DOE complex. The technical task plans (TTPs) and industry/academia proposals that are received in response to these calls and RFPs will then be prioritized.

Prioritization of the individual TTPs and proposals that will eventually constitute the technical baseline is a multiple tiered process. The priority of the TTP or proposal is based not only on the technical merits of the proposed work, but also on the priority of the "need" that it addresses. The priority of the "need", as previously defined, is based on the priority of the treatment process flow system or systems that it is associated with, as well as the priority of the deficiency itself. The MWFA TRT, in conjunction with the WTMs, is currently finalizing the prioritization system necessary to evaluate TTPs and proposals. The evaluation criteria, and associated weightings and scales of merit, will be similar to those used to evaluate the specific technical deficiencies.

The results of the evaluation and prioritization of TTPs and proposals will constitute the MWFA technical baseline. However, the DOE system is very dynamic and constantly changing. Accordingly, the technical baseline will be evaluated semi-annually, and the supporting projects will be reviewed quarterly to ensure that the EM customer priority needs are being met in an effective manner. As the situation in the DOE complex mandates changes to the technical baseline, the WTMs will use subsequent directed calls and RFPs to address new/different needs. The prioritization system developed will be used to evaluate the TTPs and proposals that are submitted in response to these calls to ensure that the work supported by the MWFA is directly addressing the highest priority customer needs.

Quick Wins Disposition

As previously stated, quick wins, as well as identified needs with characteristics similar to quick wins, were not considered in the MWFA Needs Prioritization Process. As a result, these activities are not part of the ongoing Technical Baseline Development Process. This is because these activities are usually site specific and address small waste volume problems. A broad range of activities may be classified as quick wins, but these can generally be categorized into the following four areas.

1. Tasks that provide opportunities to transfer technologies between sites, resulting in expedited treatment schedules and/or reduced costs.

2. Tasks that provide for expedited treatment of actual mixed wastes through RCRA treatability studies, preferably eliminating the target waste stream at the treatment site.
3. Tasks that resolve regulatory issues, allowing expedited treatment of mixed waste.
4. Tasks that provide for more efficient utilization of existing equipment and capabilities within the DOE complex in treating mixed waste.

Presently, almost 30 quick win opportunities have been identified, and additional quick wins will most likely be defined as the WTMs continue to further analyze their respective waste streams. A modified form of the evaluation process and criteria utilized to prioritize the DOE Complex mixed waste needs has been finalized by the WTMs. An internal call for proposals is expected to be distributed to the sites during the first quarter of CY-1996. A set funding level has been established specifically for addressing quick wins, and as the quick wins are prioritized, the available budget will determine the activities that are supported by the MWFA.

CONCLUSION

The initial DOE complex needs identification, categorization, and prioritization is an excellent step toward resolving the needs of the EM customers. The information collected to date is recognized as the first phase in development of an implementable technical baseline. Completion of the initial MWFA Technical Baseline will be accomplished in the ensuing months.

The quarterly and semi-annual reviews of the MWFA program will ensure that the customer needs are identified, addressed, and eventually resolved. In addition, these reviews will ensure that the technical baseline is continually updated and properly documented.

The use of directed calls for proposals and RFPs, which is a departure from the historical approach of issuing general calls for proposals, is seen as one of the most beneficial tools that the MWFA will use. Technically specific calls for proposals will be issued that are directed at resolving particular customer needs. This process is so beneficial because it allows the MWFA to consistently make comparative evaluations of proposals, ensuring that the prioritization process produces the most efficient, cost-effective, timely decisions. The end result will be a technically defensible, documented baseline that is guaranteed to meet the EM customer needs, as they are understood at the time of evaluation.

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PLANS FOR AND PRELIMINARY RESULTS FROM THE DOE INTEGRATED ON THERMAL TREATMENT SYSTEMS STUDY*

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ABSTRACT

The U.S. Department of Energy (DOE) has undertaken a study of Integrated Nonthermal Treatment Systems (INTS) for mixed low level waste to assess the present state of technology and to identify research and development requirements for implementation. For the purposes of this study, nonthermal is defined as processes operating below 660F. Flowsheets and conceptual designs were developed for five systems treating a representative spectrum of mixed waste. Mass balances were performed, residual volumes calculated, and total life cycle costs estimated for each system. This study is patterned after a recently published study of Integrated Thermal Treatment Systems (ITTS). This paper describes the nonthermal systems and preliminary results of the INTS study.

INTRODUCTION

The mixed low level waste in the DOE complex consists of a variety of organic and inorganic solids and liquids contaminated with hazardous and radioactive substances. These waste streams include aqueous and organic liquids, soils, organic and inorganic sludges, and combustible and non-combustible debris. Mixed wastes containing hazardous organic compounds, which are typically treated by thermal technologies, are a particular problem due to the current difficulty of permitting and siting incinerators. Although incineration is the Environmental Protection Agency (EPA) best demonstrated available technology for many organically-contaminated wastes and it can treat a wide variety of wastes, there is a lack of public acceptance due to potential emissions of toxic metals and products of incomplete combustion (1).

These issues associated with incineration, and the need expressed by stakeholders for information on nonthermal systems to allow informed decisions on waste treatment options (2), prompted DOE to initiate the Integrated Nonthermal Treatment Systems study. Five nonthermal treatment systems were studied which included the following technology categories: washing, thermal desorption, organic destruction by chemical processes, aqueous waste and offgas treatment, and waste stabilization. The purpose of this study was to evaluate the potential for treating the DOE mixed waste by nonthermal processes, identify research and development requirements, and to compare these systems to each other with respect to performance, effluent production and disposal volume, and life cycle cost. To allow a subsequent comparison with thermal systems previously studied, the same assumptions and waste profile were used for the INTS study as were used in the earlier ITTS study (3).

For these studies, nonthermal is defined as less than 660F, which is below the temperature at which dioxins are thought to form in an offgas stream and below the temperature at which most metals will volatilize (mercury being the exception). Integrated systems include all facilities required to treat the defined waste streams from the time they enter the facility through disposal. This includes administrative and laboratory support, material handling, separation and destruction of organic materials, immobilization of inorganic materials, treatment of residue to comply with appropriate regulations, and disposal.

The goals for the treatment systems are to remove organic contaminants from the solid matrices, destroy the separated and stored organic material, and stabilize the inorganic contaminants with the solid matrices to meet the Toxicity Characteristic Leaching Procedure tests prior to disposal. The organic contaminants must be removed and/or destroyed to the Universal Treatment Standards to meet the EPA land disposal restrictions. The inorganic contaminants may be removed during the washing or organic destruction processes, and therefore contribute to the secondary waste stream as precipitated heavy metals or radionuclides. This paper briefly describes the systems and technologies that were evaluated as part of the nonthermal studies, the waste stream configuration entering the systems, and the partitioning of the wastes to the various treatment processes. The results of the study are presented in terms of effluent, waste requiring disposal, and cost. It should be pointed out that these analyses are based on preconceptual design concepts, and very limited relevant performance data. A considerable research and development effort will be required before such systems could be considered for implementation for the treatment of DOE mixed waste.

WASTE STREAM DESCRIPTION

The waste profile shown in Fig. 1 represents an average composition of wastes from the 20 largest DOE sites (excluding the Hanford tank waste and Rocky Flats pondcrete), which is about 99% of the stored waste in the complex. These data were extracted from the Interim Mixed Waste Inventory Report (IMWIR) (4) and used in the ITTS and INTS studies. Although the IMWIR data have been updated several times since the ITTS study was performed, the older IMWIR data were used in the INTS study to allow a valid comparison between the two studies. The same waste profile was used for each system, and each system was sized to treat 2927 lbs/hr.a In addition to the waste streams identified in Fig. 1, salts comprised 0.3% of the total, and lead and mercury comprised 1%. The wastes are sorted at

the front end of each treatment system and routed to the appropriate treatment process as shown in Table I.

Fig. 1

Table I

In the following discussion, soft debris is defined as combustible debris (e.g., wood, paper, rags, plastic, etc.), open debris consists of items with easily accessible surfaces that can be decontaminated (e.g., sheet metal, bricks, concrete, rocks, rubble, etc.), and complex debris is defined as items such as pumps, valves, and motors that have internal surfaces that trap contaminants and cannot be easily accessed for decontamination, and that are difficult to shred or break open. Complex debris is a small part of the total waste stream.

TREATMENT SYSTEMS

A summary of the systems evaluated is provided as Table I. In comparison to thermal systems, the waste entering the facility requires extensive characterization, sorting, and size reduction because nonthermal treatment processes are not as versatile or robust as thermal processes. Size reduction requirements will vary depending on the waste matrix and the method of treatment. For example, solidified inorganic sludge will require extensive size reduction to allow access to the organic contaminants trapped within the solidified matrix. Debris may require little or no size reduction, or extensive size reduction, depending on the type of debris and method of treatment.

Bulk metal may be separated from the general category of debris for size reduction, decontamination, and storage for subsequent recycle. Thus, complex bulk metal debris with internal contamination may be recycled, or macroencapsulated in grout without treatment. Other bulk metal items with surface contamination (metal drums, lead bricks, sheet metal, etc.) will be decontaminated and recycled. Lead blankets and lead gloves will be macroencapsulated in polymer in accordance with EPA's debris rule. There are also some debris items that cannot be decontaminated, e.g. highly deteriorated drums and firebrick, which may contain radionuclides but no organic contaminants. Such items will be grouted without treatment. As with the ITTS study, bulk contaminated metallic mercury that can be sorted from the waste stream is amalgamated, and funds have been allocated for treating special wastes, but no specific treatment methods are identified.

All systems use a gas phase corona reactor (GPCR) to destroy trace levels of volatile organic contaminants in the offgas streams (5). The GPCR consists of a cylindrical ceramic or glass reactor. When a high voltage alternating electric field is established across the gap between concentric electrodes separated by dielectric beads, corona discharges occur between the beads creating a plasma that decomposes the organic contaminants in the gas flowing through the reactor.

The quantity 2927 lb/hr is based on a nominal hourly throughput of 2000 lb "combustible" waste, as defined by the ITTS study. The additional 927 lb/hr includes all other waste streams and materials associated with the waste, including metal drums, lead, mercury, aqueous waste, and special wastes (requiring as yet unspecified "special" handling).

Ultraviolet (UV) photooxidation (6), a relatively common and commercialized process, is used in all five systems to destroy organic contaminants in the aqueous waste. Organic compounds are destroyed through an indirect photochemical process that uses UV energy to generate hydroxyl radicals from either hydrogen peroxide or ozone in the bulk

fluid. These radicals oxidize the organic species in the water. Pretreatment of the waste stream is required for proper functioning of the system and to minimize down time and maintenance requirements. This includes removal of suspended solids and immiscible organics to allow penetration of UV light into the wastewater, removal of dissolved solids that may produce scale on the UV lamps or windows, and removal of hydroxyl scavengers that diminish the ability of the system to destroy the organic contaminants.

System #1 - Grout Debris

In System #1, all debris are stabilized in grout without treatment to comply with the EPA debris rule. This is similar to the process proposed in many of the DOE site treatment plans.

Soils and inorganic sludges are treated in a vacuum desorber (7) to thermally desorb high boiling point organics, such as PCBs. Thermal desorption may also remove volatile metals, such as mercury, and volatile metal compounds; however, a process for solubilizing mercury or mercury compounds is included after the desorption step to remove incompletely volatilized mercury compounds. The mercury-contaminated matrix is mixed with a solution of potassium iodide and free iodine (8), and the solids drained and washed with water. A soluble mercury/iodine complex is formed and the wash solution containing the complex is reacted with zinc or iron to precipitate mercury metal and form an amalgam to stabilize the mercury.

Desorbed organics that are condensed in the offgas treatment system, stored organic liquids, and organic sludges are destroyed using a silver-based mediated electrochemical oxidation (MEO) process (9). Several MEO processes are being developed; however, the process evaluated in these studies uses Ag(II) as the oxidizer in a supporting electrolyte of HNO3 at ambient pressure and near ambient temperature. The oxidizer is reduced in these reactions, is regenerated (oxidized) at the anode of an electrochemical cell, and then migrates back into the bulk electrolyte to oxidize the organic compounds. Although Ag(I) forms insoluble AgCl in the presence of halide anions generated during the destruction of halogenated organics, a process has been identified to recover and recycle up to 99.9% of the silver from the AgCl precipitate (9). The principal cathode reaction is the reduction of nitric acid to nitrous acid, which is reacted with oxygen to regenerate nitric acid.

The treated soil is stabilized in grout, and the treated sludge is stabilized in polymer. Secondary salt residue resulting from precipitation of dissolved solids from aqueous waste, the MEO process, or neutralization of acid gases are polymer stabilized.

System #2 - Thermal Desorption

System #2, and all the following systems, involve enhanced debris treatment wherein the debris is treated to remove organic contaminants before stabilization and disposal. In this system all the debris, soil, and inorganic sludge are thermally desorbed using a batch vacuum system, and mercury is removed using the KI/I2 process. Temperatures less than 250F are required when desorbing soft debris to prevent melting of plastics and excessive offgassing of the organic debris matrix. Complex debris is reduced in size to allow access to internal surfaces. The MEO process of System #1 is replaced with a catalytic wet oxidation (CWO) (10) process to treat organic liquids and organic sludges .

The CWO process uses Fe(III) in a hydrochloric acid solution to oxidize waste organics while solvating heavy and/or radioactive metals. Catalysts

increase the oxidation rate for organics, and Fe(II) formed in the oxidation process is oxidized back to Fe(III) by a second catalyzed reaction with oxygen. The process typically takes place at 212-570F, and at pressures of 20-200 psig, depending on the waste feed and composition. The residual solids are filtered from the solution, rinsed, and polymer stabilized for disposal. The reagent is recovered and recycled.

System #3 - Washing

For all solid waste matrices, System #3 uses aqueous washing with selected surfactants or other additives to solubilize and remove organic contaminants; in the process some of the inorganic contaminants are removed as well. As in System #2, complex debris is size reduced to allow access to internal surfaces. An agitation wash process, similar to that used in typical radiation facility laundry, is used for soft debris to prevent rags, paper, etc. from sticking together. A high pressure spray wash (11) is used for shredded complex and open debris, and a soil washing process is used for soils and inorganic sludges (12). The MEO process is used to treat organic liquids and organic sludges. Surfactants are recovered and the treated wastewater recycled back to the washing process to minimize consumptive use of water.

System #4 - Combination with Acid Digestion

Phosphoric/nitric acid digestion (13) is used to decompose combustible/soft debris to decrease the volume of waste sent to disposal, and to treat organic liquids and organic sludges. Complex debris is grouted without treatment, thereby eliminating the need for extensive size reduction, and grout is used to stabilize treated open debris. High pressure spray wash is used to decontaminate open debris, and aqueous soil washing is used to remove organic contaminants from the soil matrix. Vacuum thermal desorption is used to treat inorganic sludges and mercury is removed using the KI/I2 process. Phosphate-bonded ceramics (14) are used to stabilize soils, inorganic sludges, and insoluble salts and oxides, and polymer is used to stabilize soluble salts.

As with all chemical destruction processes, acid digestion requires shredding of soft debris because of the strong surface area dependence of the dissolution and reaction rates. Many organic materials are completely oxidized at atmospheric pressure and below 350F; however, decomposition of more stable compounds may require conditions in the range of 390F and 15 psig. As the reaction progresses, NO and NO2 are released from solution and recovered as nitric acid. Upon cooling the solution, heavy metals and radionuclides will precipitate as highly insoluble phosphates or oxides.

System #5 - Combination with Chemical Wet Oxidation

This system is identical in design to System #4 except that chemical wet oxidation replaces phosphoric/nitric acid digestion, and polymer and grout stabilization replace phosphate-bonded ceramics.

STUDY RESULTS

The results of the study include an estimate of the system effluent (gas, liquid, and solid) and costs. The ASPEN PLUS computer code (15) was used to model each unit operation and simulate the process for each system. Simplifying assumptions are used to combine several unit operations, where appropriate. The code solves for equilibrium chemistry using Gibbs energy minimization techniques, or uses experimental data if available for reactions, but no kinetics are considered. All mass (waste, chemical reagents, air, water, etc.) is accounted for in the calculations, and the mass output from each system equals the input. The output volume is used

in the cost calculations to assess the cost of disposal, and the sizes of process units used in the cost analysis are based on throughput data obtained from the technology developers.

Effluent

The discharge rates of gas, liquid, and solid effluent from the five systems are shown in Fig. 2. The volumetric discharge rates (shown in Fig. 3) of solid waste forms sent to disposal are lowest for systems (#4 and #5) in which soft (combustible) debris is decomposed and is, therefore, not part of the solid waste leaving the systems. However, the difference is small (approximately 1600 lbs/hr or 10 ft3/hr). The ratio between input and output waste volumes varies between 1.02 and 0.80; however, these calculations are based on an assumed waste density for the input waste (64 lbs/ft3) that may vary by a factor of two depending on packing density of the as received containers.

It is assumed that the wastewater is treated and recycled to minimize consumptive use; however, these systems generate more water than is required, so that there is some discharge. As seen in Fig. 2, this discharge is similar for all systems and is approximately 1000 lbs/hr (1.4 gpm) or less. As expected, the gaseous effluent from these systems is relatively small and consists primarily of CO2 from the destruction of organics. Because combustible debris is destroyed in Systems #4 and #5, the offgas from these systems is higher than from the other systems in which all debris is stabilized as solid waste. The higher offgas from the desorption system (#2) compared to system #1 is due to volatilization of the organic constituents of the soft debris, which are subsequently condensed and oxidized.

Cost Analysis

Total life cycle cost is approximately the same for the five nonthermal systems as shown in Fig. 4. The lowest cost system (desorption) is within 8% of the highest cost system (CWO), with the major differences being in the higher operations and maintenance (O&M) and capital costs associated with the CWO system. The largest contribution to the total cost of all the systems is O&M, followed by disposal and then capital cost.

Fig. 2

Fig. 3

Fig. 4

A typical breakdown of subsystem costs for the acid digestion and desorption systems is shown in Fig. 5. Most subsystem costs are similar for all the systems, with disposal and front end costs comprising the bulk of the system costs followed by stabilization. Front end handling consists of sorting, size reduction, and all the costs associated with receipt and preparation of waste for treatment.

Fig. 5

SUMMARY AND CONCLUSIONS

Five nonthermal treatment systems have been evaluated in the INTS study. These systems are similar in effluent discharge (gas, liquid, and solid). Changes in the waste volume range between a 2% increase and a 20% decrease. Total life cycle costs for each system are also similar. The largest contribution to the total cost of all the systems is O&M, followed by disposal and then capital cost. The major subsystem cost contributors are disposal and front end handling.

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MIXED WASTE MANAGEMENT AT FERNALD: MAKING IT HAPPEN QUICKLY, ECONOMICALLY AND COMPLIANTLY

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ABSTRACT
At the end of calendar year 1992, the Fernald Environmental Management Project (FEMP) had approximately 12,500 drums of mixed low-level waste in storage and the Fernald Environmental Restoration Management Corporation (FERMCO) had just begun to develop an aggressive project based program to treat and dispose of this mixed waste. By 1996 the FERMCO mixed waste management program had reduced the aforementioned 12,500 drums of waste once in inventory to approximately 5800 drums. Projects are currently in progress to completely eliminate the FEMP inventory of mixed waste. As a result of these initiatives and aggressive project management, the FEMP has become a model for mixed waste handling, treatment and disposal for DOE facilities.
Mixed waste management has traditionally been viewed as a singular and complex environmental problem. FERMCO has adopted the viewpoint that treatment and disposal of mixed waste is an engineering project, to be executed in a disciplined fashion with timely and economic results. This approach allows the larger mixed waste management problem to be divided into manageable fractions and managed by project. Each project is managed by problem solving experts, project managers, in lieu of environmental experts. In the project approach, environmental regulations become project requirements for individual resolution, as opposed to what had formerly been viewed as technically unachievable environmental standards. With compliant disposition of mixed waste being the primary goal, and achieving compliant disposition as quickly and economically as possible being attendant goals, several general principles have been established in the FERMCO mixed waste program.

1. Wastes must be treated using technology which is proven, reliable, and currently available. Technology development is not performed within the program. Newly developed technology may be considered when that technology will be readily available and proven within a period compatible with the project schedule.
2. Commercial vendors are utilized to the maximum degree possible. The economic incentives for commercial vendors to develop and market treatment technology are utilized and cultivated to the maximum extent prudent.
3. Treatment and disposal technologies which achieve results above and beyond what is compliant are generally undesirable.

Experience at the Fernald site has demonstrated that successful mixed waste management can be achieved through aggressive management of mixed waste management tasks as projects. Projects must be managed around established scope, schedule and budgets. All actions necessary to ensure that the project is successful must be taken. No excuses are acceptable and the project schedule is managed strictly. All projects have a beginning. All projects have an end. Whatever is not on the line between these two events is secondary.

FERMCO SUCCESS AND LEADERSHIP

At the end of calendar year 1992, the Fernald Environmental Management Project (FEMP) had approximately 12,500 drums of mixed low-level waste in storage and the Fernald Environmental Restoration Management Corporation (FERMCO) had just begun to develop an aggressive project based program to

treat and dispose of this mixed waste. FERMCO had also just assigned a manager with strong project management skills to manage and develop a staff to address the FEMP mixed waste inventory.

By 1996 the FERMCO mixed waste management program had reduced the aforementioned 12,500 drums of waste once in inventory to approximately 5800 drums. Projects are currently in progress to completely eliminate the FEMP inventory of mixed waste. As a result of these initiatives and aggressive project management, the FEMP has become a model for mixed waste handling, treatment and disposal for DOE facilities. The balance of this paper addresses fundamental elements of FERMCO's success over the past three years.

DESCRIPTION OF MIXED WASTE PROGRAM

The DOE complex has traditionally viewed mixed waste management as a singular and complex environmental problem. Therefore, personnel possessing expertise in environmental issues alone were assigned to resolve enormous perceived environmental dilemmas.

FERMCO has adopted the viewpoint that treatment and disposal of mixed wastes are engineering projects, to be executed in a disciplined fashion with timely and economic results. This approach allows the larger mixed waste management problem to be divided into manageable fractions and managed by project. Each of these projects is then managed by problem solving experts, project managers, in lieu of environmental experts. In the project approach, environmental regulations become project requirements for individual resolution, as opposed to what had formerly been viewed as technically unachievable environmental standards. The project approach allows waste treatment and disposal to be accomplished while meeting environmental requirements.

The FERMCO mixed waste program was initially established, and continues to be a project based program. Each project has a documented scope, schedule and budget, and a dedicated project engineer. The project engineer is responsible and accountable for all aspects of a specific project, including those activities performed by support organizations.

General Program Principles

Compliant treatment and disposal of waste is an obvious absolute requirement for any mixed waste program. Cost and schedule together form a secondary critical project parameter. Cost and schedule are typically opposing project elements and must be balanced based upon relevant internal and external factors, and good management practice. Scope must be controlled within the confines of good management practice and potentially dynamic technical and regulatory requirements. With compliant disposition of waste being the most important goal in any mixed waste project, and achieving compliant disposition as quickly and economically as possible being attendant goals, several general principles have been established in the FERMCO mixed waste program.

1. Wastes must be treated using technology which is proven, reliable, and currently available. Technology development shall not be performed within the program. Newly developed technology may be considered when that technology will be readily available and proven within a period compatible with the project schedule.

Time is of the essence in any mixed waste treatment and disposal project. Waste must be treated and disposed before the ever changing regulatory environment requires treatment and disposal in a more time consuming, more expensive, and potentially unavailable manner. Waste treatment methods that are proven, reliable, and currently available often are the

same technologies which are economical, commercially prolific, and inherently low risk. Furthermore, DOE complex wide experience has demonstrated that prompt and decisive action is required to solve mixed waste treatment and disposal issues. New and complex treatment methods are not needed for a majority of mixed wastes. Managers of mixed waste programs must utilize the technologies they have available to them today and end their wait for the cure-all technology which is supposedly just around the corner. Mixed waste will never be more easily or economically treated than it can be today.

2. Commercial vendors shall be utilized to the maximum degree possible.

The economic incentives for commercial vendors to develop and market treatment technology must be utilized and cultivated to the maximum extent prudent.

Treatment of mixed wastes utilizing applicable hazardous waste treatment technologies occurs regularly in the commercial sector. Commercial vendors are becoming increasingly aware of the mixed waste challenges and opportunities to be found in the government facilities business area. An environment of such opportunity causes commercial vendors engaged in similar work in other business sectors to make their technology and expertise available within the DOE complex. Furthermore, commercial enterprises can be made aware of government sector mixed waste management issues and these commercial enterprises will perform the research and development to solve these issues at their own expense. Commercial sector vendors are assured to be able to perform any research and development more quickly and economically than the government sector. Mixed waste managers must allow commercial enterprise to work to our advantage and never reinvent the wheel.

3. Treatment and disposal technologies which achieve results above and beyond what is compliant are generally undesirable.

In order to be successful, mixed waste managers must focus on compliant treatment and disposal, not technically ideal treatment and disposal. Mixed waste treatment technologies should be evaluated for compliance with RCRA requirements, including Land Disposal Restrictions, with cost and schedule being considered concurrently. Utilizing technology which achieves results above and beyond that which is compliant is generally more costly than those technologies which are simply compliant. Furthermore, technologies that achieve results above and beyond simple compliance are likely to have a longer planning and operational duration. Expanding costs and extending schedules in implementing waste treatment and disposal activities which exceed regulatory requirements are not good business practices and do not represent proper stewardship of public funds.

At the genesis of the FEMP mixed waste program, the total FEMP inventory of mixed waste was divided into groups which could be treated and disposed, or directly disposed, together. These groupings were generally based upon like treatment standards and similar physical matrices. Each individual waste grouping was then defined as a project.

A scope statement is written for each project. This scope statement must define the project as necessary to completely define the established waste grouping, determine what treatment and disposal alternative will be utilized, implement treatment and disposal, and complete all project closure tasks. The scope statement will define these elements of the project in general terms as technical information available at that time in the project permits. Upon completion of the project scope statement, a

detailed project schedule is prepared. This project schedule includes all tasks necessary to complete the project as defined in the project scope statement. The project schedule divides the entire project into tasks of sufficiently short duration that monthly schedule updates will demonstrate if the project is behind or ahead of schedule.

Each activity contained in the project schedule is resource loaded. The resources for each task include all project management personnel, support organizations, subcontractors and materials necessary to complete that task. The sum of the costs of all resources for all tasks constitute the project budget, excluding contingencies and management reserve. This budget includes all costs which can be attributed to the project from inception through final close-out. Each project is completely cost self-contained.

Disposal under Case-by-Case Variance to LDR - Debris Rule

On May 14, 1993, the U. S. EPA promulgated the second and final Case-by-Case Variance to the Land Disposal Restrictions (58 FR 28506). This variance applied to hazardous waste which met the definition of debris as published by U. S. EPA on August 18, 1992 (57 FR 37194). U. S. EPA defined debris as any man-made material or nonindigenous rock or soil which is at least 60 mm in one dimension. U. S. EPA recognized that many waste streams which contain debris also contain other materials which do not individually meet the definition of debris. For these waste streams, U. S. EPA stated that any waste stream which contains a mixture of debris and non-debris is to be managed as debris if the waste stream is primarily debris based upon visual inspection. Within specified limits, hazardous waste which met this definition of debris could be disposed directly to the land without treatment. U. S. EPA required that all individuals who wished to dispose hazardous debris waste under this variance provide to U. S. EPA no later than August 12, 1993, proof that the individual had made a good faith effort to identify treatment capacity for the hazardous debris waste in question. U. S. EPA stated that contact with ten potential treatment vendors would constitute a good faith effort. The May 14, 1993 variance expired on May 8, 1994.

FERMCO published a Commerce Business Daily (CBD) announcement on June 18, 1993 seeking qualified vendors which could supply capacity to treat mixed debris waste in storage at the FEMP. 50 potential vendors responded to the CBD announcement. No qualified vendors were identified. FERMCO utilized the data collected from the various responses to the CBD announcement to document a good faith effort to identify a qualified treatment vendor in a document pursuant to the U. S. EPA rule making of May 14, 1993 (58 FR 28506). This document was approved by the Department of Energy - Fernald Area Office and transmitted to the U. S. EPA on August 9, 1993. An informational copy of the document was also transmitted to the Ohio EPA.

FERMCO identified 368 containers of mixed debris waste which would qualify for land disposal under the aforementioned rule making. All 368 containers were inspected for free liquids using real-time radiography. Containers which held free-liquids were emptied onto a sorting table. All mixed debris waste was placed back into the container from which it was removed and liquid waste removed from the container was placed into storage for disposition as part of the Liquid Mixed Waste Project discussed later in this document.

The mixed debris waste identified for land disposal was divided into two groups for the purpose of sampling, analysis and profiling pursuant to

Envirocare of Utah's waste acceptance criteria. These groupings were established based upon the physical nature of the debris waste, the manner in which the debris waste was generated and the potential contaminates of concern present in the waste.

Samples were extracted from a statistically determined number of randomly selected waste containers in each waste grouping by FERMCO hazardous materials workers. These samples were analyzed by a State of Utah certified analytical laboratory. Envirocare waste profiles were completed by FERMCO Hazardous and Mixed Waste Management personnel for each waste grouping based upon the Utah certified data and transmitted to Envirocare for review and approval. Upon approval of these profiles, Envirocare provided FERMCO with a Notice to Transport (EC-1800). This notice documents Envirocare's approval to ship the waste described on the submitted profile. FERMCO released the first shipment of six total shipments of mixed debris waste to Envirocare on April 12, 1994 and released the sixth shipment on April 21, 1994.

All mixed waste at the FEMP which met the definition of debris and was eligible for disposal at Envirocare was disposed during the aforementioned waste disposal campaign.

Disposal of D018-D043 & F-Listed Waste

On September 19, 1994 (59 FR 48045), U. S. EPA promulgated a final rule establishing the treatment standards for newly identified toxicity characteristic waste. The types of waste referred to are those hazardous wastes carrying waste codes between D018 and D043 inclusive. This rulemaking went on to state that for hazardous wastes carrying these waste codes which also contain radioactive materials, the effective date of the rule is September 19, 1996. Because this final rule does not apply to mixed waste until September 19, 1996, mixed wastes carrying waste codes between D018 through D043 inclusive may be disposed to the land without treatment until that date.

In response to this rulemaking, FERMCO identified all mixed waste carrying only waste codes between D018 and D043 inclusive. As a part of the same campaign, FERMCO identified a population of listed mixed wastes carrying 'F' codes which exhibited concentrations of the RCRA treatment constituent of concern below the established RCRA Land Disposal Restrictions treatment standard. Listed wastes which meet treatment concentration standards without treatment may be disposed as if that waste had been in fact treated. The specific waste streams identified were a D018 through D043 trash waste stream and a cement stabilized F-listed waste stream.

Each container of D018 through D043 mixed waste was examined utilizing real-time radiography. Any container found to contain free liquids was excluded from shipment to Envirocare and re-assigned to the proper mixed waste treatment project.

Samples were extracted from a statistically determined number of randomly selected waste containers in the D018-D043 and F-listed waste streams by FERMCO hazardous materials workers. These samples were analyzed by a State of Utah certified analytical laboratory. Envirocare's waste profiles were completed by FERMCO personnel for each of the two subject waste streams based upon the Utah certified data and transmitted to Envirocare for review and approval. Upon approval of these profiles, Envirocare provided FERMCO with a Notice to Transport (EC-1800). This notice documented Envirocare's approval to ship the waste described on the submitted profile. FERMCO released the first of four total shipments

of cement stabilized F-listed mixed waste to Envirocare on March 3, 1995 and released the fourth shipment on March 6, 1995. FERMCO released the only shipment of D018-D043 mixed waste on March 30, 1995.

Liquid Mixed Waste Project

The Liquid Mixed Waste Project encompasses approximately 2,300 containers of mixed liquid waste ranging in size from 5 gallons to 110 gallons. Like the Mixed Waste Stabilization Project and the Mixed Waste Chemical Treatment Project described later in this document, this project is being documented under CERCLA Removal Action 9, "Removal of Waste Inventories." The CERCLA Work Plan for this project was approved by the U. S. EPA and the Ohio EPA prior to the initiation of waste transfer operations. Many of these 2,300 containers hold both liquids and sludges. The liquids will be treated at the K-1435 TSCA Incinerator at the Department of Energy's Oak Ridge facility and will likely be disposed by TSCA Operations. The sludges in these drums will be treated and disposed as part of the Chemical Treatment Project described later in this document.

The TSCA Incinerator accepts liquid wastes in bulk tanker truck deliveries only. For this reason,the subject liquid wastes must be bulked before shipment. Furthermore, the 2,300 containers of waste included in this project represent in excess of 100 separate waste streams generated at different times and by various means, demonstrating that each of these waste streams is within the waste acceptance criteria at the TSCA Incinerator would be cost and schedule prohibitive.

For the purpose of bulking, the various waste streams were divided into compatibility groups based upon information available from the characterization file for each of the various waste streams. Five compatibility groups were identified which encompassed all of the waste streams within the 2,300 container population. In order to assure compatibility of the wastes when actual commingling of the wastes occurs in the field, samples of the various generic types of waste from each compatibility group were secured. These samples were combined in the FEMP analytical laboratory in the same order and ratios in which the actual wastes in the field would be combined. Laboratory commingling of these waste samples and monitoring the commingled waste for possible physical or thermal reactions was performed as per ASTM Method 5058, Standard Test Method A. No physical or thermal reactions were observed in any of the combined waste forms.

In order to combine the five compatibility groups of waste, portable tanks generically known as "Frac Tanks" were rented. A total of six tanks were rented in order to provide one backup tank in the event of a leak or other unplanned occurrence in any of the five primary tanks. The portable tanks are approximately 40 feet in length, 8 feet in width, 11 feet in height, and have a total capacity of approximately 21,000 gallons. Each tank is equipped with a single tandem axle for transport of the tank while empty only. The tanks are also equipped with side and top manways, OSHA compliant stairs and catwalks leading to the top manway, high-level alarms, bottom loading fittings with lockable ball valves, fluid level indicators, and carbon air emissions filters.

The six tanks were field installed within a poly compound inflatable containment system. This poly containment system is physically similar to and resembles a rubber raft. The secondary containment was fabricated of materials resistant to the various wastes to be placed in the portable tanks.

The design of the temporary waste storage area comprised of the six portable storage tanks and inflatable secondary containment was designed to comply with State of Ohio regulations regarding tank systems for storage of hazardous waste. The primary driver from Ohio RCRA regulations for the temporary waste storage area is the capacity of the secondary containment. The secondary containment must have a volume adequate to contain 10% of the total volume of waste stored in the various tanks within the containment or the volume of the largest tank within the containment, whichever is largest, in addition to a 25 year rain event occurring over a 24 hour period. In this project, the volume of the largest tank is the controlling factor.

The transfer of liquid waste is achieved with the use a 2 inch diameter dual diaphragm pneumatically powered pump. All wetted surfaces of the pump are stainless steel and the diaphragms are Teflon. The pump is connected to the bottom loading valve of the tank to which the liquid waste is to be transferred via a 2 inch diameter high pressure Teflon lined and wire reinforced hose. The suction side of the pump is connected via an identical hose to an approximately 48 inch long, 2 inch diameter stainless steel tube with a 1/16 inch stainless steal screen welded over the end of the tube. Each of these hose sets is equipped at each end with stainless steel ball valves and female stainless steel camlock fittings. The temporary storage tanks and pump are equipped with male stainless steel camlock fittings. The transfer of liquid waste is achieved by configuring the hoses in the aforementioned manner, starting the pump, and carefully placing the end of the stainless steel tube into the waste storage container until all liquids within the container have been transferred. All power tools utilized on this project are pneumatically driven and sparkless, all hand tools are sparkless, and all flashlights used are intrinsically safe.

Transfer of liquid phase waste from existing storage containers into portable tanks began on May 1, 1995 and was completed July 19, 1995. Sampling personnel from the Oak Ridge facility sampled Batch 1 pursuant to TSCA Incinerator Waste Acceptance Criteria on May 26, 1995 and Batch 5 on July 25, 1995. As of September 19, 1995, FERMCO has made twelve total tanker shipments of waste to the TSCA Incinerator. The first shipment was released from the FEMP on July 27, 1995 and the twelfth shipment was released on September 19, 1995. The twelve shipments of waste that have been made comprise a total of 165 cubic meters of liquid mixed waste. A total of 123 cubic meters of liquid waste remain to be shipped. Shipment of these liquid wastes is expected to be complete during the spring of 1996. Waste shipments are currently suspended for the winter due to cold weather and resultant thickening or freezing of the waste.

Mixed Waste Stabilization Project

Approximately 1,550 containers, primarily 55 gallon drums, of solid phase waste contaminated with RCRA regulated heavy metals are stored at the FEMP. This waste are being treated onsite utilizing mobile stabilization equipment and disposed at Nevada Test Site.

This project is being documented under CERCLA Removal Action 9, "Removal of Waste Inventories." Documenting this project as a Removal Action allows RCRA regulated wastes to be treated at the FEMP without a RCRA Part B permit for treatment. CERCLA and its implementing regulations, "The National Oil and Hazardous Substances Pollution Contingency Plan," state that actions under CERCLA are not subject to state administrative requirement, i.e. a RCRA Part B permit, in states with an authorized RCRA

program, such as Ohio. However, actions under CERCLA are subject to state substantive requirements.

On December 19, 1994 a Commerce Business Daily announcement was published describing the waste to be stabilized, stating that a Request for Proposal for onsite stabilization of the waste described would be available in the future, and requesting that interested and qualified vendors provide a statement of interest to FERMCO. Approximately 70 statements of interest were received.

On April 10, 1995 a Request for Proposal (RFP) for mobile mixed waste stabilization services was transmitted to all respondents to the Commerce Business Daily Announcement. This RFP included descriptions of all waste to be stabilized, including a description of the general physical character of the waste and applicable RCRA waste codes for each waste stream. The RFP also included other documents necessary for a vendor to prepare a proposal, such as the Mixed Waste Stabilization Project approved Project Plan. Proposals were accepted on May 2, 1995 and the stabilization subcontract was awarded on May 30, 1995.

The Ohio EPA stated that their review and approval of detailed equipment drawings and process procedures for this project were necessary before treatment may begin. Many types of stabilization equipment are utilized for the stabilization of hazardous waste. Therefore, the details requested by Ohio EPA can not be provided until a vendor is chosen and the requested information can be submitted to FERMCO by the vendor. After award of the stabilization subcontract to the successful vendor, the vendor has approximately 30 days to submit documentation listed in the RFP to FERMCO, including detailed equipment drawings and operational procedures. This detailed information will be combined with various types of environmental information required in a CERCLA work plan and submitted to the U. S. EPA and Ohio EPA for review and approval. Once approval of the work plan is secured from both agencies, FERMCO will authorize the vendor to mobilize and begin stabilization activities. Stabilized waste will be placed in half height white metal boxes and shipped to the DOE Nevada Test Site for disposal.

The CERCLA Work Plan for the stabilization project, including the engineering mentioned above, was provided to the Ohio Environmental Protection Agency (Ohio EPA) and the U. S. Environmental Protection Agency on September 7, 1995. The FEMP received approval to proceed with stabilization field operations from the Ohio EPA on September 29, 1995. FERMCO authorized the stabilization subcontractor to proceed with mobilization of treatment equipment immediately. FERMCO completed an Operational Readiness Assessment of the Stabilization Project on November 20, 1995 and received DOE-Ohio Field Office permission to proceed with operations on November 21, 1995. The stabilization subcontractor began stabilizing waste on November 16, 1995 and as of February 9, 1996 had stabilized a total of approximately 1100 drums of waste.

Mixed Waste Chemical Treatment Project

This project encompasses approximately 4,000 containers of mixed waste which represent all the mixed waste which will remain in the FEMP mixed waste inventory after completion of the previously discussed projects. The wastes included in this project represent the most challenging wastes to treat at the FEMP. A majority of the wastes in this project will require multiple treatment steps or technologies to achieve compliance with RCRA Land Disposal Restrictions. This project is termed the Chemical

Treatment Project because the multiple treatment steps required to treat these wastes will be chemical in nature.

A Commerce Business Daily announcement was published on March 16, 1995 requesting expressions of interest from qualified parties to provide chemical and physical treatment for various types of mixed waste. As of May 30, 1995, 36 responses to the announcement had been received. Responses have been received from well established and proven vendors as well as previously unknown vendors of unknown capability. The expressions of interest received have varied in form from detailed explanations of the vendor's experience and capabilities to simple requests to receive any future Request for Proposal.

Current schedule projections indicate that the Request for Proposal for mobile Chemical Treatment will be issued near the end of February 1996. Completion of the Chemical Treatment Project is scheduled for November 1997.

Treatment flow diagrams for the various waste streams in this project have been developed. Treatment processes with multiple treatment steps for one waste stream are referred to as "treatment trains." Various combinations of the following technologies are anticipated to be necessary:

1. Solvent Extraction - Sludges and Soils
2. Decontamination - Lead Solids
3. Macroencapsulation - Lead Solids
4. Deactivation - Reactives and Oxidizers
5. Neutralization - Acids and Caustics
6. Precipitation - Barium Salts and Mercury Salts
7. Washing - Debris
8. Amalgamation - Elemental Mercury
9. Chemical Oxidation - Waste Waters

CONCLUSIONS

Remove Historic Barriers to Success: Mixed waste can be managed compliantly, effectively and economically in a timely manner. Do not allow the traditional reasons why mixed waste can not be treated and disposed prevent project progress. Barriers to successful management of mixed waste are usually issues of perception, not fact. Aggressively remove barriers to success.

Utilize Technology Which Exists: Utilize mixed waste treatment technology which is proven, reliable and currently available. Technology exists to treat and dispose most mixed waste streams. Avoid spending time and money on developing new technologies or further researching existing technologies. Do not accept the "not invented here" argument. Implement mixed waste treatment projects utilizing current technology now. If an option exists; use it. Mixed waste treatment is cheaper now than it will ever be again.

Take Advantage of Commercial Sector Resources: Utilize subcontractors to treat waste whenever possible. The private sector recognizes their potential for profit if they can solve the DOE complex's mixed waste challenges. Allow subcontractors to absorb the cost of research, development, equipment procurement, and process risk. Take advantage of commercially proven expertise in waste treatment. Do not perform work in house that someone else can do for you cheaper, faster and better.

Be Compliant, Not Perfect: Plan treatment and disposal projects which achieve compliance, not the perfect solution. The ideal or most advanced

technology to treat waste is not necessary. Use the quickest, most economical, safe and compliant treatment and disposal option available. Projectize the Work: Divide your total inventory of waste into manageable portions that have a beginning and an end. Determine the scope, schedule and budget for the work. Assign responsibility, authority, accountability, and resources. Everything has a beginning. Everything has an end. What is not on the line is irrelevant. Get on the line. Start now. Finish!
Take Action: If waste can be disposed without treatment; dispose it. Determine what compliant treatment is available for waste requiring treatment and aggressively move forward to accomplish that treatment. Take advantage of regulatory latitude. Do not wait for new technology. Tomorrow never comes. Move forward today. Their is no excuse for failure to act on available treatment and disposal options. Start! Finish!

12-5

DIAL'S INTEGRATED PLASMA TORCH DEVELOPMENT PROGRAM

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ABSTRACT

Accurate characterization and control of plasma torch facilities for mixed waste remediation is currently a major effort at the Diagnostic Instrumentation and Analysis Laboratory at Mississippi State University. This work is being supported by the U.S. Department of Energy within the Office of Science and Technology (EM-50) of the Office of Environmental Management. Although plasma torch systems have been used commercially for many years, most notably in material science applications and for the treatment of incinerator ashes, the extension to mixed waste presents challenges. Specific issues are concerned with the composition of the input waste stream, the longevity of torch components such as electrodes and vortex generators, refractory wear, the quality of the final waste form, downstream gas compositions and the performance of air pollution control devices. Evaluation of these factors requires a systematic approach including instrument development, materials studies, systems integration, modelling and control system development.

Over the past two years, work at DIAL has proceeded in these areas. The plasma torch facility has been assembled. Testing of refractories capable of withstanding the harsh environments in the torch furnace have been reported. Work on instrumentation for characterization of the plasma and

for the assessment of gas stream conditions, chemistries, and emissions has been performed. Efforts directed at non-intrusively measuring glass viscosities are in progress. Knowledge of facility parameters and specific measuring objectives (heavy metals, inorganics and organic concentrations, viscosities, and plasma properties etc.) provide usable data which can be related to the operation of the plasma torch and associated facility components. Integration of these data into a suitable control scheme allows a precise means for efficient operation of the system. In what follows the DIAL plasma torch facility, some of the instrumentation being applied to the characterization of the process, and the control systems are described.

PLASMA TORCH FACILITY

DIAL's thermal test facility includes two plasma torch systems as well as a combustion test stand. Both plasma torches are hollow electrode DC torches. The smaller of the two systems uses a 100 kW plasma torch mounted in a four foot diameter chamber capable of vacuum operation. The chamber is fitted with several optical ports and is used strictly for "clean" experiments such as plasma diagnostics and instrument development. The 100 kW torch can be used in either the transferred or the non-transferred mode of operation. In non-transferred mode the plasma torch produces an electrical arc from its internal hollow anode electrode to an electrode which is on the front of the plasma torch. A flow of gas through the torch produces a plasma plume which exits at the front of the plasma torch. In transferred mode the electrical arc from the internal hollow electrode attaches to an external electrode. A graphite billet is used for the external negative electrode for the 100 kW system.

The larger system is a 250 kW hollow electrode DC torch, see Fig. 1. This system is intended for actual vitrification experiments. The plasma torch has electrical 3-axis position control (as does the 100 kW system) which is useful for power control and for maintaining a consistent temperature throughout the melt. This plasma torch is usually operated in the transferred arc mode. A graphite electrode at the bottom of the vessel acts as the negative electrode until a sufficient amount of soil or surrogate waste is added to the system and a conductive melt is established. The melt then acts as the negative electrode. Four Optics ports on the top of the vitrification vessel allow optical access for diagnostic instrumentation. Optical penetrations are also located directly downstream of the furnace.

Fig. 1

The 250 kW system is used for instrument and control system development as well as for vitrification research and testing. Soil from the Savannah River site as well as a surrogate material designed by WSRC have been vitrified using this system (1,2). A series of DIAL/WSRC refractory tests were performed using 32 refractory samples to determine what type of refractory might be useful in plasma torch vitrifiers.

The offgas from both of these plasma torch systems is fed to a combustion test stand which can be used to provide secondary combustion or simply as a pathway for the plasma systems offgas to get to the pollution control system. Optical access is provided at many locations along the test stand channel. The pollution control system includes a quench, venturi scrubber, and packed column tower. This system provides both particulate removal and acid gas neutralization. Dual HEPA filters follow the wet system and are used to remove smaller particulates. Figure 1 shows a block diagram of the 250 kW system and the pollution control system.

DIAGNOSTIC INSTRUMENTATION

Many parameters are of interest for characterizing and controlling a plasma torch facility. Specific diagnostics can be applied to measurements of the input waste stream, plasma properties within the torch furnace, metal, particulate, and gaseous emissions, and specialized techniques for evaluating the quality of the final waste form. Methods for evaluation of the partitioning of nuclear species are underway at various laboratories and are not discussed here. A brief description of some of the instruments under development are given below. Work on sensors for inlet waste streams is planned. Some of the diagnostics, for example, FTIR and LIBS, may be suitable as continuous emission monitors as well as for process control. Additional work is in progress for fully characterizing the benefits of these applications.

Plasma Diagnostics

A major concern with the use of plasma torches for hazardous waste vitrification is the need to prolong electrode lifetime (3,4,5). Because unchecked erosion of the electrode into the torch cooling system may lead to catastrophic failure, effective monitors for electrode erosion and wear are needed. The goals of the present work include characterizing several fundamental torch plasma parameters, using these results to develop monitors for torch electrode wear and failure, and investigating techniques for extending the electrode lifetime. Studies have been performed to evaluate the stability of a plasma torch arc and its suitability for spectroscopic interrogation. Subsequent investigations have been concerned with obtaining temperature and electron density profiles of the torch arc, and with evaluating the torch electrode erosion rate through the observation of atomic copper emissions from the plasma. Initial studies have also been performed using electrodes doped at a given depth with an indicator material. Monitoring the plasma emission for signals associated with this indicator has been used to signal erosion to a particular depth.

Experimental studies are currently carried out using the 100 kW plasma torch operated in the transferred mode using nitrogen, air, or argon. A single 2f lens is used to form a 1:1 image of the torch arc on a fiber optic cable. This cable is routed to a 0.5 meter spectrometer and the resulting spectra recorded using a charge coupled detector (CCD). The input to the fiber is mounted on a horizontal translation stage, allowing the observation of spectral emissions from any position across the arc image. This horizontal scanning capability allows the reconstruction of profiles of the plasma temperature, electron density, and emission intensity. These results may be compared with two-dimensional images of the plasma arc obtained using narrow band interference filters and a CCD camera. Emission from a specific element (e.g. copper generated by electrode erosion) may be imaged by selecting appropriate interference filters. Although the gas flow from the torch rapidly becomes turbulent after exiting the plasma torch electrode, spectral emissions from near the electrode orifice are quite stable and are amenable to spectroscopic interrogation.

Using the relative intensities of atomic emissions from the plasma allows a Boltzmann plot determination of the plasma temperature. For example, during nitrogen operation, line-of-sight average temperatures ranging from approximately 6000 K at the arc edge to approximately 12,000 K at the arc center are consistently observed. The electron density profile of the plasma torch arc may be determined from the full Stark width of the

hydrogen b line emission (6). The addition of a small amount of hydrogen into the torch operating gas allows the observation of this line. For nitrogen operation the electron density profile rises smoothly from a minimum of approximately 6 x 1020 e-/m3 at the arc edge to 2 x 1022 e-/m3 at the arc center. As part of an effort to monitor the erosion of the primarily copper torch electrode, copper atomic emissions are evaluated as a function of position within the plasma arc. These results clearly demonstrate that the most intense copper emission occurs at the edge of the plasma arc.

One of the most promising results obtained thus far has been the use of a doped electrode to signal erosion to a particular depth. Electrodes are modified by drilling a small hole (or several holes) parallel to the electrode axis, then filling this hole with silver solder. Atomic silver emission is detected from the plasma only after the surface of the electrode is eroded to the depth of the silver solder. In Fig. 2, emission spectra recorded before and after erosion into the silver solder are depicted. Silver emission is clearly visible at 520.8 nm after arc contact with the solder, yet clearly absent before. This results suggests that doped electrodes provide a simple yet effective monitor for electrode wear, and may be used to warn of an imminent failure of a torch electrode.

Fig. 2

Methods For Determining Viscosity

A critical parameter in the vitrification process is the viscosity of the molten glass which has an effect on the quality of the waste form and the processing of waste (7,8). Two methods are currently being investigated as on-line monitors. The first is directly applicable to the measure of viscosity in a rotating furnace which is an integral component of many plasma torch facilities. The method is based on correlation of the fluid flow dynamics with the viscosity at different rotational velocities of the furnace. The second method being developed will measure the viscosity of the molten glass as it is being poured. The technique involves determining the reflection coefficient of ultrasonic shear waves from a solid-molten glass interface. This coefficient is a function of the viscosity of the molten glass. Methods are being investigated using laser generated shear waves, as well as with conventional ultrasound.

Work done at Westinghouse Savannah River Company (WSRC) has established a glass viscosity window of 20-100 poise for the Defense Waste Processing Facility (DWPF), and also for other melters to be used for low-level wastes (9). If the viscosity is too low, < 20 poise, excessive foaming may occur. Also, increased convection currents in the glass will cause increased refractory and melter electrode erosion and/or corrosion. Higher viscosity glasses (> 100 poise) may lead to plugging of the tap and/or voids in the product. Additives (fluxes containing alkali metals) can be used to reduce the viscosity of molten glass, but these additives also weaken the molecular structure of the glass. Thus, excessive or unnecessary use of viscosity reducing fluxes will be detrimental to the ability of the glass to pass leachability tests.

Viscosity Measurement in Plasma arc Centrifugal Treatment (PACT) System

The PACT system consists of a large, cylindrical furnace vessel, open at the top, which rotates about its vertical axis (10,11). Typically, this cylindrical drum might be 6 feet in diameter and will rotate at about 40-50 rpm. The wastes to be vitrified are introduced into the top of the drum. The rotation of the drum and its contents (untreated waste and the

glass products of vitrification) creates centrifugal forces which force the drum contents toward the outer periphery of the drum. A tap is located in the center of the furnace floor; however, centrifugal forces keep the drum contents away from the drain hole. As the drum contents rotate, they periodically pass underneath a plasma torch. After sufficient wastes have been fed into the drum, the feeding is stopped, and the wastes are processed until they are vitrified. The rotation of the drum is slowed, and the glass flows through the tap and into a mold.

DIAL has developed a novel idea for measuring the viscosity of the glass in the PACT system prior to a pour. This idea takes advantage of the fact that, in the PACT system, the glass whose viscosity is to be measured is contained within a rotating drum. The behavior of rotating liquids has been the object of intense study. Greenspan (12) discusses the "spin-up" problem, in which a liquid-filled container at rest is suddenly started rotating at a constant angular velocity. The liquid in the container is also initially at rest, and gradually acquires the same rotation speed as the container. The characteristic spin-up time is proportional to

Eq. 1

where W is the angular velocity of the container, L is a characteristic length scale of the container, and n is the liquid viscosity. The time required for the liquid in a rotating container to adjust to a change in the rotational speed of the container is hence dependent on the viscosity of the liquid. The concept involves therefore changing the rotational speed of the furnace and monitoring the response of the glass. A numerical model has been developed for the dynamics of the spin-up process, and an experimental investigation is underway.

Laser Ultrasonic Viscosity Measurement

Theoretical results show that the shear wave reflection coefficient from a solid-fluid interface is a function of the viscosity and density of the fluid (13,14). The reflection coefficient is a complex number and either its phase or magnitude can be used to calculate the viscosity. The density and shear wave speed of the solid also enter in as factors. By choosing solids with desirable properties, the reflection coefficient can be made sensitive to different viscosity ranges. With this in mind, the goal is to determine the actual viscosity in the 20-100 poise range to within 10 poise and to be able to determine when it is not within this range. Further discussions on ultrasonic methods are given elsewhere (14,15).

In conjunction with the theoretical analysis, a series of experiments has been performed using conventional ultrasonics. In order to measure the reflection coefficient of shear waves, an ultrasonic shear wave transducer was mounted on one end of a solid block. The other end of the block was immersed in the viscous liquid. For most of the experiments a 2.25 MHz shear wave transducer was used in pulse-echo mode. A standard ultrasonic pulser unit served to excite the transducer and receive and amplify the reflected signals, which were recorded with a digital oscilloscope and transferred to a computer. Using algorithms developed during the theoretical studies, these signals were processed to predict the viscosities of the various liquids tested. Results are collected in Table I. Agreement within 10 poise was obtained in many of the trials.

Table I

Experiments were also performed to demonstrate the feasibility of laser ultrasonics to measure viscosity. The technique is basically the same as with conventional ultrasonics except that ultrasound is generated by a

pulsed laser and detected using a laser interferometer. The technique is completely noncontact as far as the transducer is concerned. The feasibility of this technique was demonstrated using piezoelectric transducers instead of an interferometer to detect the reflected shear wave. The results of these experiments are shown in Table II and indicate good agreement.

Table II

Further development in both the conventional ultrasonic and laser ultrasonic approaches is in progress. These efforts include further development of an interferometer detector, studies on improving the signal to noise ratio of the measurements, and pilot scale experiments.

Metals Emissions And Glass Composition

Determination of metals partitioning within the plasma torch facility is of interest from process control and regulatory perspectives. DIAL is currently investigating three methods, Laser Induced Breakdown Spectroscopy (LIBS), Laser Optogalvanic Spectroscopy (LOGS), and inductively coupled plasma (ICP) spectroscopy where the ICP is operated on air. LIBS can serve a dual role for compositional determination of the glass and for process control. The ICP is being designed as a portable field unit and may also find application as a downstream module for low-flow gas streams as an organic destruction unit. LOGS has the potential for quantification well below current regulatory levels and will most likely find use as an emission monitor. Details of the ICP work can be found in a companion paper submitted to this conference.

Laser Induced Breakdown Spectroscopy (LIBS)

LIBS is a laser based advanced diagnostic technique for measuring the concentration of various elements in the test medium (16-21). In the LIBS technique, a pulsed laser beam is focused at the test point to produce a spark. The spark in the focal region generates a high density plasma which produces and excites various atomic elements in the test volume. Atomic emission from the plasma is collected with a collimating lens and sent to the detection system. The intensity of the atomic emission lines observed in the LIBS spectrum are then used to infer the concentration of the atomic species.

DIAL has been actively involved in evaluating various application of LIBS for environmental remediation (16-22). LIBS has been applied to a harsh, turbulent and highly luminescent coal fired magnetohydrodynamic (MHD) gas stream (17). A preliminary study of the determination of Uranium (U), Plutonium(Pu) and Neptunium(Np) by LIBS has also been performed (23). The details of the LIBS experimental setup is described in references 16 and 22. In brief, a frequency doubled 532 nm Nd:YAG laser beam is focussed at the sample. The emission light from the laser-induced plasma is sent to the detection system by an optical fiber. Data acquisition and analysis were performed using a notebook computer. A portable, mobile, and versatile LIBS system has been developed for various applications. It has been used to measure the concentration of toxic metals in the off-gas from a Savannah River (SR) surrogate vitrified with DIAL's plasma torch facility (18). Off-gas measurements have also been made at the Western Energy Technology Office (WETO) torch facility/Mountain State Energy (MSE), Montana (19). In addition, the performance of this system has been evaluated in the Advanced Analytical Instrumentation Demonstration (AAID) test at Science Applications International Corporation (SAIC)'s STAR Center, Idaho Falls, ID. (20) Various LIBS measurements have been evaluated to develop a process control for hazardous waste remediation

(22). It has been used to monitor the major species in the melt glass from an EnVitco melter test run at Clemson University (23). The results of the LIBS measurements of the concentration of metal in the off-emission and melt glass have showed that LIBS can provide the information about metal partitioning during hazardous waste thermal processing. Recently, LIBS spectra were recorded in the off-gas of DIAL's plasma torch facility during various SR surrogate test runs. LIBS spectra in a wavelength region of 404-421 nm was used to measure the concentration of Pb, Ce and Fe. Broad vibrational peaks due to CN are also observed in this region. CN is produced from the reaction of C and N which are produced in the spark. Other elements present in the off-gas emission were identified from spectra in various spectral regions. Those identified were: Al, B, C, Ca, Ce, Cd, Cr, Cs, Fe, K, Mg, Mn, Na, Pb, Si, Ti, Y, and Zr. Special attention was given to the spectral regions with Pb, Cr, Cd, Cs, and Ce atomic transitions. The LIBS spectra were also collected continuously to study the variation of metal concentration with time. The typical sample rate in this test is 3.5 seconds. Generally, the metal concentration increases right after new feed was added to the plasma torch. Then the metal concentration was nearly constant for a few minutes and then starts decreasing. However, this also depends on the arc attachment point in the melt glass which is quite random. The concentration ratios of various metals are also monitored with time. Figure 3 shows the concentration of Cr, Fe, Pb, and Ti inferred from the LIBS data collected on March 6, 1995. The time that briquettes were added is marked with an x in the figure.

Fig. 3

Laser Optogalvanic Spectroscopy: (LOGS)

The concentration of metallic species entrained in the off-gases of practical thermal treatment systems can be monitored in real time at ultra-low levels using DIAL's Laser Optogalvanic Spectroscopy (LOGS) system by extracting a sample of the off-gases via a slipstream. Since the metallic species in the downstream portion of the off-gas system exist primarily as particles rather than as elemental or molecular species, an atomization source is used to atomize the particles, permitting the measurements of low concentrations. In a LOGS measurement, a pulsed, tunable laser is tuned into resonance with a transition of a species (atomic or molecular) in the atomization source, temporarily increasing the excited state concentration of that species. Because the energy necessary for ionization is less for an excited electronic state than for the ground state, the rate of ionization temporarily increases due to laser-enhanced electron impact ionization, and/or due to direct laser photoionization. The process can be monitored as a transient (tens of microseconds) voltage change if a high voltage electrode (relative to ground) is inserted into the atomization source. The concentration of the species is obtained by relating the magnitude of the LOGS signal intensity to the concentration of the species of interest via a calibration curve. Only the LOGS response in the vicinity of the electrode is probed. Because LOGS uses electrical rather than optical detection, this technique alleviates problems associated with monitoring small absorptions or weak fluorescence in the presence of a strong optical background signal. LOGS has inherently greater sensitivity than optical detection techniques because the collection of charges can be significantly more efficient than the collection of photons. Table III compares our current limits of detection for LOGS for selected RCRA and

radioactive metals to those attainable by air ICP, (24) which is a competing real-time, on-line technology. It should be emphasized that air ICP values in Table III are the current state of the art and the LOGS values reported in the table were obtained without optimizing the operating conditions; optimization of the LOGS operating conditions (higher detection electrode voltages, higher laser powers, optimization of detection electronics) will decrease these LOGS limits of detection by between one and, in the best cases, three orders of magnitude.

Table III

Permanent Gases and Products of Incomplete Transformation

Current studies aimed at characterizing the gas streams resulting from air and nitrogen plasma operations are contained in a companion abstract submitted to this conference. Fourier Transform Infrared (FTIR) spectroscopy studies are also being directed toward the determination of particle size distributions. Such measurements are important in mixed waste remediation facilities for measurements where conventional extraction techniques are restricted owing to radiation guidelines. Other efforts which will be described in more detail in the accompanying abstract include an advance MS/MS method and the use of GC/MS and gas analyzers.

Fourier Transform Infrared Spectroscopy

Fourier transform infrared (FTIR) spectroscopy offers a number of advantages for process screening and component quantification. The technique is based on the absorption or emission of infrared radiation as a molecule undergoes a transition from one vibration-rotation level to another. Every molecule, except homonuclear diatomics (N2, O2, etc.), will exhibit a unique infrared signature. The extent of absorption or emission of the infrared light will depend on the concentration of the molecules present in the flow, the gas temperature, and the effective optical path length. The method is rapid and capable of quantifying many molecules from a single measurement.

A number of different applications to mixed-waste remediation processes are being pursued at DIAL. These studies include on-line, non-intrusive, emission and absorption measurements, particle scattering studies, and conventional sample extraction. Certain limitations will apply to the on-line and extractive measurements. In a non-intrusive configuration the path length will be limited by the distance across the facility channel, whereas the extraction measurements are performed at a longer distance consequently resulting in lower limits of detection. The on-line measurements do not involve sample handling. In this case, however, instrument calibration may be complicated by the high facility temperatures as compared to those conveniently available for establishing calibration sets in the laboratory. It is anticipated that both on-line and extractive instrument configurations will find application in providing data for control schemes.

Measurements have recently been conducted on the PACT plasma torch facility at WETO operated by MSE (25). Extractive measurements were conducted upstream and downstream of the NOx control unit while passive (emission) spectra were collected at a high temperature facility location between the plasma torch furnace and the secondary combustor. The purpose of the emission experiments was to examine the composition of the gas stream as it exited the torch and to determine a gas stream temperature at the facility location. This FTIR emission experiment offers an advantage over conventional thermocouple measurements in that heat

transfer corrections are not necessary, concerns about corrosion of the probe are eliminated, and higher temperatures are measurable.

Typical emission spectra are collected in Fig. 4. Molecules which account for the band and line intensities are H2O, NO, and CO2. Emission lines from CO were not observed in any of the spectra downstream of the torch furnace indicating that any organics present in the feed were converted to CO2. Determination of a rotational temperature from the NO fundamental lines was hampered by the interference from H2O. A spectrum of H2O at a temperature of 1200 K and a concentration of 3% was generated using the HITRAN data base and the FASCODE infrared signature software (26,27). The synthetic H2O spectra was then subtracted from the emission spectra to leave the NO rotational envelope which was then analyzed based on the fact that the different rotational lines will be populated according to the Boltzmann distribution (28). Corresponding temperatures averaged 1070 K or 1466 F. The subtraction process, described above, did not seem to alter the determined temperature but does tend to increase the scatter in the plots.

Fig. 4

Previous work on high temperature combustion gas streams indicated that the spectra will systematically change with changes in combustion conditions (29). This work resulted in a patent application and the corresponding instrument may find application to secondary combustion of plasma furnace gas streams.

CONTROL SYSTEMS

DIAL's plasma torch and process controls structure depicted in Figure 5 consists of a supervisory control, coupled with an artificial intelligence (AI) advisor, with monitoring links to various diagnostic instruments. This provides both robust operation and flexibility, with embedded graphics, data archival, on-line reconfigurability, and correlation capability as some of the convenient features. The system will accommodate 40 inputs and 20 outputs; more with minor modification. A list of I/O presently configured is given in Table IV.

Fig. 5

Table IV

Diagnostic instrumentation for use in control schemes must meet specific requirements. The systems must be sufficiently sensitive and rugged. Factors such as signal acquisition, control structure, and actuation must all be considered in the development of a viable control package.

Signal acquisition can span considerable range. For some diagnostic monitors, the control system must accommodate a few seconds inherent latency. Other detectors provide information over a spectrum that must be reduced to a few variables. The control system at DIAL has a dual acquisition structure - one for continuously available signals, and another for asynchronous data. This allows inferential techniques to be applied as needed.

The diagnostic instruments are being developed for specific end uses. In some cases the systems are expected to provide valuable information at multiple locations within a facility. Examples of this are the LIBS and FTIR systems. Additional experimental effort is underway for identification of the most appropriate uses of the diagnostics with respect to the control structure. This work involves the consideration of all aspects of the facility, waste preparation, the primary chamber and final waste form, the downstream components, and associated emissions.

Actuation is an item of concern in vitrification processes, since there are only a few variables available for actuation. Generally these must exceed the number of quantities of concern (metal volatization, other off-gas hazards, product quality, volume reduction, process efficiency ...). And it would be nice if there was a one to one correspondence between actuation and effect - but there are other side effects. This is another reason for a strong experimental effort, with definition of a viable control structure as the final goal.

Considerable data has already been obtained from many of the diagnostics described in this work. These measurements have contributed to the establishment of the data base upon which the control scheme is being developed. Examples of this work have been described in other submissions from DIAL personnel (see references).

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THERMOCHEMICAL TREATMENT OF MIXED WASTE BY USING POWDER METAL FUEL

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ABSTRACT

Thermal treatment (e.g. incineration) of combustible wastes provides a high reduction of volume and yields residues suitable for conditioning, transport and disposal. A variety of thermal treatment concepts has been tried throughout the world. Scientific and Industrial Association "Radon" uses on industrial scale excess air incineration and slagging incineration also. Now a new concept of thermal treatment (e.g. thermochemical treatment) of waste is under development at SIA "Radon". Investigations were done on thermochemical treatment of wastes by using special powder metallic fuel of very low off gas production. Thermochemical treatment involves chemical destroying of the organic portion of wastes as in conventional incineration. The main advantage of new fuel is its reduced gas production as well as involving of waste water and other components in chemical reactions. As result very high efficiency occurs for treatment process particularly for humid waste. non complexity of equipment is also an advantage of new treatment process. Powder metallic fuel consists of aluminum, magnesium and aluminum-magnesium alloy with some technological additives. It provides heat release 25-27 MJ per kg. Products of fuel oxidation are non volatile. Actual gas production consists only 17-55 liters per kg of fuel. Duration of treatment process is 1-2 hours, after this the ash is discharged for conditioning and/or disposal. Gas purifying system operate in the regime much softer than at conventional incinerators due to redox reactions between fuel and off gases particularly acidic components.

Preliminary tests were performed on the treatment of low level simulated radioactive waste. Their specific radioactivity was lower than 10 kBq per kg. Ion exchange resins, paper, wood, etc. were burnt. Biological active waste such as animal's cadavers were treated in field conditions. Practical complete incineration occurred. A mobile installation will be designed for the incineration of organic radioactive wastes on the base of these investigations.

INTRODUCTION

The necessity of thermal processing of mixed waste (containing both organic and non-organic harmful substances as well as radioactive nuclides) leaves no doubt due to their biological, chemical and radioactive danger. The danger for man and for the environment can increase with time because of the decomposition processes in the organic components of the waste and a possibility of spontaneous fire. In the latter case the danger is not only in the fire itself, but also in the combustion products, the most harmful among which are radioactive aerosols.

The technique of thermal decomposition leads to a considerable reduction of waste in volume and obtaining of fire-safe product. Due to the destruction of chemically complex components of the waste and the obtaining of simple oxides as the end result, the product of the thermal processing has minimal biological and chemical activity. At the thermal

treatment the majority of radionuclides remains in solid phase products, thus making it easy to locate them for further conditioning and burial. The process of the thermal decomposition of the waste can be realized in a number of ways. Among the most developed ones one can note the processes of burning with excessive oxygen, pyrolysis, shaft furnace treatment, plasma processing, the treatment in a shaft furnace with plasma burners for melting the ash residue, etc. (1,2). All the techniques for burning organic radioactive waste are characterized by a large amount of gas, therefore the majority of the burning devices form the gas cleaning system.

Within its activities SIA "Radon" has come across the necessity of treatment small quantities of combustible radioactive waste collected at sixteen Specialized centers of the system "Radon" (3). This was the aim of developing a technique and a mobile unit for burning radioactive waste with a capacity of 20-30 kg/hour. The fact that the unit should be mobile requires the burning technique to have low gas emission and a low chemically toxic components content in gases.

THE CHOICE OF BURNING METHOD

Usually an additional absorption of heat is required for burning waste. It is obtained by burning gas or liquid hydrocarbon together with the waste. When the fuel burns, resulting gases transfer the heat to the waste. A large amount of fuel is needed for burning waste, on average 0.25 kg of liquid oil products per 1 kg of waste (2). If kerosene, solar oil, or furnace fuel is used as fuel, 14.6-14.8 kg of air is required for a complete combustion of 1 kg of waste. 15.5-15.8 kg of gaseous products (carbon oxide, water) are obtained as a result, as well as approximately 0.1 kg of solid phase products in the form of ash. As a result, the amount of heat taken away by outgoing gases is much greater than the amount of heat remaining in the solid product. Now we must remember that the fact that the heat is taken away by gases is not a favorable process, it becomes clear that the thermal processing with the help of hydrocarbon fuel is very inefficient not only from the point of view of the complexity of gas purification, but also energetically wasteful.

In reality the production of outgoing gases during the thermal processing of the waste is not an indispensable characteristic of the process, and it depends on the method of its realization. First of all it is necessary to choose the fuel so that the end products of the combustion were non-volatile even at the temperature of the process. If fuel is chosen successfully, it is possible to have some of the lightly volatile products of the waste destruction interact with the fuel and also remain in a solid phase. The most successful for this purpose today are small off gas metal fuels - powder metal fuel based on aluminum and magnesium blends.

THE CHARACTERISTICS OF POWDER METAL FUEL

The powder metal fuels (further PMF) have been created on the basis of technological blends produced in Russia on an industrial scale. These blends are designed for contact heating objects with humidity of various degrees. The content of PMF is shown in Table I.

Table I

The PMF has a bulk density of 0.5-0.6 kg/dm3. It is chemically and physically stable, hit-proof, with the self combustion temperature of 500C. The PMF has a high porosity, approximately 50 - 60%.

The main combustion products of the PMF are aluminum and magnesium oxides. At the combustion of the PMF the main products is solid slag,

approximately 99.2 wt.%. The gas phase carries away only 0.8% of the mass of the burning products, leaving almost all the energy of chemical reactions in the solid phase. The air stoichiometrical coefficient of PMF is 3.15 kg/kg. At the same time, since the components of the PMF react with water, the water stoichiometrical coefficient is 1.2 kg/kg. The heat release of PMF is within 25 - 27 MJ/kg. Since the gas productivity of the PMF is small, the heat is transferred from the fuel to the waste by contact heating.

It is important to note that magnesium and its compounds are not harmful. The same is true for aluminum. Moreover, they are only slightly soluble in water. As a result, the combustion products of the PMF are safe for the environment. The fact that the PMF forms a compound with gaseous products of the combustion of organic waste, such as CO_2, SO_2, CO, NO_2, NO, reduces the influence on the gas purification system. This partially solves the problem of trapping harmful components which result from burning, as well as reducing the corrosion of the equipment. The interaction of both metals with water is favorable for burning damp waste.

COMBUSTION OF POWDER METALLIC FUEL

When burning the PMF the most important reactions are oxidization of metals which take place in a diffusion controlled regime. At the same time the burning of the PMF, unlike, for example the burning of termite welding mixtures is slow. The burning time of PMF can be between 7 and 45 minutes depending on the type of feed for the oxidizer.

The process of burning the PMF can be theoretically divided into two stages: the oxidization by own oxidizer and the oxidization by oxygen from the air.

The PMF includes potassium nitrate or sodium saltpetre, which acts as own oxidizer at the initial phase of the burning. This stage is short, 20-30 seconds. During this stage the metals in the fuel burn in the oxygen which resulted from the decomposition of nitrates. The process of the decomposition of nitrates of alkaline and alkaline-earth metals takes place in three stages: at first nitrate decomposes into nitrite: $2NaNO_3=2NaNO_2 + O_2$. After this the nitrite is decomposed into oxygen and peroxide: $2NaNO_2 = Na_2O_2 + O_2$. Finally peroxide decomposes, forming oxygen: $Na_2O_2 = Na_2O + O_2$. Using as fuel the energetic reducing agents Mg and Al a deeper decomposition of nitrite may take place: $Na_2O + Mg = MgO + 2Na$ + 44 kkal (174 kJ). This is accompanied by heating the PMF to temperatures 400 - 450C, which is necessary for the steady burning of the fuel in the oxygen from the air.

At the second stage of the burning the components of the PMF are oxidized by oxygen and water steam. This stage, depending on the conditions of burning, lasts 10 - 45 minutes. Since the reactions are diffusion controlled, the time required for this process greatly depends on the scale of burning. The process starts at a temperature of 400C, then the temperature increases to the temperature of burning the aluminum. The main compounds formed during this process, are AlN, Al_4C_3 and MgO at a ratio of 1:1:1. When the melting of aluminum is finished, the majority of oxides are Al_2O_3 and MgO.

The diffusion of gaseous components through PMF's pores and the diffusion of oxygen through the films of oxides appearing on the surface of the fuel's parts limit the speed of the process. However, as temperature rises, the evaporation of components is possible. The MFP burning process

can be easily controlled by supplying air into the area of burning. In this case temperature varies between 800 and 2500C.

EXPERIMENTAL UNIT

An experimental stand was constructed to test new technological process of waste burning by using PMF. This stand contains a vertical multi-sectional furnace with a diameter of 400 mm.Each section of this furnace is 400 mm height, the furnace having 1 - 4 sections depending on required productivity (see Fig. 1). Smaller netted drums with the diameter 350 mm were used to maintain waste inside of each section. These small drums were filled preliminary by waste-PMF mixture. The sections were installed on the stand, after this charged by filled (or empty) drums. The last section was connected to gas purifying system.

Fig. 1

The gas (air) inlet is located on the bottom of furnace, the inflow of air from tightness of furnace being negligible. Typical simulated radioactive and non radioactive wastes were used in experiments: paper (dry and damp), wood, cleaning cloth, PVC and ion exchange resins. In some cases this waste was preliminary prepared for experiments, for example PVC was shredded before loaded.

Special attention was given to the thermal destruction of wet ion exchange resins based on a polystirol matrix, containing up to 50 wt.% of water. The level of contamination due to presence of Cs-137 was not higher than 10 kBq/kg. The discharging of furnace was provided by dismantling furnace and removing netted drum with ash residue.

EXPERIMENTS

The purpose of a number of experiments done at the experimental stand was to determine the appropriated method for loading furnace (netted drum) with the PMF and waste, their proportions and mode of burning initiation. As the result it was selected the method when a layer of the PMF 0.5-1 cm thick is put on the lower fire bar grate, followed by a layer of waste about 10 cm thick, then another layer of the PMF, and so on, with the top layer in the section being the PMF. The proportion between PMF and waste was chosen within 0.15 - 0.20. The combustion process was started with initiating the bottom layer of the PMF. The average chemical formula of fuel was Mg30.35 Al8.99 C0.712 H1.148 K0.098 N0.098 O0.03.

It was established that combustion process go in two regimes depending on the air supply. When air is lacking, there takes place the burning of the PMF and the pyrolysis of the waste immediately near to the fuel layer. The pyrolysis gases do not burn. This regime leads to the process fading away, as the burning does not spread over the whole furnace. When a required portion of air is supplied, a controlled spreading of burning over the whole furnace takes place. In this case some amount of pyrolysis gas burns when mixing with the air inside the furnace. The burning process here can be divided into two stages: the pyrolysis and coke formation stage and the coke burning stage. During the pyrolysis stage the drying, pyrolysis of the organic materials and the burning of the pyrolysis gases take place. This stage is characterized by the formation of a large amount of burning gases. As the waste decomposes and coke is formed, the burning comes into the next stage. During the coke burning stage, the coke burns forming CO2 and ash. When the air is excessive all the processes are accelerated and higher temperatures are achieved.

The combustion process from the initiation of burning up to discharging of ash residue lasts 1 - 2 hours. The temperature in the furnace during

combustion took a value up to 1150oC, average temperature being between 850 - 1000oC.

RESULTS AND DISCUSSION

Complete destruction of organic part of waste was achieved in all experiments. The incomplete combustion (non organic) in the ash residue in different experiments was between 0 and 10wt.%. Radionuclide release directly from furnace during experiments has been not higher than 18%, usually being lower than 5%. The productivity of furnace was estimated to be about 10 kg/h per section for ion exchange resins Specific capacity in this case was about 80 kg/m2h.

The results of analysis of exhaust gases formed during the combustion of wet ion exchange resin are shown in Fig. 2.

Fig. 2

The analysis of ash-slag residue was performed by using X-ray difractometer (DRON-2) and scanning electron microscopy (REMMA-202). The following phases were found in the slag: MgO(main products), spinel $MgAl_2O_4$, a small amount of nitrides AlN, Mg_3N_2, the aluminum and magnesium alloy as non burnt fuel, Al and K_2CO_3. Besides, there was detected MgOHCl as a result of Cl trapping from off gases.combustion in field conditions

Other experiments were carried out in field conditions without special equipment for the combustion of non radioactive biological dangerous waste. Many experiments have shown that it is necessary to use not less than (0,40 - 0,45) kg of oil products per 1 kg of the biological waste in the field conditions (even during dry summer weather). The new feature when using the PMF for burning biological objects is that the system PMF + biological object, accordingly to thermodynamic calculations, burns even without air due to the exothermic reactions of proteins, fats and water with metals in the PMF at temperatures up to 1100-1500 K. Moreover, the presence of excess air leads to an increase in temperature of up to 3100 K.

A no-equipment method of combustion biological objects in field conditions has been developed. In accordance with this method a trench of a volume approximately equal to the volume of the biological object is dug in the place where the biological object is found. Taking into consideration the direction of air, special air pipe trenches are dug up to the trench. A 1-2 cm layer of the PMF is spread over the bottom of the trench. The biological object is placed in the trench, after this 1 - 2 cm layer of the PMF is also put on top of the biological object. The proportion waste - PMF is 0.2 wt. The combustion process after the initiation of burning takes place without any interference from the operator. Duration of complete combustion is 5 - 6 hours for a 100 kg biological object.

CONCLUSIONS

A new concept of thermochemical treatment of waste is under development at SIA "Radon". This concept is based on applying a new sort of fuel to maintain incineration of various waste. The main advantages of new fuel are: its reduced off gas production, heavily heat transfer, involving of waste water and other components in chemical reactions. As result very high efficiency occurs for treatment process particularly for humid waste. Preliminary investigations on thermochemical treatment of waste by using special powder metallic fuel of very low off gas production demonstrated challenging feasibility of this method.

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Session 13 -- RECYCLING AND REUSE OF RADIOACTIVE CONTAMINATED MATERIALS AND FACILITIES
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13-1

A METHODOLOGY FOR ASSESSING RECYCLING AND DISPOSAL COSTS ASSOCIATED WITH SURFACE CONTAMINATED SCRAP METAL

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ABSTRACT
Decommissioning and demolition of facilities or the discontinuation of some programs has left the Department of Energy (DOE) with radioactively contaminated scrap metal some of which is potentially recoverable or recyclable while the rest requires disposal. While there is pressure within DOE to conserve resources and recycle scrap metal, there is also a demand that actions be cost effective, and it is often whether recycling is economically viable. This problem is increased by the lack of a consistent approach to making appropriate cost comparisons and by changing market and other variables.
This paper, based on an report in preparation, addresses this issue as it relates to surface contaminated scrap metals, especially carbon steel. More specifically the report attempts, in some ways, to standardize and facilitate the cost analysis process and analyzes the sensitivity of various cost factors. A series of graphs are included which facilitate the analysis of costs associated with the survey, decontamination and recycle of surface contaminated scrap metal and allow comparisons of these factors to the cost of sizing, packaging, transportation, and disposal of those metals. The graphs have been developed to cover the range of costs covered by these elements. The range of costs were identified using data in the literature and actual costs incurred at DOE sites that are decontaminating metal for free release and are compared to data from sites currently disposing of radioactively contaminated scrap metal.
The graphs facilitate analysis of the effects that various cost elements have on the overall cost. If the total cost of recycling exceeds the cost of disposal, the graphs can be used to determine areas requiring cost reductions necessary to achieve comparability with the cost of disposal. Cost elements include 1) costs for preparing the metal for transportation; 2) options for purchasing or renting a variety of

containers that could be used for recycling; 3) transportation costs, including transportation of metal that cannot be decontaminated and secondary waste from the decontamination process; 4) decontamination costs; and 5) disposal costs, including disposal of metal that cannot be decontaminated and secondary waste from the decontamination process. Transportation costs have been simplified so that estimates may be calculated knowing only the distance or the weight of the metal to be transported. Disposal costs are based on costs currently charged by disposal sites, both DOE and commercial, and expected future costs at new disposal facilities. Decontamination costs are per pound costs based on vendor quotes provided to various DOE sites and in published vendor data. Other variables that may be assessed for their impact on the overall cost of either option include the percentage of the surface requiring decontamination, the percentage of the metal that can be decontaminated to free release guidelines, the cost of surveying the metal and that of truck or rail transportation, the distance to the disposal site or decontamination facility, the quantity of secondary waste generated and the cost of burial.

SUMMARY AND CONCLUSIONS

Based on the case developed in this paper using parameters reported by DOE sites, it is evident that there are two major factors in determining whether to decontaminate and recycle structural steel or to package it for disposal. These are the cost of decontamination which is roughly 63% of the total cost of recycling and the burial charge which constitutes more than 56% of the total cost of disposal. Chart 1, Recycle Option Costs and Chart 2, Disposal Option Costs (Fig. 1), show appropriate relative percentages of the various cost elements for the case used in this study.

Fig. 1

Based on the costs used in the sample case, the break even cost for disposal (at a decontamination cost of $0.76 per pound) is about $23 per cubic foot. Of the DOE sites currently accepting waste from other sites, all charge more than $40 per cubic foot, except the Nevada Test Site (which is less than $20 per cubic foot). Therefore any decontamination process that costs less than $0.76 per pound would be cost effective for sites that do not use the Nevada Test Site for disposal.

COST COMPARISON METHODOLOGY FOR MILD CARBON STEEL

The data presented below can be used in conjunction with Figs. 2 through 11 to estimate and record specific costs associated with the recycling of mild carbon steel versus those associated with disposing of the same material as waste. Descriptions of how each graph is to be used are presented below. Costs estimated from the figures should be recorded and totaled in Table I. The values in the table are from the sample case. Log-log scales are used so when interpolating between two lines the mid-point between them represents a cost about 70% of the way between the lines.

Based on the size and type of container to be used, select the appropriate line from Fig. 2 of Graphic 1. Draw a horizontal line from the total weight of metal being analyzed to the line which represents the container to be used and the weight of metal to be shipped in each container, i.e., the net weight of contents. This is point P-1. It identifies the number of containers needed for the quantity of metal to be shipped. Extend the horizontal line through Figs. 3 and 4 in Graphic 1. From intersection point P-1 draw a vertical line to determine the

number of containers needed. Extend this line down through Figs. 5, 6 and 7 in Graphic 2. If different containers are being considered for disposal and recycle, two values will have to be determined.

Fig. 2

Fig. 3

Fig. 4

If the weight of metal that will be placed in each container is not known, but the bulk density of the metal in the package is available, multiply the bulk density by the interior volume of the container to get the weight of metal in the container. Be sure to check this weight with the capacity of the container, and use the smaller of the two numbers to prevent overestimating the amount of metal that can be shipped in a given container (specific containers are certified for a specific gross weight). If a given container has two weight limits (depending on its use as a strong tight container or a DOT approved 7A container) and the weight of metal in a container falls between the two, use the larger number and the appropriate cost for that container. (See Table II on Graphic 1 for container weights and allowable weights of contents.)

The on-site preparation cost is the sum of the surveying, sorting, sizing and loading costs for a particular project. Find the appropriate line on Fig. 3 in Graphic I which best approximates this total cost per ton of metal and label the point of intersection with the horizontal line for the weight of metal to be dispositioned as point P-2. Draw a vertical line from point P-2 to find the total on-site preparation cost. If site policy calls for different degrees of preparation related to any of these activities (i.e., a more detailed survey for disposal vs. shipment to an NRC licensed decontamination facility, or more sizing to get the material into smaller shipping containers for disposal) then two values should be determined, one for disposal, C1D, and one for recycling, C1R. Enter these costs in Table I.

The cost charged to decontaminate metal for recycling is negotiated between the DOE site and the vendor who provides the service. Typical costs range from fifty cents per pound to slightly over a dollar per pound. Under some circumstances this cost may include the cost of transporting the metal to the vendor in which case a separate transportation cost would not be charged.

If all the metal requires decontamination, find the average decontamination cost line on Fig. 4, in Graphic I, or approximate between lines, and the intersection of this line with the horizontal line for the weight of metal to be processed. This is point P-3. Draw a vertical line to the lower axis to find the total decontamination cost, C4R. Enter this cost in Table I, under the Recycle column only.

In the event that a significant fraction of the metal will not require decontamination (i.e., already meets release guidelines), it may be cost effective for the vendor to survey all the metal prior to decontamination. While this would increase the processing cost for the contaminated metal, it could lower the overall cost by avoiding decontamination of already clean metal. Based on data for survey costs it is estimated that the additional survey could cost between $0.015 per pound and $0.10 per pound of contaminated metal. Handling costs for container unloading, movement of the metal onsite and delivery to a central pickup point are estimated to add an additional $0.10 per pound to the cost for surveying only. Therefore, if a significant fraction of the metal does not require decontamination, the vendor cost for

processing this metal could be approximately $0.20 per pound while the cost to decontaminate could increase by $0.10 per pound.
If only a fraction of the metal will required decontamination, add $0.10 per pound (worse case) to the average decontamination cost, find this cost line on Fig. 4, in Graphic I, or interpolate. The intersection of this line with the horizontal line for that fraction of the total metal weight to be processed is point P-3A. Draw a vertical line to the decontamination cost, C4Ra. Enter this cost in Table I, under the Recycle column, Decontamination Cost.
For the remaining fraction of the metal, draw a horizontal line from the weight of this metal to a unit cost line representing the cost for surveying and releasing only. This is point P-3B. Draw a vertical line down to find the decontamination cost, C4Rb. Enter this cost in Table I, under the Recycle column, Survey Only Cost.
Fig. 5
Fig. 6
Fig. 7
Table I
Table II
Starting at the number of containers calculated from Fig. 2, enter Fig. 5, on Graphic 2, and draw a vertical line to the container cost. This is point P-4. Draw a horizontal line from P-4 to the vertical axis to determine the total container cost. Enter the cost, C2D, in Table I, under disposal.
Starting with the number of containers for disposal, 1D, and recycle, 1R, draw a vertical line to the graph(s) which represents the type of container to be used and the weight of metal to be shipped in each container in Fig. 6, on Graphic 2. A horizontal line to the vertical axis will determine the number of shipments required to ship the metal. These are points P-5D (disposal) and P-5R (recycle) and will be used later to determine transportation costs.
Again, starting with the number of containers identified in Fig. 2, enter Fig. 7, on Graphic 2, at that point and determine the total burial volume of the containers used for disposal. This is point P-6D, and will be used in Fig. 9, on Graphic 3, to determine disposal costs. The line on this graph for drums should be used in assessing the cost of disposing secondary waste generated as a result of metal decontamination activities. A line for SeaLand type containers is not included since they are normally not used for disposal.
Fig. 8
Fig. 9
To determine the transportation costs for disposal and recycle it is necessary to estimate the cost of a single shipment. This is primarily a function of the transportation distance.Generally speaking an average cost of $1.43 per mile or $0.04 per pound (including the weight of the container) can be used. Using the number of shipments for disposal, 5D, and the number of shipments for recycle, 5R, enter the figure at these points and draw a horizontal line to the appropriate per shipment cost. A vertical line to the horizontal axis will identify the transportation costs for disposal, C3D, and for recycle, C3R. Enter these costs in Table I for transportation. The recycle option contains a second transportation cost which is related to the cost of transporting secondary waste to a disposal site. This cost will be calculated later.

Using the burial volume for the disposal option from Fig. 7, identify the line in Fig. 9, on Graphic 3, which is closest to the unit burial cost, $/ft3, for the site where this material would be buried. Based on a specific burial cost, $/ft3, identify the total burial cost C6D, and enter this cost in Table I, in the Burial column.

Fig. 10

Fig. 11

These figures are used only for the recycle option when containers for the transportation of the metal to the decontamination facility are leased for the duration of the project. Figure 10, on Graphic 4, should be used when the transportation distance is short enough that the turnaround time for a shipment, including the time to fill the containers and load them on the truck is one week or less. Figure 11, also on Graphic 4, is based on a turnaround time of two weeks. Using the appropriate figure enter at the weight of metal to be handled and draw a horizontal line to the graph which represents the type of container to be rented and the weight of metal that would be shipped in each container. From this point draw a vertical line to the total rental cost, C2R. Enter this cost in Table I.

SECONDARY WASTE

The generation of secondary waste during decontamination is a consequence of the decontamination process. Some of this waste may be considered as generated by the vendor providing the service. In this case the vendor is responsible for material disposal and its cost will be reflected in the price of the service being provided. Other wastes, which can be identifiable as resulting from activities relating to metals from a DOE site, including disposable workers clothing, wipes, solidified chemical solutions, laboratory waste (wipes, swipes, etc.), and any metal that cannot be decontaminated, are the responsibility of the originating DOE site.

Once the quantity of secondary waste has been determined, Fig. 2 and Figs. 5 through 9 can be used to determine the costs associated with the purchase of burial containers, the transportation of the containers to a disposal, and the cost of disposal following the steps presented above. If the soft wastes (clothing, wipes, etc.) are to be packaged in boxes along with the metal scraps and pieces that can not be decontaminated, then the disposal costs for these materials can be determined as a percentage of the metal to be decontaminated, typically about 2% by volume. Follow through the graphs as previously instructed to obtain costs associated with containers (enter this cost in Table I as Item C2RD); the cost associated with transporting these materials (enter this cost in Table I as Item C3RD) and the cost associated with the disposal of this waste (enter in Table I as Item C5RD).

All cost data as tabulated in Table I can be summed and the totals compared.

SAMPLE CASE

A sample case has been developed using the following input:

Total weight of metal to be handled (recycled or disposed) is 2,000,000 pounds, or 1,000 tons.

The on site preparation cost is $100/ton for recycling and $70/ton for disposal due to more stringent acceptance criteria at the recycling facility than at the disposal facility.

In both cases B-25 boxes will be used for transportation of the material. Each box will be filled with 4,000 lbs of steel, equivalent to the use of a B-25 box as a strong tight container.

The purchase price of a B-25 box for disposal is $425. B-25 boxes are leased from the decontamination facility for transportation and the shipping distance is far enough that container turn around will take approximately two weeks.

Per shipment transportation costs were as follows:

To the decontamination facility - $500.

To the disposal facility - $1000.

Secondary waste from the decontamination facility back to the originating site and then to disposal - $1500.

30% (600,000 pounds) of the metal is believed to not require decontamination and will be surveyed for free release at a cost of $.20/lb. A decontamination cost of $0.76/lba has been assumed with an additional cost of $0.10/lb for a pre-survey to identify metal which does not require decontamination.

Disposal of the metal as waste will occur at a DOE facility at a cost of $20/ft3. Ten percent of the original weight of the metal will be disposed of as secondary waste and it will be disposed at the same site. Other secondary waste generation is assumed to be small (roughly 2% by volume) and not significant.

The results of this case are presented in Table I.

While the disposal option in this case is less expensive than recycling there are several factors that could easily change the economics. Some of these factors can be identified as follows:

1) Burial Cost: With a cost difference of roughly $282,000 and a difference in volume of material to be buried as waste of 45,000 ft3, an increase in burial cost of slightly more than $6/ft3 would make disposal more expensive. $20/ft3 for disposal is consistent with current charges at NTS. An increase to $26/ft3, which is significantly less than the current disposal cost at many of the other DOE sites such as Hanford, Savannah River, Idaho or Oak Ridge, would make recycle the less expensive alternative.

2) Decontamination Cost: Again, with a cost difference of $282,000, the decontamination cost would have to drop by approximately $0.20 per pound ($282,000/1,400,000 pounds; 70% of the 2,000,000 pounds requiring decontamination) for the recycling option to be more cost effective than disposal at a disposal cost of $20.00/ft3. Recycling would also be competitive with disposal (at $20/ft3) if only half of the metal required decontamination and the other half could be released after survey.

3) Use of SeaLand Type Containers: At 40,000 pounds net weight of contents, 2,000,000 pounds of metal can be transported to a decontamination vendors facility in only 50 shipments at one container per shipment compared to 500 B-25 boxes each containing 4,000 pound of metal. The use of B-25 boxes would also require 50 shipments of ten boxes each. However savings could be realized in lower rental costs for the container and lower packaging costs since less sizing of large pieces is required.

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ABSTRACT

BNFL has almost completed the decommissioning of a major nuclear enrichment facility in the UK - the Capenhurst Diffusion Plant. This massive facility, 1,200m long and 150m wide and housed under a single roof, consisted of a cascade of 4,800 'stage units' of various sizes connected by 1,800 km of process gas pipework.

Dismantling the plant yielded over 160,000 tonne of suspect surface-contaminated material. By the time the project is fully completed, around the middle of 1996, over 99.5% of the contaminated material will have been safely and cost-effectively treated such that it can be recycled for unrestricted use in a non-nuclear environment. The remaining material, as well as minimal quantities of secondary wastes arising from decontamination activities, will have been size-reduced and/or encapsulated to maximize the cost-effective use of the UK low level waste burial facility at Drigg.

Although a number of novel and specialized techniques have been developed for use in this project, maximum use of "off-the-shelf" equipment, customized as necessary, has resulted in total project costs below 100M; i.e. in the order of 600 per tonne of material processed.

The project has utilized a wide variety of size-reduction measures, including robotic plasma cutting, novel decontamination techniques, melting for high and low melting point metals, and incineration of non-metallic arisings. The project has also met the challenges posed by the need to accurately and cost effectively measure very low activity levels after decontamination in order to confirm that the UK criteria for unrestricted release of material into the external market have been met.

All activities have been performed without compromising environmental discharges, the safety of plant operators, or the general public. In particular the personnel involved with the exercise have received confirmed radiation exposure levels of less than 0.2 mSv/yr.

INTRODUCTION

The Capenhurst Diffusion Plant was built in the early 1950s at which time it was the largest industrial building in Europe under a single roof, (Fig. 1). Some 1,200m long and 150m wide, the plant consisted of a cascade of 4,800 "stage units" of various sizes connected by 1,800 km of process gas pipework up to 550 mm dia, with numerous valves and associated process services (Fig. 2).It was originally built to produce highly enriched uranium for defence purposes but in the early 1960s the high enrichment section was isolated, emptied and put under care and surveillance. The rest of the plant was then modified to produce low enriched uranium for civil use.

Fig. 1

Fig. 2

By 1982, URENCO centrifuge enrichment plants were operating successfully at Capenhurst and the no-longer economic Diffusion Plant was shut down. As has previously been reported (1), a decommissioning project was begun to deal with over 160,000 tonne of suspect surface contaminated metal, concrete and other potentially hazardous materials. The significant nuclides involved in the contamination were Uranium and its daughter products, together with Technetium 99 (Tc99) and Neptunium 237 (Np237) from re-enrichment of reactor-recycled uranium. By the end of 1995, less

than 1% of the total mass of materials arising from this project will have been consigned for burial as low level waste. Over 99% will have been successfully treated and recycled. Less than 725 tonnes will have been despatched for land burial.

DECOMMISSIONING PHILOSOPHY

The main aim of this project has been to recycle as much of the above material as possible for unrestricted re-use, while minimizing the impact on the environment and the dose commitment to the workforce.

It was recognized from the outset that a significant 'up-front' investment would be required to develop suitable techniques to achieve this aim at minimum cost. The first two years of the project were therefore focused on research and development into cost effective decommissioning techniques, looking outside the nuclear industry for off-the-shelf equipment which could be used to meet the project needs without 're-inventing the wheel'. Figure 3, which represents the culmination of this exercise, identifies the activities involved in decommissioning the Capenhurst Facilities in the form of a process flowchart. The R&D investment has also left BNFL with the expertise needed to design, procure, operate and maintain specialized decommissioning plant and equipment of this type.

Fig. 3

SPECIALIZED TECHNIQUES

Plant Pre-treatment

Before the Diffusion Plant was shutdown, exploratory test work provided radiological and criticality data for subsequent use. A fluorination process was then employed to convert solid deposits within the plant to volatile fluoride compounds which could be safely disposed of. With the plant at standstill, further local operations dealt with any significant remaining pockets of solid or gaseous contamination.

Plant Dismantling

The main dismantling task was to break down the process plant into manageable units, many of which were then stored pending the availability of size reduction and decontamination facilities. Economical methods for safe penetration and in-situ cutting were developed. Units were then removed, sealed and placed into managed outside storage of up to 7,000 tonne of material.

Much of the main buildings, with the cells which had originally housed the plant and ancillary structures such as eleven cooling towers, pumphouses and an electrical substation, were demolished, monitored and sold as clean scrap. Floor slabs were decontaminated, ground samples checked and confirmed to be free from contamination, and the area returned to greenfield status. Stage units, ranging from smaller 750 kg steel shells to aluminum fabrications weighing 7 tonne and occupying 27m3, around 200,000 lengths of piping. 3,500 tonne of electric motors and 800 tonne of process valves were removed and treated.

Size Reduction

When removed from the plant, most items were too large and complex to be monitored or decontaminated without cutting. For each type of item, a cutting plan specified how to dismantle it further and, for wet decontamination, how to fill the baskets used in the decontamination equipment so as to expose all surfaces to the cleaning liquors. Most cutting was repetitive and was therefore automated using dedicated equipment installed in disused areas of the Diffusion Plant. A wide

spectrum of cutting options was examined and the most appropriate selected for each case.
The following examples show some of the varied size reduction techniques used.

Robotic plasma cutting - The size and complexity of the 750 aluminum stage units posed particular problems. After studying all the available options it was decided to develop a facility based around two large industrial robots adapted for plasma arc cutting. A special ventilation system, which dealt with the large volumes of fume generated, and fully remote operation minimized operator exposure. This type of flexible automation is available for future work (Fig. 4).
Fig. 4

Remote gas cutting - 1,400 of the nickel plated steel stage units were cut up using semi-automatic gas cutting, again using special ventilation and remote operation.

Large bandsaw - 2,200 tonne of aluminum pipework up to 450 mm dia was satisfactorily size-reduced using a large capacity bandsaw with rollerbed feeding.

Tube trepanning and stripping - 2,500 heat exchangers contained a total of 98,000 uncontaminated cupronickel tubes sheathed with contaminated aluminum fins. Special equipment was developed to trepan the tubes out of the tube plates and then strip the clean tubes away from the fins without cross-contamination.

Routing - Where hot cutting is inappropriate, routing is an economical alternative, again needing special facilities to contain swarf and minimize operator intervention.

Decontamination

The success of the project depended crucially on development of a decontamination process to remove surface contamination from a wide variety of metals and surface textures, ranging from bright wrought aluminum plate to heavily rusted steel. As noted earlier, the full range of nuclides resulting from handling reactor recycled uranium had to be removed, including Tc99 and Np237, the most obstinate sources of contamination.
Following extensive laboratory and pilot plant investigation, a full scale wet decontamination plant was developed, and finally completed in 1989 (Fig. 5). As probably one of the largest decontamination facilities in the world, it remains in operation capable of processing up to 100 tonnes per week. Suitable size-reduced items are loaded into baskets (nominally 2m x 1m x 1.5m) and then automatically pass through successive stages of washing in specially formulated process liquors which entrain the nuclides for subsequent transfer into ion exchange resins. The resins, which represent relatively small and manageable volumes of active residues, and other solid and liquid wastes of acceptably low activity and toxicity are later encapsulated in a cement matrix.
Fig. 5

Melting

Melting of low-level active metals has been reported in France (2), Germany (3), Sweden (4) and Japan (5) amongst others. The main aims have been towards products restricted for use within the nuclear industry, or for volume reduction prior to burial. By contrast, the Capenhurst melting facility primarily aims to produce ingots which are below free-release levels for unrestricted re-use. This is achieved by first decontaminating the feedstock as described above, and then using melting to produce

metals in a form such that they can easily be monitored to confirm the level of contamination is at or below free-release levels. Melting produces a homogeneous ingot, samples from which can be monitored to confirm for free release or sentenced for burial. In the latter case, there is still a value in reducing burial volume and in fixing activity within the ingot.

Following pilot trials, a full scale melting facility was developed and installed (Fig. 6) to process a range of contaminated metals, including items with non-metallic additions. Aluminum, lead, copper, bronze, cast iron, steel and nickel have all been melted, and it has been possible to separate lower melting point metals from higher melting point metals. Non-metallics have been incinerated with full environmental protection.

Fig. 6

The facility, which has a throughput up to 150 tonne per week, is a controlled active area with full radiological engineered safety systems. These include a highly sophisticated fume cupboard and treatment system which removes all particulate and volatile radioactive and toxic substances in strict compliance with Environmental Protection Regulations. A facility control system, including dual redundant programmable controllers, ensures safe shutdown of the facility on detection of any excursion from defined limits, thus minimizing the risk of personnel dose uptake and preventing undesirable environmental discharges.

Special attention is given to determination of the radioactive content of the ingots which are at levels close to those occurring naturally. Radiochemical determination involves dissolution of samples of a few grams weight, extraction of the nuclides and counting by liquid scintillation. Detection limits are 0.1 Bq/g, but the method is relatively expensive. Delayed neutron counting is cheaper, but requires a 'fingerprint' of the isotopic ratios - not a consistent parameter within the Diffusion Plant. The more costly techniques have been used to calibrate a low level alpha/beta counter which is used to determine the activity within disc specimens (60 mm dia and 6 mm thick). Due to the low background level of this counter, less than one count per minute, and the ability to count up to ten discs simultaneously, count times of about one hour gave the required accuracy, and large numbers of samples can be processed at low cost. It is necessary to exert close control over sampling and counting techniques to avoid cross-contamination.

The initial campaign treated around 4,000 tonne of metals which were decontaminated to relatively low levels of contamination. They still could not be released, however, because their complex geometry prevented full monitoring. Examples include small bore pipework never exposed to process gases, closed voids within weldments, and re-entrant surfaces generally. Following successful demonstration of the containment of contamination within the melting facilities, a justification has now been made to extend the capability of the facilities to process metals contaminated at higher levels. Data is also available on the degree of partitioning achieved during the melting process.

Encapsulation of Waste

The small volume of waste destined for land burial comprises both solid metals and a variety of less easily packaged materials. These included secondary waste such as evaporated process liquors, ion exchange resins, melting drosses and slags, filter media, swarf and floor sweepings. A skid mounted, transportable encapsulation plant has been designed by BNFL

and is in operation at Capenhurst. This plant immobilizes the wastes into an ideal non-leachable form for low level waste disposal at Drigg using a specially formulated cement encapsulant in standard 200 liter drums.

RECYCLED MATERIALS AND WASTE STREAMS

Solid Arisings

Once plant contamination has been characterized, it is essential to define the waste streams for which process routes will have to be established, and with which all other decommissioning decisions will have to be consistent. It is then necessary to identify the disposal options for the original contaminated material as well as any secondary waste streams arising from the process routes utilized.

Unlike most other countries, the UK has clearly established criteria for unrestricted release of material into the external market. The criteria exempt substantially insoluble solids with specific activity below 370 Bq/kg of any nuclide, with additional exemption for uranic activity below 11,100 Bq/kg. Whilst international rationalization of clearance levels may ultimately lead to a more nuclide-specific set of release criteria, with higher limits for soft beta emitters such as Tc99, the timescale for such legislation is likely to be extended and the operations at Capenhurst needed to be carried out in line with the more restrictive application of the current regulations. The development of cost-effective techniques to confirm the criteria are met has also been a very important consideration for melted ingots.

The only available disposal option within the UK for materials exceeding the criteria for unrestricted release is land burial at Drigg, the BNFL Low Level Waste Disposal Site near Sellafield. Burial of large volumes of contaminated material would have imposed massive cost penalties on the project, as well as occupying an unacceptable and unnecessary proportion of the valuable low level waste site capacity. Effective decontamination and sentencing of the maximum amount of material to free release levels was therefore highlighted as an essential part of the project.

Against this background solid arisings from the project have been categorized as follows:

Clean scrap - materials which can be proved by monitoring to be suitable for free release.

Metals for decontamination - metals which can be decontaminated and subsequently can be proved by monitoring to be suitable for free release.

Metals for decontamination and subsequent melting - metals which can be decontaminated but because of complex shapes/surfaces, cannot be economically or effectively monitored but which would be suitable for melting and recasting as ingots. After melting, the material would then be in a form capable of being monitored to confirm its suitability for free release.

Materials not suitable for free release - Material (the remaining small quantities of material including some secondary waste arisings) which is not suitable for free release and is therefore consigned for land burial.

Other Arisings

The main source of liquid waste is from the decontamination process and is subject to discharge authorizations covering radionuclides and toxic metals which limit discharge into local water courses to below 100 Bq/liter/day and to below 1.8 ppm total metals. Other liquid wastes up to 1,000 Bq/l are acceptable to the Regulators for landfill disposal.

Gaseous discharges are limited by the discharge authorization for the whole Capenhurst site to less than 50 MBq annually. Actual emissions are

less than 1MBq annually. A comprehensive routine sampling and analysis program exists to monitor and confirm environmental discharges. Compliance is further supported by regular audits by relevant Government Departments.

PROJECT SAFETY

An overriding concern throughout the project is to minimize risks of radiation exposure to decommissioning operators, personnel on site and the public generally. Alpha in air monitoring, with an alarm level set at 3 Bq/m3, is used extensively throughout the facility as is routine surface monitoring. A criticality detection system is installed, and strict criticality control procedures applied at every stage. Specialized workshops were constructed using redundant cells from the Diffusion Plant itself for the various operations with dedicated and effective ventilation.

Operators use personal air samplers, respiratory protection, and full protective clothing as appropriate, (Fig. 3) and the BNFL personnel radiation protection program has confirmed exposure levels at less than 0.2 mSv/yr. Full account is also taken of non-radiological hazards, particularly those associated with other toxic or hazardous materials removed during dismantling.

COST DRIVERS AND SAVINGS

The total cost of the project on a historical cost basis is less than 100M - in the order of 600 per tonne of material. About 20% of that total (representing a major investment), was applied to decommissioning R&D. Major costs drivers for BNFL included external purchases of hardware, costs of burial at Drigg, labor and charges for health physics and radio-analysis. The decommissioning of the Diffusion Plant will be completed within budget.

The following features all contributed to what will be a major achievement:

Early investment in identifying and categorizing the various wastes was vital in laying the foundation for well focused investment in specific decommissioning techniques.

Most importantly, the efficiency of wet decontamination avoided burial costs, and generated income from sales of metals for unrestricted re-use.

Where the shape or surface structure of cleaned metals did not allow cost-effective monitoring, melting, supported by cost effective techniques to confirm that the UK criteria for unrestricted release of material into the external market have been met, proved an ideal adjunct to facilitate further recycling.

Value engineering was used to good advantage through the project. Finally, the value of a well motivated, well trained and enthusiastic workforce contributed substantially to the successful outcome.

CONCLUSIONS

The philosophy and techniques described above are likely to be relevant to a wide range of nuclear processing plants throughout the world, particularly larger plants with low levels of primarily surface contamination.

The Diffusion Plant decommissioning program will be successfully completed by the middle of 1996.

Of the 160,000 tonne of metals and concrete comprising the structure and contents of the plant, over 99.5% will have been recycled for unrestricted re-use as clean material.

The project demonstrates how the Capenhurst Diffusion Plant, a major nuclear processing plant facility, has been safely and cost-effectively decommissioned.
The experience gained from this project has resulted in methodologies being developed to minimize the amounts of nuclear and other hazardous wastes arising from decommissioning activities and demonstrated cost effective techniques to confirm that the UK criteria for unrestricted release of material into the external market have been met.
The techniques developed can be applied to other nuclear or non-nuclear processes.

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RECYCLING OF MATERIAL FROM NUCLEAR INSTALLATIONS ACTIVITIES OF THE EUROPEAN COMMISSION AND THE OECD NUCLEAR ENERGY AGENCY

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ABSTRACT

The recycling of contaminated material could be a major means of reducing the quantities of low level waste produced by the decommissioning of nuclear facilities. Both the European Commission (EC) and the OECD Nuclear Energy Agency (NEA) have been working on this subject during the last few years.
The European Commission has sponsored research and development on the reuse and recycling of low level solid waste materials originating from the decommissioning of nuclear installations, within the framework of three sequential five year research programs, the first of which started in 1979.
A series of R & D projects were carried out on the reuse and recycling of various materials originating from the refurbishing/dismantling (steel, copper, aluminum and concrete). Particular progress was made for the recycling of LLW steel, using the melt technique not only for volume

reduction but also for the production of specific items for the nuclear industry.

The EC has also set up, under the terms of Article 31 of the Euratom Treaty, a group of experts to examine the implications for radiation protection of recycled materials from nuclear installations with a view to establish appropriate recommendations for clearance levels. First recommendations issued in 1988 (Radiation Protection Series No. 43) relate to the recycling of radioactive steel. Further recommendations for metals, including copper and aluminum are under preparation. A similar document for concrete is intended. A task group set up by the OECD/Nuclear Energy Agency's Co-operative Programme on Decommissioning has been studying recycling during the last three years. It has noted that considerable material is being released from regulatory control today, but mostly on a case-by-case basis, with widely varying release criteria in different countries.

Both the EC recommendations for release criteria and those proposed by the International Atomic Energy Agency are essentially based on radiation protection considerations. The NEA task group has attempted to view recycling in a broader context, evaluating both radiological and non-radiological detriments as well as social, economic and environmental aspects. This is seen to be fully in harmony with the ICRP concept that the justification of a practice should take into account the total detriment and not only the radiation detriment.

This approach has led to the consideration of a tiered system of clearance levels, allowing a more rational material management, where release levels would depend on whether the materials were being released directly, sent for melting, melted in controlled facilities before release for remelting or reused for specific purposes etc.

Both the EC and the NEA see the need for internationally agreed clearance standards, specially when much larger quantities of material become candidates for release, when today's operating reactors come to the end of their useful lifetimes.

INTRODUCTION

Recycling of contaminated material has been identified as a major means of reducing the volumes of waste for ultimate disposal arising from the decommissioning of nuclear facilities. The incentives lie both in the huge quantities of contaminated material produced over a relatively short period of time as well as the steeply escalating costs of final disposal space in most countries.

The European Commission (EC) has, in a series of three five year programs, sponsored R & D projects in this area, through shared costs contracts with various organizations and companies in the European Union. These programs were carried out between 1979 and 1994. The EC has also produced recommendations (9) for recycling radioactively contaminated metals. These criteria are aimed mainly to minimize the radiological risks to potential users of the recycled material.

The OECD Nuclear Energy Agency's (NEA) Co-operative Program on Decommissioning set up, in 1992, a task group to study recycling and reuse of material from nuclear facilities. The Co-operative Program covers 30 decommissioning projects from 10 countries and thus represents the largest group of potential users of the criteria being drawn up in the area of recycling. Understandably, the NEA task group has treated recycling in a broader context than exclusively radiological aspects and

has studied it also from other viewpoints such as costs and the effects on the environment.

In the following is given a brief overview of the most relevant of the recycling projects sponsored by the European Commission as well as its recommendations for clearance criteria. The paper then goes on to describe some of the results of the work of the NEA task group and gives examples of case histories of recycling projects including cost comparisons with disposal as radioactive waste. The paper concludes with a brief outlook over future work being proposed in this area.

RECYCLING ACTIVITIES OF THE EUROPEAN COMMISSION

Recommendations for Clearance Criteria for Recycling of Metals (9)

Legal basis

The legal base for any action linked to radiation protection is the EURATOM Treaty set up in 1957. As far as radioactive material like metal scrap is concerned, the Treaty considers aspects of health protection, safety, and research and development. It gives priority to free access of goods to a common nuclear market and aims at establishing far-reaching Community competencies in a number of areas; the most important one concerning clearance levels for metals being the regulatory competence for setting up basic safety standards in radiation protection. These standards provide for common criteria which would have to be implemented in national legislation.

Technical and Scientific Background

Support for developing clearance levels was provided through the specific research and development programs on "Radioactive Waste Management and Disposal" and "Decommissioning of Nuclear Installations". Furthermore, in the "Plan of Action in the Field of Radioactive Waste Management", an instrument existed which allowed harmonization of practices and development of recommendations in semestrial meetings between Member States and the European Commission.

First contracts were concluded in 1983 with the UK National Radiological Protection Board and the French Commissariat l'Energie Atomique, to develop a methodology for evaluating radiological consequences of the management of very low-level waste, particularly solid waste from dismantling of nuclear power plants. The analysis of waste categories showed significant quantities of only slightly contaminated steel and concrete.

A range of activity values and scenarios for unrestricted release and decay times up to 100 years before disposal was assumed, and radiological consequences were found to be rather small. It was demonstrated that other factors than radiological consequences, such as costs for transport, treatment and measurements, and regulatory and social factors, for example acceptance by the public, may well be more important. The methodology developed in both contracts were later enlarged to recommendations on steel scrap recycling (9).

R & D work on release and recycling of contaminated steel and concrete has been an important part of the three five-year decommissioning programs carried out since 1979.From the results of these programs (1,3,4,5,6,7,8), input parameters could be derived, which were or will be used in scenarios calculations for proposing exemption and clearance levels.

Recommendations for Release Criteria

A Group of Experts had been set up under the terms of Article 31 of the Euratom Treaty, which convened a Working Party in 1984 to establish

radiological protection criteria for the recycling of materials from nuclear installations. The Experts prepared recommendations for criteria for recycling steel which were issued in 1988 (9). In the light of further information on recycling coming from later studies as well as due to revised radiation protection criteria, the Experts expanded and updated the 1988 recommendations to consider:

Criteria for other metals (steel alloys, aluminum and aluminum alloys, copper and copper alloys),

Criteria for surface contamination specific to recycling,

Covering all nuclear fuel cycle plants including installations for uranium enrichment, fuel production, power generation and reprocessing (but excluding mining or milling operations as well as final repositories).

A draft document has been prepared in 1994 suggesting nuclide specific clearance criteria for slightly radioactive materials of the types listed above. The radiation protection principles used by the Experts has been the 10 mSv per year individual and the 1 manSv per year collective dose per practice as recommended by the IAEA Safety Series No 89 (10). Separate values are given for equipment, components or tools which will be reused directly (only surface specific limits, Bq/cm2) and for material which will be recycled by melting (both surface specific, Bq/cm2, and mass specific, Bq/g). In the case of many nuclides, the surface specific clearance levels for material expected to be melted is a factor 10 higher than those for direct reuse.

Research and Development in the field of Recycling

Numerous studies, investigations, developments and technical demonstrations have been carried out mostly based on shared-cost contracts with a series of EU organizations and enterprises, such as for example TV Bayern, NRPB, KEMA, Siemens, Siempelkamp and CEA (1-8). The objectives of these activities have been waste volume reduction, more accurate activity measuring of the waste for further conditioning, free or restricted release. Melting of carbon and stainless steel components from nuclear installations for reuse is now state-of-the-art in Germany (Siempelkamp), where industrial conditioning of components originating from refurbishing or decommissioning of nuclear installations (e.g. KWO, KRB-A) is carried out in the authorized melt facility CARLA. Investigations at laboratory and pilot scale are being carried out with Siemens and Siempelkamp to investigate the decontamination effect of melting a- and H3-contaminated steel waste. Some of these projects are described below.

Melting of Ferritic Steel arising from the Dismantling of the G2/G3 Reactors, Marcoule (6)

The objectives were to condition by melting a large quantity of ferritic steel having a specific contamination in the order of 20-40 Bq/cm2 in a 15 t electric arc furnace and to determine the U-distribution in the molten product during a large-scale melt campaign of approx. 100 t (8 melts of 12.5 t each) of mild steel contaminated with uranium originating from the dismantling of the Pierrelatte reprocessing facility (Usines basse et moyenne). The amount of secondary waste arising was considered in particular.

Over 4300 tons of ferritic steel pipes were cast in 25 kg ingots, from which nearly 200 tons of total secondary waste were obtained. The ingots (25 kg) are stored on site, because the activity level for free release has not yet been decided in France.

Radioprotection measurements carried out by CEA's health physics specialists during the melts showed that no specific problems appeared, neither for the people working around, nor for the buildings. Detailed interpretation and results will be given in the final report to be published in the first half of 1996.

Melting of Tritium-containing Steel (8)

Important quantities of steel cannot be released from radiological surveillance because of their tritium contamination. The work consisted in trapping the tritium released from steel during heating and melting in a specially adapted exhaust system of the Siempelkamp CARLA melt facility.

During the melt process, the tritium passes into the exhaust gases where it was quantified in a bypass measuring section. Basically, it was demonstrated that steel contaminated with tritium can successfully be decontaminated. The predominant part of the tritium, more than 96%, is released as HTO or T2O in the course of the melt. The main release already occurs at temperatures below 500oC. The residual activity of the castings and the slag is negligible. To retain the tritium bonded to the air humidity, drying of the intake air is necessary.

Melting of Alpha-contaminated Steel (6)

The main objectives are to improve the decontamination effect by pre-oxidation process and the new slag former, as well as the radiochemical analyses of the behavior of Pu, Th and U-isotopes during melting, in particular the U-235 distribution in the molten ingot.

Laboratory melts aiming at the identification of the most suitable crucible material and slag former were followed by large-scale melts with subsequent detailed analysis of the prevailing alpha-distribution in and between steel, slag and filter dust (about 100 t of U and Th-contaminated material).

Decontamination factors of 44, 46 and 69 were obtained respectively.

Remelting existing steel ingots (+ 5 t) with subsequent radiological characterization was difficult due to the hardness of available ingots (no sampling possible). The use of uranium glass as an additive to the melt, to restore the natural isotopic ratio, gave good results (no dust and DF of 71/U, 192/Pu and 886/Th).

Using concrete as the new slag former gave no significant improvement.

Detailed interpretation and results will be given in a final report to be published during the first half of 1996.

Other Projects (6,7)

In addition to the above, there have been R & D projects carried out in the following areas:

- Recycling of aluminum and copper by melting,
- Doses due to the reuse or recycling of slightly radioactive steel,
- Industrial recycling and reuse in Germany,
- The onion melt technique for waste minimization and reuse of LLW steel in container fabrication,
- Volume reduction of contaminated/activated concrete,
- Concrete recycling in France,
- Radiological aspects of recycling concrete debris in Germany,
- Recycling of contaminated steel as reinforcement in concrete.

ACTIVITIES OF THE OECD/NEA TASK GROUP ON RECYCLING AND REUSE

The Task Group was set up by the Co-operative Program of the OECD Nuclear Energy Agency in 1992. The scope of the Co-operative Program is basically to exchange technical and other information between on-going

decommissioning projects. Since its start in 1985, it has grown to cover 30 projects from 10 countries, including a wide selection of reactors as well as fuel facilities. The Program has thus become the international forum for nuclear decommissioning projects. As such, it is deeply interested in all possible means of reducing the quantity of waste produced by such decommissioning. The recycling of such material, instead of disposing it as low level waste, was identified as a major method of achieving this aim.

The objectives of the Task Group were to:

make a survey over material currently being released from nuclear installations,

review the on-going work on proposals for clearance levels from the perspective of the potential users of such criteria,

make state-of-the-art surveys over the technologies necessary for recycling,

compare recycling/reuse of material with the alternative approach of disposal/replace from the viewpoints of

- total health risks (not just radiological risks)
- environmental impact
- socio-economic aspects
- costs.

During the three-year period covered by this paper, the work of the Task Group has been generally focused on metals. Its findings and conclusions to date are summarized in a report, currently being reviewed by the Nuclear Energy Agency's Radioactive Waste Management Committee and Committee for Radiation Protection and Public Health, prior to publication later this year. In the following is briefly summarized some of the findings of the Task Group.

Current Practice

One of the first activities of the Task Group was to map the current policies regarding the release of materials from nuclear facilities in various countries. Information was solicited from participating projects regarding the quantities and types of material released as well as the conditions governing their release. Information was received regarding 24 projects as well as the existing regulations in 6 countries. The projects included reactors, fuel facilities, uranium milling plants and isotope production facilities.

The survey revealed that over 360 000 t of material had been released between 1979 and the early 90's. The released material included carbon and stainless steels, lead, aluminum, concrete, soil and gravel. A variety of clearance alternatives had been adopted ranging from release for unrestricted reuse or disposal for about two-thirds of the material to restricted reuse within the nuclear industry. Most of the material was released after consideration on a case-by-case basis, even in countries where there are national regulations in place for the unrestricted release of material from nuclear facilities. An overview of the material released and the types of release practices is given in Table I (11).

Table I

A scrutiny of the released criteria used in the various projects and in the national regulations shows a wide range of values. For instance, Sweden has a beta-gamma surface contamination limit of 4 Bq/cm2 for unrestricted release, while in neighboring Finland, the corresponding limit is 0.4 Bq/cm2. The Finnish mass specific release limit is 1 Bq/g compared to the corresponding Swedish limit of 0.1 Bq/g. There are also

differences in measurement and documentation practices as well as in quality assurance requirements.

Health Risks/Environmental Impact

The International Atomic Energy Agency (IAEA) and the European Commission (EC) have both prepared recommendations for criteria for releasing material from regulatory control. Both proposals are based almost entirely on radiation protection and regulatory considerations.

The NEA Task Group represents projects which are the potential end users of such criteria. Understandably, the Task Group has studied clearance in a broader context, basing its approach on the recommendation of the International Commission on Radiological Protection (ICRP) regarding the justification for the adoption of a new practice involving exposure to radiation: "..... The radiation detriment may only be one of several aspects that comprise the total detriment associated with a particular practice. Consequently, the justification of a practice goes far beyond the scope of radiological protection.....".

The Task Group has therefore examined the justification for release of radioactive material by considering not only the risks from radiation, but also major non-radiological socio-economic, environmental and health effects. In this context, a "tiered" system of release criteria, with defined end uses for each tier, has been used as the basis for evaluating the various detriments and thus comparing recycling with the alternative approach of disposal of material as waste (12). The "tiered" approach is described in more detail in the section, "A Tiered Approach to Release Criteria".

Some of the results of these studies are summarized in the following Tables II and III.

Table II

Table III

Melting

The melting of contaminated metallic components has the advantages of

- a "decontamination" effect on volatile nuclides such as Cs 137 as well as on Uranium and other oxides,
- simplifying the radioactive characterization of components with complex geometries.

Four plants have been operating on an industrial scale during the last few years: the CARLA plant at Siempelkamp, Germany; the INFANTE plant at Marcoule, France; the SEG plant at Oak Ridge, USA; and the Studsvik Melting Facility in Sweden. At the decommissioning of the Capenhurst Diffusion Plant, U.K., a melting plant is being used to recycle large quantities of metals. A new plant is being constructed at Oak Ridge, USA.

Decontamination

For decontaminating equipment to recycling (release) levels, the methods used must have high decontamination efficiencies. At the same time, it must be possible to condition the secondary waste satisfactorily. Two methods satisfying these requirements have been demonstrated on a fairly large scale.

The Swedish SODP chemical decontamination method for PWR systems was used to decontaminate steam generators from the 80 MW gesta PHWR to near release levels. The steam generators were segmented and melted after decontamination.

A dry abrasive blasting method has been developed to recycle about 1 600 t of steel from the Eurochemic Reprocessing Facility. The method was first tested on a semi-industrial scale on a batch of 32 t.

Measurement

The currently proposed release levels are very close to background levels. This places very stringent requirements on the accuracy and reliability of the equipment and techniques used in the qualifying measurements for release. A large number of reports testify that it is possible to satisfy such requirements but most of the work in this area hitherto been on a laboratory scale. The practical problems of measuring very large quantities of material at extremely low activity levels and at a reasonable cost should not be underestimated.

The NEA Task Group has analyzed the case histories of 18 projects that used varying methods of recycling material from decommissioning projects. The projects are listed below in Table IV. The costs of some of these projects are compared with the respective estimated costs if the material in question had been sent for disposal as low level waste. The savings achieved by recycling are illustrated on Fig. 1.

Table IV

Fig. 1

A Tiered Approach to Release Criteria

As mentioned earlier, the evaluations of the Task Group are based on a "tiered" approach, with the highest tier (A) corresponding to unconditional release and each lower tier applicable to specific (conditional) release conditions. In the system used in the group's evaluations, the next highest tier (B) consists of material that is melted in a radiologically regulated plant and then is conditionally released for subsequent remelting and fabrication in non-nuclear commercial facilities. The products could then be used without radiological restrictions. The next tier (C) is material similar to that in tier B, except that the material is released for a specific initial industrial use. Material in the lowest tier (D) is recycled within the nuclear industry.

It should be noted that this is only one possible manner of applying the tier concept. One which would reflect the approach of the EC as well as practices in several countries is shown in Fig. 2.

Fig. 2

Here the tiers A and B correspond respectively to the EC criteria for direct release and for recycling by melting at a commercial foundry. The surface limits for recycling are more relaxed than those for direct reuse, because of the inherent advantages of melting. Tier C would cover waste metals melted at a controlled facility and then sent for remelting at a non-nuclear foundry. The release levels for the ingots leaving the controlled area could be expected to be higher than those for tier B, for equivalent levels of radiological safety. Tier D would consist of similar material as tier C but released for specified initial industrial use, while tier E material would be recycled within the regulated environment. In Fig. 2, tier C is similar to the management of contaminated metals at Studsvik in Sweden, while tier E reflects the activities at Siempelkamp, Marcoule and Oak Ridge.

OUTLOOK FOR THE FUTURE

As the technical problems of melting and recycling of low-level radioactive metal, if any, seem to be mostly at an industrial level, future activities in the EU may focus (besides some more specific R & D work, e.g. for alpha and tritium-contaminated materials, concrete recycling) on the development of common clearance levels and reliable standard procedures for their application, not only to contribute to the

establishment of harmonized regulatory requirements at Community level, but also with respect to future EU Member States and Eastern European countries which have vast quantities of radioactive material to deal with, such as the Chernobyl area. For this, co-operation between competent experts from the EU and the USA would be extremely useful. The NEA Task Group will continue in its efforts to place recycling in a perspective where it can be compared to other industrial activities. It will continue to work with other bodies to establish an international consensus concerning both the exemption and clearance of radioactive materials. In the development of regulatory criteria for recycling, the Task Group will work for a "global" optimization and not just optimization from a radiological viewpoint. This would lead to a more rational management of material in the nuclear industry.

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DESIGN AND CONSTRUCTION OF A STATE OF THE ART METALS DECONTAMINATION AND MELT FACILITY

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ABSTRACT

Manufacturing Sciences Corporation, in partnership with British Nuclear Fuels, has designed and built what is recognized in the industry to be a state-of-the-art metals decontamination and melt facility. The partners have combined talents and experiences of their respective businesses to design and construct a facility that directly challenges the problems associated with existing approaches to metal recycle. This they achieved from concept to reality in less than 15 months.

This paper addresses the central design principles of minimizing worker exposure, industrial safety, Just-in-Time philosophy and process control. It will describe the comprehensive computerized traceability system, the extensive automated/remote handling, the main line decontamination processes, and the technologically advanced melt facility. Further, the approach to project/construction management will be discussed to demonstrate the value of extending the partners' ability as team players to include subcontract design and construction groups.

The purpose of this paper is to present a successful approach to the design and construction of a facility, with new and novel features, to meet an extremely aggressive schedule.

INTRODUCTION

Since the dawn of the nuclear age, maintenance and equipment replacement at power stations and other nuclear facilities have given rise to the need to dispose of contaminated scrap metals and equipment. These contaminated materials have been stored, buried, or decontaminated and released. Over time, responsible waste management become recognized as a significant cost of doing business in the nuclear industry. Beginning about 15 years ago, an industry emerged to provide low-level radioactive waste management services to the nuclear industry. Decontamination and release of those contaminated metals and equipment that could easily be cleaned and surveyed, and volume reduction and burial of those that couldn't, became a central offering of this embryonic industry. Early methods were manual, slow, and of low throughput. This decade has brought the end of the cold war and the prospect of decommissioning of U.S. Department of Energy (DOE) sites formerly used to produce nuclear materials and nuclear weapons. The DOE cleanup is expected to generate vast quantities of contaminated scrap metals requiring processing capacities far greater than those existing in the industry today. Recognizing that a safe, efficient and cost effective contaminated metals recycle capability to meet the expanded capacity and more stringent regulatory requirements of the future, two companies, Manufacturing Sciences Corporation and British Nuclear Fuels joined forces to design and build a modern contaminated metals recycle facility in Oak Ridge, Tennessee. This paper reviews the process of design and construction that resulted in a modern contaminated metals recycle facility.

DESIGN PHILOSOPHY

To address the issue of demonstrable capability through private sector investment on a scale which could address the potential needs of the customer, it was necessary to leap-frog existing technology used for recycle of radioactively contaminated scrap metal. Basic principles were established against which the facility design was developed:

- Zero emission objective
- Focus on industrial safety
- Just-in-Time philosophy
- Process control
- Recognition of local environment

To achieve a firm focus on the basic principles of minimizing worker exposure, high standards of worker safety and best practices in manufacturing, the fundamental obstacles of material handling and tracking had to be overcome. The varying form, type and configuration of materials to be handled drives processors away from automation and remote operation. The partners wrestled with this problem, finding this to be the major limiting factor on facility throughput. This issue required a paradigm shift to question how can the material be standardized. The solution identified was to transfer material in standard containers; 20 foot sea vans B25's, special design boxes and baskets etc. Handling systems for transferring material and processing material could then be designed around a standard container, components/material could then be tracked by virtue of the container in which it is housed. This shift in philosophy allowed the partners to adopt a number of special features for the facility which include:

- Automated undercover storage of incoming contaminated scrap metal.
- Remotely operated material handling.
- Semi-automated mechanical and chemical decontamination processes.
- Vacuum induction melting.
- Automated radiometric survey of material for free release.
- Fully computerized material tracking systems.

These features allowed development of a facility which in 115,000 square feet provides a full decontamination and melt capability for up to 10,000 tons of contaminated metals per year. The process is outlined in Fig. 1 and reflected in the facility layout in Fig. 2. Each stage of the process is discussed to highlight the significance of the change in philosophy to create an integrated facility.

Fig. 1

Fig. 2

PROCESS

Incoming Storage (Fig. 3)

Standardizing, where possible on 20 foot freight containers, allows automated storage and inventory control. Using a fully automated laser controlled crane for materials handling allows containers to be stacked two-high in a close packed storage array. Containers can be delivered to and released from the storage array without operator involvement, minimizing the associated operator dose uptake. This approach to storage, with a strict storage management regime, ensures containers with high activity levels can be shielded by containers with low activity levels, further reducing operator dose levels. This storage system allows the footprint of the storage area to be minimized allowing fully covered storage. On receipt containers are coded into the database of the Management Information System (MIS) using bar code scanners, and the manifest data is entered by the operator. All material movements from

this point and throughout the process, are electronically recorded to provide full on-line material tracking and control.

Material Preparation

The aim of standardizing processes is critical to achieving high volume optimized throughput but, to deal with the array of material and the differing types of customer projects, it has been necessary to implement two separate routes through to processing. The standard processing route involves tipping the contents from a container then remotely sorting and sectioning metals prior to decontamination. A special projects route allows for large component disassembly and sectioning for further processing and equipment decontamination and refurbishment prior to sale or return to the customer.

Unload, Sort and Section Area (Fig. 4)

This is an unmanned area which is used to routinely empty freight containers and to sort and section the discharged metals in preparation for later decontamination processing. The operator working in an enclosed control room operates a hydraulic lifting table and an industrial robot. The robot is used to distribute metals for sectioning to various saws and shears. Metals are tipped from the container and transferred using robust vibratory conveyors to the grit blasting equipment at the end of the process line or to the chemical processing area depending upon the desired method of cleaning. This approach reduces the workforce's exposure to contaminated materials and eliminates a primary source of industrial safety hazards.

All material leaving this area is loaded into standard material handling boxes and each box is weighed and coded into the MIS system using bar code scanning. This allows all material to be tracked back to the original freight container.

Special Projects Area (Fig. 5)

This area is dedicated to a variety of projects which are not suitable for processing in the automated area. These projects include dismantling of large components as well as refurbishing of a variety of valuable equipment. This area includes fully enclosed and locally ventilated painting, welding, grit blasting and sectioning. Relocatable enclosures and exhaust systems are used to allow reconfiguration of this area to accommodate the varying requirements of each task. Equipment that is broken down in this area for processing elsewhere is put in a standard material handling box, weighed and coded into the MIS using bar code scanning prior to transfer.

Material Handling

Materials are moved around the facility in boxes on two automated monorail systems. One monorail is dedicated to materials which have not been decontaminated, the other to materials which have been through the decontamination process. Each box loaded or removed from the monorail is bar code scanned to record the transfer in the MIS. The boxes used on the monorail have been specifically designed as a standard material handling box; these boxes are the only containers used to move material between processing areas. This approach to material handling significantly reduces operator involvement in material transfers, and minimizes the material stored in work areas, further reducing workforce dose uptake and exposure to industrial safety hazards. Additionally, it is a major step that introduces Just-in-Time production control techniques. Material is only delivered to an operator on request and material storage is limited

to that held on the monorail. This enforces a move away from the traditional problem of material stockpiles.

Mechanical Decontamination

The primary method of mechanical decontamination is grit blasting. The conveyors from the unload and sort area discharge directly into a fully enclosed batch process tumble blaster. This allows a process which has predominantly been carried out in manual grit blast booths to be carried out remotely, further reducing operator dose uptake. Additionally, using this approach, once experience is gained in material sizing, grit media and cycle times, process control techniques will be used to optimize the process and minimize wastes.

Chemical Decontamination (Fig. 6)

Material requiring chemical decontamination enters the area via the monorail system or the conveyor from the unload sort area. Material is loaded into baskets which are dipped in a series of chemical tanks and a spray washer via a monorail hoist system. The monorail system for this area and the chemical decontamination process are controlled automatically in a Programmable Logic Controller (plc). The operator bar code scans the basket and selects the appropriate chemical decontamination cycle for the material. On completion, the basket is transferred to a tipper station where the material is inspected and tipped into a box and bar code scanned prior to transfer to the survey area or foundry via the monorail system.

The tanks are provided with containments and extraction to remove fumes. The fumes are extracted through a scrubber system to the HEPA filters. The process equipment design ensures that the facility, with appropriate additions, can use any commercially available chemical decontamination processes. As with other areas previously discussed, the automated handling and operation further reduces workforce exposure and industrial safety hazards. The extent of automation, material flow control and process control provides an extremely efficient decontamination approach, allowing for process optimization and minimum waste.

Waste Handling (Fig. 7)

Wastes generated from processing operations are delivered to the area in boxes via the monorail system. Material is sorted into compactible and non-compactible wastes. Compactible wastes are processed via a shredder and drum compactor. Non-compactible wastes and wastes requiring stabilization are grouted in boxes. Drums and boxes filled in this area are dispatched to the special projects area for painting and labeling prior to shipping. As in all other areas using the MIS system, material delivered to this area can be traced back to the original container, allowing accurate allocation of wastes to each contract.

Foundry (Fig. 8)

Metals are delivered to this area in boxes by the monorail system. The contents of the boxes are segregated into batches for melting. All material used to make up a melt is recorded in the MIS to provide full traceability. Melting is carried out using a Vacuum Induction Degas and Pour (VIDP) furnace which produces high quality steel ingots. All scrap is cleaned prior to melting to ensure practically no slag is produced from the process. The furnace is fed on-line through a vacuum lock while melting is in progress. The VIDP is capable of refining steel by the introduction of reactive gases through a porous plug in the bottom of the melting crucible. A melt, once ready for casting, is delivered to the mold chamber via a separate vacuum chamber. Use of vacuum as opposed to

air melting ensures a high quality product, it minimizes slag production, and minimizes the potential for airborne contamination. All gasses produced during the melting process are extracted directly into the HEPA ventilation system, minimizing the reliance on room extraction to capture potential contaminants.

The main furnace operations are carried out by an operator in a remote control room. Various support operations are necessary for the melting process including mold preparation, which is carried out on a large fire brick lay-down area, crucible relining which is carried out in a separately ventilated room off the main foundry area, scrap metal sectioning for volume reduction prior to melting, and ingot preparation for future processing. As in other areas, reducing operator involvement with materials to a minimum reduces worker dose uptake and exposure to industrial hazards.

Laboratory

The laboratory services three main functions: radiochemistry, environmental analysis, and metallurgy. To support the QA program, the principles of Just-in-Time and the minimizing of in-process inventory, it has been necessary to provide a comprehensive range of instruments. It is necessary to provide this equipment on-site to obtain immediate results, thus avoiding the multi-day turnaround from outside services.

Survey for Release

Following each decontamination process, metals are dispositioned for melt or for survey prior to free release. The disposition decision is based on the level of remaining contamination, and the level of certainty with which the material can be successfully surveyed. Metals with residual contamination, or that which will not be surveyable, will be dispositioned for melt otherwise they will be dispositioned for free release. Metal ingots from the melt process can be used to fabricate products for restricted use.

Metals are received in the final survey area in boxes via the monorail system. The area contains semi-automated systems for counting and tracking 'removable' and 'fixed' contamination. Material is collected in scrap hoppers which are coded into the MIS system by bar code scanning. When results are available from the survey systems, the MIS system reports the status of the material for release. Once passed by QA, the material is moved to outside storage containers. Material that fails the survey will either be sent to the finishing area, where small areas of contamination will be removed, or it will be sent via the monorail to the foundry for melting.

DESIGN AND CONSTRUCTION MANAGEMENT

The design and construction of the RSMRP was carried out by a core project team of seven, with representatives from each of the main partners; two from BNFL in the UK, two from BNFL's USA subsidiary and three from MSC. This team was supplemented with various short term contract support as necessary. The project team extended the teaming approach adopted by BNFL and MSC to encompass the main contractors,including the Architect/Engineering Group, the general contractor, and major suppliers. In many instances, these external groups had representation at internal project meetings. In an environment of openness and clear objectives, rigorous project management and control techniques could be used in a positive and constructive manner. External groups were introduced to the project with a briefing on project philosophies. This briefing covered: budgets, schedule, contractor

relationship, styles of management, and most importantly, team concepts. Acceptance of these philosophies was a prerequisite to a group's involvement within the project. The result was a seamless relationship between the project team and the architect/engineers which fostered an open and critical design approach benefiting both parties. This approach was extended to the general contractor, their active involvement and the application of their experience, and resulted in a major value engineering exercise which reduced the facility's complexity and cost. The approach was initially treated with suspicion by major vendors, but as they began to experience the benefits of being allowed to do what they do best without the customer meddling, they became active members of the team. Clearly, the approach had varying degrees of success, but where it was successful, the results were dramatic, deliveries were on time, there were minimal contract claims, equipment actually conformed to requirements and partnerships were formed that will ensure the success of future projects.

The primary approach to scheduling was via simple milestones. Responsibility for meeting these milestones was delegated to those responsible for doing the work. The result was control systems ranging from detailed Primevera critical path networks, through Microsoft project schedules, to simple action lists. Using the appropriate tool for the task ensured support and ownership by the user. Backed by weekly project reviews, issues were constantly visible and corrective action was almost immediate. This process was backed by a detailed engineering and cost control system which ensured deviations due to errors, or changes in scope, were instantly highlighted. The approach allowed the team to complete the project within a $28M budget and on schedule:

- Site and vendor preparation October 1994
- Funding March 1995
- Site opening up April 1995
- First steel June 1995
- First receipt of active materials December 1995
- Operational March 1996

CONCLUSION

This project has demonstrated that the potential for future business can stimulate private investment in the creation of cost-effective and timely solutions for the customer. The imperative to find efficient solutions to the recycle issue has encouraged an innovative approach to break the barriers of existing metals recycle technology. This process has been strengthened and enhanced by a performance-oriented teamwork approach.

13-5

ELECTROLYTIC DECONTAMINATION OF STAINLESS STEEL MATERIALS IN A SODIUM NITRATE ELECTROLYTE FOR HAZARDOUS WASTE MANAGEMENT

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ABSTRACT

Electrolytic decontamination of plutonium and americium from stainless steel and uranium surfaces has been demonstrated. This decontamination process is similar to industrial electropolishing processes but is carried out in a sodium nitrate electrolyte from which the metals can be

precipitated. The separation of the metals from the electrolyte allows for recycle of the electrolyte; hence, no aqueous waste stream is produced. The generated waste is in the form of a precipitated and is therefore very minimal and compact.

An example application is a "can-out" process for the decontamination of radioactive material storage containers being developed at Los Alamos National Laboratory. This process utilizes electrolytic decontamination as its primary component. The "can-out" decontamination system and the underlying chemistry is outlined. Other successful applications of this technology are the in-situ decontamination of gloveboxes and highly enriched uranium.

INTRODUCTION

There are a variety of reasons for decontaminating surfaces, including lowering the radioactive waste categories of items, decontaminating items during processing to remove them from plutonium (Pu) gloveboxes, and enabling the disposition of special nuclear materials. In the past, many material surfaces were cleaned with concentrated acids that in turn produced large amounts of radioactive toxic waste. These acid wash methods are also inefficient for reducing contamination to desired levels.

Electrochemistry represents a viable alternative to these existing technologies for decontaminating a variety of radioactive and toxic wastes throughout the nuclear complex. Electrochemical methods produce significantly less waste and can also increase the efficiency of many existing processing technologies.

Electrolytic methods have been shown to be superior to acid washing methods. At the Los Alamos National Laboratory Plutonium Facility, we have been using electrolytic methods to clean a variety of conductive surfaces, including stainless steel and uranium. The focus of this paper is the application of this technology to a "can-out" procedure for the decontamination of nuclear material storage containers.

ELECTROLYTIC DECONTAMINATION

Electrolytic decontamination is similar to the common industrial practice of electropolishing and is accomplished by applying a low dc voltage through a suitable electrolyte to induce a chemical reaction at a mass-diffusion limited rate. Material is removed at the anodethe working electrodeand goes into solution. The cathode, or counter electrode, can be constructed from a variety of materials, but stainless steel is typically used. The electrolyte of choice for this process has been sodium nitrate at neutral to high pH because the radio active contaminants and major steel components form a precipitate. This precipitate formation leads to easy separation of the waste components from the solution; thus, the electrolyte solution can be recycled, greatly reducing waste as compared to past acid wash processes.

INDUSTRIAL ELECTROPOLISHING

Typical cold-rolled type 304L stainless steel has a microscopically rough surface as shown in Fig. 1. Allen and Randall (1) have documented many steel surfaces produced by various finishing techniques and have shown these surfaces to be disadvantageous when the material is introduced into a situation where surface contamination with radioactive species may occur. The disadvantage lies in decontamination difficulties related to the entrapment of contamination within these surface cracks and crevices. This is the case when dealing with containers that are designed for long-term storage of special nuclear materials. It is preferable to remove

this surface roughness before a possible contamination situation. Furthermore, mechanically polished materials are inadequate because of the many scratches and groove introduced by abrasives used in the polishing procedure. An electropolished surface, however, is microscopically smooth and creates no surface scratches that could entrap radioactive materials.

In industrial electropolishing methods, the material to be polished is submersed in an electrolyte bath and anodically polarized. With the passage of current, the surface layers of the metal oxidize to ionic species and then diffuse into solution. Under appropriate conditions, high points on the surface are preferentially removed resulting in a leveling of the surface. This polishing effect only occurs when the dissolution process is occurring at a mass-transfer limited rate. Typical conditions for electropolishing are well documented, based on the material to be polished (2). For stainless steel, the best electrolytes are composed predominantly of phosphoric and sulfuric acids in varying ratios. For a pretreatment of non-radioactively contaminated metal parts, these electrolytes are satisfactory. A typical electropolished 304L is shown in Fig. 2.

Fig. 1

Fig. 2

In the "can-out" process, the storage cans are electropolished by such an industrial technique prior to introduction to the glove box line. This prepolishing step assures that removal of only a few monolayers of the steel will be necessary to decontaminate the microscopically smooth surface of the can. As an example, a five-monolayer removal equates to only about 0.5 mgcm-2 of type 304 stainless steel.

ELECTROLYTIC DECONTAMINATION

Once a piece of steel has been surface contaminated with radioactive species, these industrial electropolishing electrolytes are not desirable. The high acid characteristic of these electrolytes would serve to dissolve oxides of Pu and Am and keep them in solution. After only a few articles are decontaminated, the material buildup within the electrolyte solution would mandate a changing of solutions and result in large quantities of radioactive toxic waste that must be treated and/or disposed. A better solution is to pick an electrolyte with a very low toxicity from which the radioactive materials can easily be separated; thus, the electrolyte can be recycled. Such a choice is a solution of sodium nitrate in water.

Passing an anodic current through a piece of stainless steel submersed in a sodium nitrate electrolyte results in dissolution of the steel. Childs and Winkel (3) have shown that this electrolytic removal of the surface layers of the steel can result in the removal of surface contamination. At the Los Alamos National Laboratory Plutonium Facility, we have demonstrated electrolytic decontamination of Pu and Am from stainless steel (4) and uranium (5) to contamination levels that enable materials to be removed from glovebox containment or that lower the radioactive waste category. For example, we have demonstrated the removal of alpha contaminants, like Pu and Am, from >1,000,000 counts per minutes (cpm) to approximately 1000 cpm by in situ electrolytic decontamination of gloveboxes (4). (The in situ method allows for the glovebox to remain installed to existing air handling.)

Removal of these contaminants lowers the waste category from a transuranic level to low level, greatly reducing disposal costs and

enhancing the likelihood of glovebox recycle. Although it is clear that the electrolytic decontamination process works, the chemistry of the process has not been fully characterized. In a previous paper (6), we reported that removal rates were found to be highest at higher electrolyte concentrations and in acidic pH. Satisfactory removal rates were found at sodium nitrate concentrations of 200 gL-1 and above. Stirring was also determined to be advantageous in producing a smooth surface. The appropriate current densities appear to be in the range of 0.1 and 0.2 Acm-2. Higher current densities result in higher metal removal rates, but adversely affect the surface morphology by causing roughening, pitting, or burning. These are all indicators of a non-uniform removal. At pH <2, metal removal rates are high and the resulting surface morphology superb, but precipitation of the removed metal and the actinides did not occur.

The dissolution reactions of the steel substrate in the electrolytic decontamination process are similar to those in electropolishing. Under mass-transfer limited current conditions in a concentrated sodium nitrate electrolyte, iron is oxidized to ferric ion and chromium to the hexavalent chromate ion. Nickel leaves the surface as nickelous ion. The other anode reaction is the oxidation of the solvent to evolve oxygen gas:

Eq. 1

Though the electrolyte solution may initially be at a neutral pH, the solution is quickly made alkaline due to the primary cathode reaction:

Eq. 2

Although hydrogen gas is produced as a byproduct of the electrolytic decontamination process, the rate of hydrogen production is so low that safety is not compromised, even in a glovebox environment (7).

Also, a small portion of cathodic current goes to the reduction of nitrate to ammonia:

Eq. 3

We have not yet quantified the significance of this reaction under the conditions of the decontamination process. Qualitatively, after running the system for extended periods of time, the faint oder of ammonia is detectable.

Under the alkaline conditions of the electrolyte solution, Fe3+ and Ni2+ have a very low solubility and precipitate as ferric and nickel hydroxides, respectively. Since the primary component in stainless steel is iron, the precipitate is largely composed of ferric hydroxide. This precipitate has a gelatinous nature and tends to incorporate other materials, including actinides, as it precipitates. In this respect, the electrolytic decontamination technique is similar to ferric flock procedures that are commonly used on an industrial scale to purify water.

CHROMATE REMOVAL

At the current densities of interest and at neutral to high pH, chromium is anodically stripped from steel as hexavalent chromium. In aqueous solution, hexavalent chromium exists in an acid-base equilibrium between the chromate and dichromate forms. At high pH, the predominant species is chromate, while at low pH, the predominating species is dichromate. Hexavalent chromium cannot be removed from aqueous solution as a hydroxide as are the other metals. This initial nonprecipitation is of concern because as the concentration of chromium rises in solution, it begins to adversely affect the electrolytic stripping process (3).

An electrochemical process for the reduction of hexavalent chromium and its subsequent precipitation has been demonstrated (8). Trivalent chromium, in contrast to hexavalent chromium, is readily removed from solution by precipitation as the hydroxide. Unfortunately, under the alkaline conditions of the decontamination process, the predominant form of hexavalent chromium is chromate. Chromate is difficult to reduce. Dichromate, on the other hand, is readily reduced either electrochemically or by addition of a suitable reducing agent such as the ferrous ion. Therefore, by acidifying the solution and carrying out a reduction, dichromate can be reduced to trivalent chromium. The trivalent chromium can subsequently be precipitated as a hydroxide by raising the solution pH.

THE "CAN-OUT" PROCESS

We are now in the stages of demonstrating a "can-out" process for radioactive materials packaging, which allows these materials to be packaged in stainless steel cans, the cans decontaminated, and removed from the contaminated glovebox line to the room environment. This process reduced Pu and Am contamination levels from the transuranic category to essentially "free release." The process utilizes the fixture in Fig. 3 for the electrolytic decontamination of the cans. This fixture is designed to suspend a welder storage can allowing electrolyte to be circulated through the gap between the interior of the fixture and the exterior can walls. The can is in electrical connection with the power supply by electrode contacts at the top and bottom of the can. The stainless steel housing of the fixture acts as the cathode during the decontamination step.

Fig. 3

The radioactive materials are placed within a previously electropolished stainless steel can and the can welded shut. The can is then placed inside the decontamination fixture from the "hot" side. A solution of sodium nitrate electrolyte is circulated through the fixture and the can polarized anodically to induce the dissolution of the outer layers of the stainless steel. As the material is stripped from the can, the iron, nickel, and actinides form precipitates within the electrolyte. These precipitates are then removed by filtration in either an in-line or batch mode. In an optimized configuration, the precipitates are filtered in-line, providing a clean electrolyte solution for recycle through the decontamination fixture. Due to the nature of the precipitate, in-line filtration is difficult, but we have had success with an appropriately scaled ultrafiltration module. Batch filtration is readily accomplished by allowing the precipitate to coagulate and settle after the decontamination step and prior to filtration. After decontamination, a water rinse is flushed through the fixture. The decontaminated can is then removed to the "cold" side of the fixture.

After a number of cans have been decontaminated, the chromate levels in the electrolyte must be reduced. To accomplish this, we are investigating an electrochemical reduction utilizing a sacrificial iron electrode (Fig. 4). The process involves first acidifying the solution to convert chromate to dichromate. The solution is then fed into an industrial electrochemical cell comprised of a stainless steel cathode and a sacrificial iron anode. As current is forced to flow through the cell, the iron electrode is oxidized to produce ferrous ion. In solution, ferrous ion undergoes a redox reaction with the dichromate to produce ferric ion and trivalent chromium. At the cathode, the primary reaction

is the reduction of water, though there is also some reduction of dichromate and some electrodeposition of iron metal.
Following reduction of the dichromate, the pH of the solution is raised by addition of sodium hydroxide. Both the trivalent chromium and the ferric ion precipitate as hydroxides and are subsequently filtered and removed from the solution. The electrolyte solution is then returned to the electrolytic decontamination process.
Fig. 4

CONCLUSIONS

The electrolytic decontamination technique has been shown to be a viable alternative to previous acid washing decontamination methods. It has overwhelming advantages in that the only waste generated is both minimal and compact. The electrolyte solution can be recycled indefinitely since all the contaminants are readily removed. Furthermore, unlike an acid washing process, electrolytic decontamination can be accomplished, no matter how much material removal is required, on a very short time scale. Material removal rate is proportional to the applied current, affording the operator control over the rate and degree of the decontamination. The "can-out" process for decontamination of stainless steel cans used in the storage of radioactive materials utilizes the electrolytic decontamination technique. Cold lab testing of the process has been very successful and a demonstration scale process will be installed in the Los Alamos National Laboratory Plutonium Facility for hot testing by the time of this conference.

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THE NATIONAL CONVERSION PILOT PROJECT REGULATORY PROCESS

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ABSTRACT

Based on changes in the mission of the Rocky Flats Environmental Technology Site, and on a desire to accomplish tangible cleanup goals, the National Conversion Pilot Project (NCPP) worked in coordination with the regulators and the public to develop a more streamlined approach to decontaminating buildings at Rocky Flats. The result of this work is an approach remarkably like that described in the joint DOE/EPA Policy on Decommissioning Department of Energy Facilities under CERCLA, which was published almost 18 months after the NCPP began.

By emphasizing coordination and a common-sense approach to regulatory compliance, two unrelated efforts arrived at essentially the same conclusions.

INTRODUCTION

More than the name at Rocky Flats has changed. The site is now called Rocky Flats Environmental Technology Site; a name that reflects the myriad of changes at this site and across the DOE complex. With the end of the Cold War, the missions of the DOE and most of its sites have been replaced with missions that consider what to do with the buildings and people, while dealing with the by-products of previous missions. It has become clear that the DOE's primary goal now is to safely decontaminate the sites and facilities that are no longer needed for producing weapons. In addition to the new missions, most DOE sites are facing smaller budgets, members of Congress calling for the elimination of the Department, and levels of regulatory involvement that are higher than those faced in the past. To deal with all of these changes, and many more not mentioned here, the DOE realized that it needed a way to clean sites effectively, show tangible results, and meet the requirements of the regulators and the expectations of the public. In short, the DOE is looking to conduct business more like a private entity.

There are many efforts underway across the DOE complex that work toward this end; one of these efforts is the National Conversion Pilot Project (NCPP). Simply put, the NCPP is designed to recycle material, facilities, and workers. The project will employ former Rocky Flats workers to decontaminate metal-working facilities at the site to agreed upon levels, and then use the facilities and workers to recycle material that would otherwise be considered Low Level Waste into products that are useful to either the DOE or the private sector.

THE NCPP MISSION

The mission of the NCPP is "to explore and demonstrate, at the Rocky Flats Environmental Technology Site (RFETS), the feasibility of economic conversion at Department of Energy (DOE) facilities." Economic conversion is the conversion of facilities and equipment owned by the Federal government to production of goods by private firms for profit. However, not only buildings and equipment are being converted. Turning buildings on a DOE facility over to private use involves changing ideas about how things will be done, and about who will do them.

The NCPP was authorized in December 1993 by Secretary of Energy Hazel O'Leary to proceed in three distinct stages with a review and positive decision required at the end of each stage before proceeding to the next. The NCPP is being conducted through a Cooperative Agreement,which was signed April 1, 1994, between the U.S. DOE Rocky Flats Field Office (RFFO) and Manufacturing Sciences Corporation (MSC).

The NCPP has been divided into three stages, Stage I (feasibility and planning), Stage II (cleanup and refurbishment), and Stage III (recycling and manufacturing). There are decision points at the ends of Stages I and II and periodically during Stage III, to help ensure careful consideration of project feasibility and the opportunity for feedback from Stakeholders. At the end of each stage, the project could be revised or terminated, and DOE will only proceed with the support of regulatory agencies and the acceptance of the community.

To enhance the probability of a favorable decision at each of these points, and to ensure that tangible cleanup activities are achieved, the NCPP employed an approach that would:

- encourage public participation,
- establish an NCPP Steering Committee,
- build regulatory relationships,
- employ an Interim Measure / Interim Remedial Action as the regulatory vehicle, and
- transition away from DOE Orders.

Together, these activities create a climate that enables the NCPP to work with the interested parties (namely, the DOE, the regulators, and the public) to identify concerns and develop responses that are agreeable to all. The specific details of other projects will differ from those of the NCPP. However, the overall concept of identifying the affected parties, sincerely seeking their input, obtaining agreement on planned activities, and maintaining open lines of communication not only apply to other projects, it is vital to their success in light of the current situation. These five activities are discussed in the following sections.

PUBLIC PARTICIPATION

Public participation is one of the cornerstones of the NCPP, as evidenced by the requirement that public acceptance of the project be obtained prior to beginning each Stage of the project. The project's primary contractor, MSC, feels that informing and involving the public increases the likelihood of obtaining public acceptance. As demonstrated at many sites across the country (both DOE and commercial nuclear) the more the public knows about the activities on a site, the more likely they are to support those activities, to one degree or another. With the end of the Cold War, Americans demand to be more well informed, and are less likely to accept the reasoning that national security prevents them from having a say in activities that may affect them. By including members of the public in the planning process, many ideas that might otherwise not have been considered are identified and examined.

The NCPP public outreach program actually began before the project was initiated. Approximately six months before the NCPP was started, MSC began a campaign to inform interested stakeholders of the proposed project. The groups and individuals that MSC met with included representatives from cities and counties in the Rocky Flats vicinity, staff members from Colorado's congressional delegation, state representatives, local activist organizations, and community groups. In addition to providing information, the purpose of these meetings was to solicit comments and questions regarding the proposal.

The comments and questions received during this effort were researched and responded to in writing under the NCPP Issue Response Process. This process was administered by the NCPP Steering Committee, and is described in Section 3, NCPP Steering Committee.

The NCPP public outreach program continues, with periodic bulletins and pamphlets discussing the NCPP progress, and soliciting further public questions and participation. These are distributed to a list of over 500 individuals and organizations. In addition, the NCPP encourages visits and tours of the NCPP activities in progress. Tours of NCPP facilities have been provided to individuals and groups including the Secretary of Energy, personnel from other DOE sites, members of the international nuclear community, and other organizations at Rocky Flats and the vicinity.

NCPP STEERING COMMITTEE

Since the onset of the NCPP, a Steering Committee has been in place to help determine project-specific policy and respond to project-related issues. The NCPP Steering Committee is composed of members from each of the following organizations: MSC, DOE Rocky Flats Field Office (DOE-RFFO), the Environmental Protection Agency, Colorado Department of Public Health & Environment, Rocky Flats Local Impacts Initiative, the Rocky Flats Integrating Management Contractor, and the Colorado Office of Business Development. By involving these interested parties, including the regulators, in this process, regulatory issues can be dealt with more quickly and effectively. The NCPP Steering Committee, while having no regulatory authority, provides the project with a forum where regulatory issues can be discussed and guidance for the project developed.

As mentioned in Section 2, Public Participation, an Issue Resolution Process was developed that entailed the use of formal and informal methods to receive public comments. Comments were received through correspondence and public meetings. The Steering Committee maintained a log of all issues raised and tracked their status during the resolution process. The log did not attribute the issues to individuals in order to assure objective answers and maximize the chance of issues surfacing.

A number of subcommittees were developed to assist the Steering Committee. These subcommittees were composed of technical experts from the plant, the regulatory community, the Governor's Office and MSC. The Steering Committee then assigned each issue to the subcommittee that could most reasonably respond to the issue. Subcommittee subjects included human resources, site support services, environmental restoration, waste management, public outreach, and business planning. Once the subcommittee had researched the issue and developed a response acceptable to the subcommittee members, the response was forwarded to the Steering Committee. When the response was acceptable to all concerned, the responses were published and released to public reading rooms associated with the Rocky Flats site.

This process, while appearing complicated, provided the regulators, the public, and the NCPP with the opportunity to discuss issues in an open manner and to develop proposed resolutions without compromising official organizational positions. The Steering Committee continues to meet monthly to discuss progress on the NCPP and to provide guidance. The subcommittees will meet on an ad hoc basis during Stage II, as issues arise.

IM/IRA

Interim Measure/Interim Remedial Action (IM/IRA) Decision Documents are typically used as a vehicle for contaminant mitigation, abatement, and/or risk reduction under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) where there exists an imminent threat to the public or the environment. Although the NCPP poses no

imminent threat to the public or environment, the IM/IRA process was recommended by the Steering Committee, and is being used to establish and direct the achievement of a baseline for identified hazards that would allow the continued safe operations of the facilities by a private concern. Another reason for using the IM/IRA process is to solicit public involvement in the project.

The NCPP IM/IRA Decision Document identifies interim remedial actions for removal of uranium and chemical contaminants of concern, such as beryllium and others, which have been used within the three buildings that are associated with the NCPP. The remedial actions are proposed, not for reasons of mitigating an imminent threat, but rather, to evaluate cleanup techniques and to baseline the buildings, for the attainment of a safe and healthy workplace allowing future operations by a private concern.

REGULATORY RELATIONSHIPS

Over the course of the NCPP, emphasis has been placed on keeping the public and the regulators informed, and on obtaining and using input from these groups. Through the process of developing and refining the IM/IRA and its supporting documents, a unique relationship between the regulators and the NCPP has evolved. By including the regulators on the Steering Committee and holding numerous face-to-face meetings with the regulators on various subjects, what began as a somewhat distrustful relationship has become one of mutual respect where teamwork and professionalism are used to identify ways to accomplish common goals and comply with the applicable regulations.

As issues have arisen, the project continues to meet with the regulators to jointly develop proposed resolutions. By obtaining input from the regulators during the resolution planning, the NCPP benefits from the regulator's knowledge of the regulation as well as from their interpretation. This enables the NCPP to greatly reduce the possibility of misunderstandings with the regulators. It also provides the regulators with more timely information on the project, allowing them to better understand the issues facing the project.

The NCPP, each of the regulators, and various groups within DOE-RFFO have named one primary point of contact within their organizations. These individuals discuss any issues that arise, and obtain technical expertise from others in their organizations as needed to determine an agreeable resolution. This arrangement has proven to be a more streamlined approach than having each organization "working in a vacuum."

TRANSITION FROM DOE ORDERS

The intent of the project is for Stage III (recycling and manufacturing) to be performed entirely by a private company, under the regulations that any other private company would have to comply with. This goal will necessarily involve the elimination of the current requirement to comply with DOE Orders, since other companies in the open market do not have to comply with DOE Orders.

Since the NCPP buildings are on a DOE facility, they are currently subject to DOE Orders as well as regulations that apply to private business performing the same activities. There is some duplication between these sets of regulations that will need to be eliminated if the NCPP buildings are to be converted for use by a private business. Additionally, there are some DOE Orders that will not apply to a private business. The DOE requirements that are either duplicative or not applicable to private businesses will be identified by MSC, and

documentation will be developed that justifies eliminating the need for the NCPP to comply with the requirements.
At this point, the justification for eliminating the need for complying with the requirement will be presented to the DOE and the regulators. Once all parties agree that the justification is adequate, each will approve the justification.
CONCLUSIONS
An interesting side-note concerns the joint DOE/EPA Policy on Decommissioning Department of Energy Facilities under CERCLA. This policy, signed by both organizations on May 22, 1995 "is the result of a joint effort by EPA and DOE to develop an approach to decommissioning that ensures protection of worker and public health and the environment, that is consistent with CERCLA, that provides for stakeholder involvement, and that achieves risk reduction without unnecessary delay." This policy builds on the joint DOE/EPA goal to "develop decisions that appropriately address the reduction of risk to human health and the environment as expeditiously as the law allows."The NCPP was not related to the Headquarters-level efforts to develop the joint policy on decommissioning. But by focusing on the end goal of tangible cleanup achievements, and cooperating to find common ground, both efforts arrived at essentially the same conclusion.

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THE REUSE OF FACILITIES, MATERIALS AND LABOR- THE NCPP AT ROCKY FLATS

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ABSTRACT
The National Conversion Pilot Project, currently underway at Rocky Flats Environmental Technology Site, Denver, Colorado, aims to show how ex-weapon production facilities within the Department of Energy (DOE) can become successful commercial operations. This project, unique within the DOE, is converting three major buildings on the plantsite in preparation for commercial metal forming/processing activities. Preparation for commercial operations involves the radiological and toxicological decontamination of the buildings, the removal of redundant facilities and the refurbishment of tooling and equipment required for operations. Project planning began in April 1994 with cleanup activities commencing in Building 883 in April 1995. Completion of cleanup work is expected during FY 1997.
Overall project management is the responsibility of Manufacturing Sciences Corporation (MSC) through a cooperative agreement with the DOE. BNFL Inc. is the principal subcontractor to MSC.
The NCPP can be viewed as a three way recycling project:
  The recycling of buildings and equipment from weapons production to commercial operation
  The recycling of scrap steel, as well as uranium and beryllium, away from weapon component production to commercial usage (e.g. the fabrication of waste storage boxes), and

The recycling of former Rocky Flats workers, made redundant following the plantsite mission change, from weapon component production workers to manufacturers of commercially viable products.

By recycling facilities, materials and labor, the project aims to show that the conversion of contaminated ex-weapons facilities into commercial uses can reduce the burden of underutilized facilities upon the DOE as well as being an economically viable commercial operation.

This paper describes the progress made to date in the decontamination and refurbishment of these facilities as part of the NCPP.

INTRODUCTION

The National Conversion Pilot Project (NCPP) is a recycling project currently underway at the Department of Energy (DOE) Rocky Flats Environmental Technology Site (RFETS) in Colorado. This paper describes the NCPP from inception through to the current status of the decontamination and dismantling work in progress on the plantsite.

HISTORY OF THE SITE

RFETS has an area of 6,500 acres and is located 16 miles NW of Denver, Colorado. The plantsite began operating in 1951, manufacturing weapons components and conducting plutonium processing operations. RFETS currently comprises numerous buildings within the plantsite boundaries with radiological or toxic contamination. In 1993, the production oriented plantsite mission changed to read "manage waste and materials, cleanup and convert RFETS to beneficial use in a manner that is safe, environmentally and socially responsible, physically secure and cost effective". This change of mission led to the cessation of production facilities with the plantsite entering a surveillance and maintenance state while cleanup plans and options were considered.

THE NATIONAL CONVERSION PILOT PROJECT

In 1993, Manufacturing Sciences Corporation (MSC) proposed to the Department of Energy (DOE) the radioactive and toxic cleanup of three major plantsite buildings in preparation for the possibility of using these refurbished buildings for commercial metal recycling and manufacturing operations.

The NCPP proposal involved three radiologically and toxicologically contaminated buildings on the RFETS plantsite (883, 865 and 444/7) with a combined floor area in excess of 200,000 ft2. Equipment within these buildings includes ovens, furnaces, presses, metal rolling equipment, shears, forges, lathes and other metallurgical processing equipment. Equipment of note includes nine vacuum induction melting furnaces, one new vacuum arc remelting furnace, two large rolling mills, a new 2000 ton extrusion press and six hydraulic forming presses including two 2000 ton presses and two 200 ton presses with 64" strokes This equipment would be valued at $92M if installed new today. Following the end of the Cold War and the change of mission at RFETS, these buildings and contents became surplus to requirements.

The buildings were selected principally because they contained machinery and tooling suitable for metal recycling operations. In addition, the buildings were located such that they could be accessed by contract personnel without affecting the security of other parts of the plantsite.

In April 1994, the DOE entered into a Co-operative Agreement with MSC to perform this work. MSC selected BNFL Inc. as its principal subcontractor to utilize experience gained by BNFL Inc.'s parent, British Nuclear Fuels plc, conducting similar cleanup at its UK sites.

AIMS OF THE NCPP

The NCPP mission is to "explore and demonstrate, at RFETS, the feasibility of economic conversion at DOE facilities". Within this mission, the NCPP has the following recycling intentions:

To re-employ former RFETS workers,

To reuse existing nuclear weapon component production facilities for the production of commercially marketable products, and

To reuse existing materials (uranium, beryllium and radioactively contaminated scrap metals) for the production of these products.

The NCPP was authorized in December 1993 by Secretary of Energy Hazel O'Leary with Stage I commencing in April 1994.

STAGES OF THE NCPP

The Project has three stages. Stage I was the feasibility and planning phase of the project and was completed in September 1994. Stage II is the cleanup and refurbishment phase during which unnecessary equipment will be removed from the facility, equipment to be retained for Stage III will be maintained and restored to operational use, and building contamination will be reduced to a level suitable for recycling operations. Stage III is the recycling and manufacturing phase of the Project when the reuse of facilities and materials will take place.

Stage I

The planning during Stage I for Stage II activities was guided by two principal considerations. These were whether or not equipment was considered useful for future Stage III operations (and if it could be economically cleaned and maintained within the Stage II funding) and the degree of contamination expected to be uncovered during dismantling. The decision on which equipment to keep was made by DOE. The remainder was categorized into the following groups in order of decreasing priority:

Equipment which could economically be decontaminated and moved to a separate DOE facility for reuse.

Equipment which could be economically decontaminated for unrestricted use outside RFETS and ownership transferred out of the DOE.

Equipment which could be economically decontaminated for unrestricted use outside RFETS and sold as scrap.

Unwanted equipment fabricated from metals suitable for reuse during Stage III which could be dismantled, decontaminated and stored pending Stage III recycling operations.

Unwanted equipment which could not be economically cleaned which will be disposed of as low level or low level mixed waste.

During Stage I, within the above guidelines, cleanup plans were established covering all three NCPP buildings. Characterization data were collected for each building which assisted cleanup planning and identified potential radiological and toxicological considerations. Stage I also involved marketing analyses, the production of training and staffing plans and regulatory oversight and site support service agreements. As well as cleanup during Stage II, it is proposed that operational assessment of equipment to be retained was to take place. Declassification of materials and tooling (approx. 272,000 lb.) and process verification for intended products will also occur in Stage II. In light of the proposed reuse or removal of items of equipment and the expected degree of contamination they may contain, a generic 10 step approach was devised which will be applied to each building in turn. Those ten steps are as follows:

Step 1 Removal of loose waste/unwanted equipment

Step 2 Cleaning of any high contamination hot spots

Step 3 Assembly of a centralized decontamination module
Step 4 Removal and decontamination of low contamination risk items of equipment
Step 5 Removal and decontamination of high contamination risk items of equipment
Step 6 Decontamination, dismantling and removal of utilities no longer required
Step 7 Repairs to building structure following equipment removal
Step 8 Dismantling, cleaning, maintenance, reassembly and testing of low contamination risk equipment to be retained
Step 9 Dismantling, cleaning, maintenance, reassembly and testing of high contamination risk equipment to be retained
Step 10 Final decontamination of complete building to operational levels
The above steps reflect the general principle of emptying the building of all unwanted equipment first, carrying out general repairs and then cleaning and refurbishing all remaining, wanted equipment.

Stage II

Stage II began in October 1994 and received regulatory sanction in April 1995 with the approval of the Interim Measures/Interim Remedial Action Decision Document which has been used to direct activities to take place in Stage II and to solicit public involvement in the project.
The Project is considered to be a recycling operation and, as such, recycling will be along three lines:

FACILITY RECYCLING

Each of the three buildings will be converted from weapons production operations to a part of a commercial metal recycling operation. Each building contains unique metal working equipment, much of it valuable, some of it unused but all still operable with, in some cases, minor maintenance. Unwanted equipment will be removed (as described above) for reuse elsewhere or, if not economically feasible to recover, either retained as recycle scrap for Stage III if suitable, or disposed of as waste. Equipment within the buildings will, as necessary, be reorganized. Wanted equipment will be refurbished after cleaning. Buildings will be returned to operational status prior to the end of Stage II.

MATERIALS RECYCLING

Each building contains materials which are suitable for recycling into commercially viable products. These materials fall into one of three categories: depleted uranium, beryllium and radioactive scrap metal. As each building undergoes cleanup, stocks of depleted uranium and beryllium, not required by the DOE, will be recovered and stored pending Stage III recycling. Radioactive scrap metal will arise from the cleanup process as unwanted equipment is dismantled and removed. Scrap suitable for use in future recycling operations will, after size reduction, be retained in the building. Not all metal scrap is suitable for recycling with stainless steel being the preferred Stage III feedstock. However, it is intended that substantial sections of other steels will be retained as additional feedstock materials. Within the three buildings, it is expected that 200,000 lb. of depleted uranium, 80,000 lb. of beryllium and 160,000 lb. of recyclable steel exist.

LABOR RECYCLING

With the change of mission, production operations ceased on plantsite with the result that labor would have to be shed. The final aspect of recycling concerns the reuse of labor made available due to the mission change. It is the intention of the project that the Stage II and III

workforce are made up from rehiring displaced RFETS workers. Many of the displaced workers have a working knowledge of the NCPP buildings, both operationally and through maintenance. Workers hoping to join the project will undergo a selection procedure which considers their existing skills and experience which are relevant to the project. Workers, when hired, will go through a cultural retraining and will receive training in essential areas where they do not have relevant experience. There is no 'blanket' training for all employees. Flexible working methods and a team approach to working practices are encouraged.

STAGE II CLEANUP METHODS

Stage II cleanup is being be achieved along the principle of local dismantling with a centralized decontamination facility within each of the three buildings. The choice of cleanup methods was determined on the basis of the known and expected contamination and radiation levels in each building at the time of planning. A predominantly hands-on approach was taken since:

- Contamination levels are not expected to be high enough to necessitate remote methods
- The project works to a fixed budget which precludes significant tooling investment
- The buildings are geographically split rendering a centralized, automated size reduction station impractical

In addition to the above considerations, the type of decontamination processes used was restricted by the type of effluent produced. Any liquid waste leaving the building will be processed through the RFETS effluent treatment facility which has stringent parameters for acceptance. Many of the effective decontamination solutions available on the market were incompatible with the effluent treatment system.

The cleanup systems chosen were wiping with acceptable decontamination reagents, vacuum cleaning, grinding, grit blasting and CO2 blasting. The CO2 system existed on plantsite though had been rarely used. The system was recovered, assembled and tested. It now operates in a HEPA-filtered booth as a non-destructive decontamination system which produces little secondary waste from its operation.

Dismantling methods are largely manual for many of the same reasons described in the previous section. For equipment to be dismantled, maintained and reassembled, hand disassembly (wrenches, etc.) are used however, for equipment to be disposed of, size reduction methods used include plasma arc and oxyacetylene cutting, cold cutting methods (saws, grinders, nibbler) as well as simple hand tools.

Wherever possible, decontamination is carried out centrally in the CO2 chamber with dismantling and reassembly operations performed locally.

CLEANUP PROGRESS SO FAR

Cleanup activities began, initially on a limited scale, in April 1995. The schedule for building cleanup is as follows:

- FY 1995: first half of 883
- FY 1996: second half of 883, 865 begins
- FY 1997: 865 completed, 444/7 cleaned

Delays in spring and summer 1995 prevented early commencement of cleanup operations in building 883 and, consequently, work on the first half of the building was completed, within budget, by the end of 1995. In anticipation of DOE funding for calendar year 1996, cleanup work has begun on the second half of the building.

At this time five decontamination teams exist and a number of decontamination and dismantling operations are in progress within Building 883. Within the first half of the building, all items to be disposed of have now been size reduced and either stored as recycle material or disposed of as waste. The items that have been dismantled and removed are the laser inspection table, nitric acid bath, sheet scrubber and small rolling mill. The equipment that will remain in the building half have been cleaned, dismantled, maintained and reassembled. These items include a large rolling mill (used for uranium sheet rolling), a 2000 ton hydraulic press, cooling water quench tanks, two shears and a radiant oven. Final work in the building half included floor repairs and building surveying and decontamination. In this half of the building, 91,600 ft2 of floor, wall and ceiling surfaces have been cleaned with 35,000 lb. of recyclable material retained for Stage III operations. In 1995, 903 ft3 of low level waste were transferred out of the building. During the same period, the amount of low level mixed waste created was 14.8 ft3.

Decontamination methods used so far have proved effective as much of the contamination encountered has been removable. Considerable success has been achieved using the CO2 system for removable material decontamination. Fixed contamination has been effectively removed using grit blasting methods where feasible.

Plasma arc cutting has proven effective and quick, particularly, in sectioning stainless steel. The use of containments with HEPA filtration has contained airborne contamination well though the smoke encountered using plasma cutters has shown a tendency to block filters prematurely if not used with effective prefilters. Other size reduction methods have been effective though more arduous than plasma cutting.

WASTES

Waste arising from all Stage II operations has been estimated at 53,500 ft3 of solid low level waste with another 20,000 ft3 of solid low level mixed waste. By comparison, the project is expected to generate over 335,000 ft3 of materials suitable for recycling in Stage III which would have otherwise been disposed of as waste.

STAGE III OF THE NCPP

Stage III is initially planned as having a five year duration with an option for extension if viable. Products to be manufactured in Stage III will utilize the facilities, materials and workers used during Stage II. Intended customers for these commercially viable products will be the DOE and other government and private interests.

It is intended that prototypes developed and fabricated during process verification activities in Stage II will become suitable products during Stage III. Typical products being investigated include coffee-tin sized stainless steel containers suitable for plutonium storage and 100 ft3 steel containers for LLW shipment and storage.

FUTURE COMMERCIALIZATION OF DOE FACILITIES

The aim of the NCPP is to provide a model of how weapons facilities can be converted into commercially viable operations. The success of the Project with the associated transition from DOE Order regulatory control to a NRC license will determine whether this type of operation can be attempted at other DOE facilities.

ACHIEVEMENTS OF THE NCPP SO FAR

Though only at the start of Stage II, the NCPP has achieved the following:

Reduction of risk by removal of radioactive and toxic contaminants
Decrease of operational, maintenance and security costs by removal of redundant equipment
Close co-operation with State and Federal regulators by involvement in project steering committee Demonstration of privatization of DOE facilities, equipment/building reuse by private industry
Hiring of dislocated RFETS workers
Obtaining public approval before commencing work
Aggressive public involvement program including community meetings, building tours
Sharing of results with other DOE sites

CONCLUSIONS

The NCPP is the first project of its kind to try and convert contaminated ex-weapons facilities into commercially viable operations. The cleanup and refurbishment operations in Stage II are a fundamental part of this project. The Project uses the experience of decontamination and dismantling operations from other facilities in the planning and execution of cleanup methods at Rocky Flats.

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THE M-100 PROGRAM: A LOGICAL USE OF RADIOACTIVE SCRAP METAL (RSM)

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ABSTRACT

The concept of a consolidated effort to provide standardized waste containers for the DOE contractor network and the commercial nuclear industry has been discussed for some time. With the advent of the RECYCLE 2000 PROGRAM and the M-100 CONTAINER PROJECT, a serious effort is under way to develop a family of standard containers to be manufactured essentially from RSM, radioactive scrap metal.

The DOE's RECYCLE 2000 has as its mission: Produce 50% of the low level waste containers from RSM by the year 2000.

The headquarters initiative M-100 CONTAINER PROJECT has as its mission: Produce a family of containers to be manufactured from RSM that will satisfy 80% of the DOE's requirements. For the M-100 CONTAINER project to succeed, site consensus is essential.

This paper describes:

the M-100 family of containers
the methods used in gaining consensus from the contractor network
the determination of unique container requirements from each of the several disposal sites
the integration of such requirements into a universal design
the adaptation of unique closure and materials handling techniques

The integration of the RECYCLE 2000 PROGRAM, the BENEFICIAL REUSE PROGRAM, and the M-100 CONTAINER PROJECT will have a profound affect upon the availability of low cost, certified and documented containers as the DOE continues to clean its house over the next several decades. It is reasonable to assume that the commercial nuclear industry will judge the efficacy of the project and sign on as well. The eminent logic of constructing contaminated waste containers from contaminated steel cannot be denied. Ultimately, the cost benefit will accrue to the tax payer.

INTRODUCTION

The Department of Energy is exploring the task of converting its growing stores of radioactive metals into products useful to the nuclear community. The logical extension of such planning has been the development of a family of waste containers to be constructed essentially of radioactive scrap metal (RSM).

This paper presents three points which explicate the essential logic of the use of RSM for the construction of waste containers. The points at issue are:

1) RSM as a material for containers - the genesis of the concept
2) The development of the M-100 Container
3) The convergence of RSM, the M-100 Container and a dedicated manufacturing facility.

We believe that this paper will provide an understanding of this forward-thinking program and its benefit to the entire waste remediation effort.

RSM AS A MATERIAL FOR CONTAINERS - THE GENESIS OF THE CONCEPT

Congruent Needs and Concerns

The sites across the DOE network face essentially the same concerns in the remediation of waste. Each has waste products to be contained, scrap metal inventories and container needs. Logic dictates that an organized, common effort to deal with these concerns emerge.

The government owns countless tons of contaminated carbon and stainless steel material. These stores increase as the dismantling of site facilities continues. Demonstration projects are underway to determine the efficacy of returning such materials to a useful state in the form of sheet and plate. These projects will provide credible recycling and technology cost data which will be used as the basis for future judgments.

The waste remediation effort knows one essential waste container, the 4x4x6 "shoe box" shaped, carbon steel weldment. Millions of dollars are spent each year on such containers by all sites with waste packaging requirements. They are purchased from 30-50 different vendors...and they are all different.

However, the industry needs such containers. Each site enters into its own specification cycle, procurement cycle, testing cycle and ultimately, acquires sufficient containers to satisfy some remediation within some budget. The process has gone on for as long as the need was there...for decades. An alternative, innovative plan has been initiated by the DOE, its constituent elements known by many names: RSM, M-100, Recycle 2000, and Beneficial Reuse.

Compelling Logic

It makes economic and environmental sense to make products useful to the DOE network out of materials presently on site at DOE facilities.

It makes eminent sense to manufacture a standard family of containers that will meet the needs of all sites.

It makes economic sense to bring recycled materials to a dedicated facility to be fabricated into containers for use by the DOE contractors. The logic of rendering useful the increasing liability of contaminated scrap metal through recycling and its beneficial reuse cannot be denied. Contaminated scrap has no intrinsic value. Through the effective disposal techniques of recycling, radioactive scrap metal can be returned to its ultimate value and beneficial reuse.

The use of RSM as a container material carries with it familiar control practices well known to the industry and implies a facility licensed and dedicated to contaminated production.

It therefore follows that efficient use should be made of such facilities. A container family, designed to be produced in quantity, and a facility tooled to accommodate such containers defines synergism.

It is possible to create a basic container design which may satisfy the requirements of 7A TYPE A as well as STRONG TIGHT with such uniform and consistent features one to the other so as to greatly enhance manufacturing economies and permit just-in-time inventories. Such is the container to be produced by a dedicated RSM facility. The M-100 short list of containers of standard design and certification can be used by all sites to satisfy their needs.

The progression from recycled metal to standard design to dedicated facility assures consistent quality, uniform design and certified, low-cost containers.

From the short list catalog, each site may satisfy the large fraction of container needs. It is understood that site bias and on-going established programs will result in reticent attitudes toward a new program.

Change comes with difficulty but change also comes with information and education.

The goal shall be to make the container design the "standard of the industry." Such an effort may require a missionary zeal but the eminent logic, the beneficial costs, and container availability will probably win.

Recycle 2000 Program

RSM (radioactive scrap metal) may be defined as, "Those metals in inventory at the several DOE sites, classed as low-level waste which may be candidates for recycling."

The program "Recycle 2000" instituted by the Department of Energy has as its goal producing 50% of the DOE's low-level waste container needs from RSM by the year 2000.

In support of this goal a number of demonstration contracts have been awarded to define recycling techniques and demonstrate their efficacy.

The M-100 Development Program

The Department of Energy as a companion program to Recycle 2000 has issued a small development contract to provide waste boxes to be made essentially from RSM. The M-100 project has as its mission: to produce a family of containers to be manufactured from RSM that will satisfy 80% of the DOE's needs.

A unique container family has been designed, constructed and tested. All have identical external dimensions. All are designed for maximum storage efficiency. The container envelope contains a nominal internal volume of 90 cubic feet. Such volume permits a pay load of compacted soil equivalent to 9,000 lb.

In an effort to gain more universal usage, we have altered the familiar box geometry to permit the containment of a six-pack of standard 55

gallon drums. While the aspect ratio varies somewhat from the conventional box, we believe the added use benefit to be an important consideration.

The scope of work requires a container design in STRONG TIGHT and 7A in both 12 ga. and 7 ga. carbon steel sheet. The logic here is to establish a design that will permit those who are only able to produce heavier materials to participate in the M-100 program.

We approached the task on a worst-case basis, selecting 12 ga. material and a 7A test protocol. By building and testing a prototype in this configuration, we are assured that the heavier 7 ga. container will pass the 7A tests as well.

The 7A prototype (Model M-101/7A/12/90) is constructed with a system of internal struts so that it can sustain the vertical load as required by the NTS Waste Acceptance Criteria (120,000 lb. approx.) and the 7A drop test as required by the Code of Federal Regulations. (See Fig. 1)

Fig. 1

If we remove some of the struts from the prototype, the container is reduced to a STRONG TIGHT design capable of sustaining the NTS vertical load requirement but not the 7A drop test. (Model M-101/ST/12/90) (See Fig. 2)

Fig. 2

At least two of the major sites do not require the NTS static load capacity. By removing additional struts from Model 101/ST/12/90, we reduce the container to a system that will sustain a vertical load equal to five (5) times its gross weight (35,000 lb. approx.). Again the 7A drop test is not required. (See Fig. 3)

Fig. 3

This reverse progression, designing initially to the 7A standard, permits the development of a container family, essentially identical in external appearance, applying lesser standards where appropriate, while protecting manufacturing efficiency and economy.

Four (4) container models have been established as the M-100 series.

Table I

Gasket

The design technique of using a flexible gasket requires a pull down of the gasket sufficient to prevent a leak path. Care must be taken to prevent extreme compression such that the gasket cells are crushed causing the material to loose its resilience (compression set). We reason that such over compression must be prevented to protect the gasket material's flexibility.

The normal array for filled storage containers is to stack one upon the other. Individual site requirements vary from a stacking load of five (5) high at SRS (about 35,000 lb.) to the NTS overburden requirement of 120,000 (approx.). Such loading sustained by the lower most container may severely crush the cells of the gasket if the container design does not protect it. If one unstacks the containers for retrieval, the crushed gasket has been rendered ineffective and no longer provides an environmental barrier.

The M-100 container is constructed such that the vertical stacking load is taken through the cover and the cover bolts and transferred to the box wall. The gasket sees no severe compression no matter what the storage conditions may be.

Fasteners

Among the criteria used in selecting a closure system must be:

- ease and convenience of use
- a requirement for no special tools
- in process opening and closing
- temporary and permanent closures
- absence of proprietary design infringement
- cost

Bolting is the simplest, most universally accepted method for mechanically fastening one or more parts. Simple is best. The Rivnut/Bolted Closure has precedent in container designs presently in use at the DOE sites. Rivnuts (or their equivalent) are used standardly by WIPP in the Standard Waste Box. K-25 has a patented box which uses a series of 38-bolts and Fernald uses a bolted flange for its 6-drum box. To close the container and compress the gasket, one must do the following:

- beginning at the corners, insert a drift pin through the cover hole and into the threaded Rivnut
- skip a hole and repeat the process
- place a bolt in the skipped hole
- repeat at all corners
- fill in the gaps

For temporary or in process closure, one need place only three or four bolts on a side.

The M-100 container series is designed with an off-set at the upper perimeter. Such a system affords the fasteners protection during test and random shipping events. In addition, the off-set provides space for a unique apparatus which can lift the container cover alone or, with the cover in place, the container and its payload. (See Fig. 4)

Fig. 4

Aside from top fastening containers such as the K-25 7A box, most closures require some space beyond the container wall to accommodate the closure fittings. The M-100 design permits absolute side by side, zero clearance, storage between boxes. The lifting apparatus which permits hoisting from the top of the container obviates the need for fork lift risers.

Risers

A compelling case can be made for inexpensive, removable or disposable fork-lift risers. Conventional, permanent risers permit a void space on the order of 4-5 cu. ft. per stored container. A container without risers can be stacked one upon the other with no clearance. For the present, the disposal sites require permanently affixed risers. Such risers are part of the M-100 designs. We postulate, however, that the industry will ultimately accept the concept of removable or disposable risers as its standard.

Burying Money

The divergent approach to container requirements among the several sites is curious. We have seen a proliferation of container designs with less than desirable storage efficiencies - that ratio of inside volume to outside cubic measure. Little attention has been given to void spaces created by closure appurtenances and risers. Literally all of these void spaces can be eliminated through innovative design. A properly designed container may reduce void space in the repository by 10-12 ft3 per stored container. Translating this to dollars at $45.00 per ft2 equals $540.00 in void space - a sum which will more than pay for the container.

The predominant use of fork lifts as the principle transporter of waste containers has necessarily placed limitations upon external container design. This fact together with an industry unwillingness to accept removable, recyclable or disposable risers causes the storage inefficiency to be preserved.

An absolutely smooth-sided, cubic container with neither appurtenances nor fork lift risers fitted with a top-lift hoisting arrangement will minimize the money buried with each container.

An indicator of storage efficiency is the ratio of internal to external volume.

M-101/7A/12/90 efficiency 93%
M-101/ST/12/90 efficiency 95%
M-101/ST/12/90/A 97%
M-101/ST/7/90 97%

The Task Force

To assist with the container definition and to maximize the contribution of ideas and concerns from the sites, a task force of twenty eight (28) people was assembled representing all of the major sites and many of the DOE offices. No funding existed for travel and site visits. With few exceptions, all communication with the task force and the DOE took place via fax or phone. Over time and with the influence of the task force, a single design emerged which has the probability of meeting the container needs of all the sites.

The task force contributions in many ways became a study in conflict resolution.

LMES Oak Ridge required a boxed riser
Fermco insisted upon a wide flange beam riser
None would agree to a removable or disposable riser
NTS ultimately decided to consider removable risers in the next revision to the waste acceptance criteria (WAC)

LMES strongly suggested that any cover lifting device must be capable of lifting the container and its contents. This extremely challenging requirement permitted a vast improvement in storage efficiency, allowing containers to be stored one upon the other without fork lift risers.

Yet to be resolved is the material thickness issue. Ten gauge steel is the requirement of at least one (1) major site. The M-100 container is constructed of 12 ga. or 7 ga.

SRS has an absolute geometry requirement for its low-level waste containers. Other sites do not. The M-100 container continues to be at odds with the SRS standard.

Three (3) members of the task force represent SRS. Two (2) are in active agreement with the M-100 effort, yet SRS has released a five year blanket order agreement (BOA) with no consideration of the M-100 effort.

Much of the waste will ultimately reside at NTS stored in STRONG TIGHT containers. Such containers require a 120,000 lb stacking load

A large fraction of the waste will be stored at SRS and LMES in STRONG TIGHT containers. Such containers require 35,000 lb stacking load.

The disconnects and the discontinuities continue. Such divergent requirements were aired through the M-100 container task force teleconferences. While the M-100 container will not be "all things to all people," we believe it to be sufficiently well considered to become the industry's standard and satisfy 80% of the container requirements.

The DOE, for its own reasons, selected a small firm to execute the M-100 task. Its support and direction have been motivating. It is, however, the

contributions from the task force members at the sites that has given the design its credibility. While the significant task (primary job) of gaining consensus remains, the task force approach with its early contribution assures a high probability of such agreement.

THE CONVERGENCE OF RSM, M-100 AND A DEDICATED FACILITY

The Next Step

The M-100 container designs have been constructed and tested and await that first opportunity for a limited "real world" production run. That opportunity may come in the form of an arrangement between Fermco and Oak Ridge wherein RSM will be shipped to LMES but returned to Fernald in the form of finished containers to the M-100 design. This opportunity will provide valuable information regarding fabrication techniques and manufacturing cost data.

If a new product is to be introduced to the market place, the essential step after constructing engineering prototypes must be value analysis; that process of evaluation which permits cost reduction without compromising the design. As the Fernald requirement evolves, such an analysis should be conducted.

The Pitfalls

The Department of Energy initiated the Recycle 2000 and the M-100 programs and has effectively managed them through the early rocks and shoals. Demonstration containers have been made from RSM. Recycling facilities are underway and the first production order is in the offing. Such is the early progress of which success is born.

To every up side there is a down side, however. The down side or pitfall in this project's road is apathy, indifference, inertia and that attitude institutionalized at many sites which says, "We have our own program. Our containers are different. This is the way we have always done it."

With the advent of RSM as a useful and available material for container fabrication and the acceptance of the M-100 design, it is reasonable to consider a fabrication facility dedicated to the task of producing containers for the waste remediation effort. Logistical considerations would suggest two facilities: one in the east, probably mid-south and one in the west. Such facilities will be planned and tooled to provide just-in-time shipment of M-100 containers from stores of inventoried containers. Since the shell components of the M-100 containers are identical (walls, struts, covers, etc.) sub-assemblies can be fabricated well in advance. The M-100 logic provides for the multiple use of components and a "building block" approach to container fabrication. Properly equipped and strategically located, such facilities can produce the ultimate container value.

CONCLUSION

We have addressed the issues of radioactive scrap metal (RSM) as a container material, described in detail the emerging M-100 container program and introduced the convergence of RSM, M-100 and the dedicated facility.

While a serious effort has been made to include the sites through task force involvement, an early and on-going effort is needed to gain acceptance to the point of implementation of the M-100/RSM containers. The compelling logic of the Recycle 2000/M-100/dedicated facility movement requires re-thinking the conventional procurement cycle, design and test cycle, and contractor/vendor interaction. It requires a paradigm shift - a new model, a new way of thinking, a collaborative effort which shall assure the most efficient use of the government's stores of

radioactive scrap metal to produce M-100 containers of high quality and the greatest value to the tax payer.

Session 14 -- WASTE MINIMIZATION/POLLUTION PREVENTION STRATEGIES AND IMPLEMENTATION SUCCESSES
Co-chairs: Patricia J. Robinson, (n,p) Energy, Inc.
14-2
WASTE MINIMIZATION VALUE ENGINEERING WORKSHOP FOR THE LOS ALAMOS NATIONAL LABORATORY OMEGA WEST REACTOR DECOMMISSIONING PROJECT

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ABSTRACT
The Los Alamos National Laboratory (the Laboratory) Pollution Prevention Program Office sponsored a Value Engineering (VE) workshop to evaluate recycling options and other pollution prevention and waste minimization (PP/Wmin) practices to incorporate into the decommissioning of the Omega West Reactor (OWR) at the Laboratory. The VE process is an organized, systematic approach for evaluating a process or design to identify cost saving opportunities, or in this application, waste reduction opportunities. This VE workshop was a facilitated process that included Department of Energy representatives and a team of specialists in pertinent areas, including of decontamination, decommissioning, PP/WMin, cost estimating, construction, waste management, recycling. The uniqueness of this VE workshop was that it used an interdisciplinary approach to focus on PP/WMin practices that could be included in the OWR Decommissioning Project plans and the specifications to provide waste reduction.
The OWR Decommissioning Project is currently in the planning and preliminary assessment phases. Preliminary waste projections were estimated, and the Laboratory's Solid Radioactive Waste Management Group established a maximum volume of waste from the OWR that will be accepted for disposal. The decommissioning of a nuclear facility, such as the OWR, is expected to generate very large quantities of waste, including low-level radioactive waste (LLW), low-level mixed waste (LLMW), and hazardous waste (HW). Many of these waste materials (e.g., concrete, steel, wood, soil) may have a potential for recovery, recycle, and reuse. Because the high cost of waste management and the limited capacities for treatment, storage, and disposal of HW, LLW, and LLMW, all practical efforts must be used to reduce the volume of generated waste.
The VE team evaluated the Preliminary Decommissioning Project plans and recommended specific PP/WMin work practices that can be incorporated into the final Project plans, operations, and contract specifications to reduce the volume of generated waste and increase material recovery opportunities. The VE team considered decontamination, source reduction, recycling, and volume reduction techniques and technologies that are currently available at the Laboratory, within the Department of Energy, and in the commercial private sector. Emerging technologies or research

and development technologies were not considered. VE workshop results included the following:

- Over 13 recommendations and action items identified
- Specific PP/WMin practices that can be immediately incorporated to result in an estimated waste volume reduction of 2,399 m3 of LLW, sanitary waste, and Toxic Substance Control Act waste and save approximately $1.17 million in waste management costs
- Potential PP/WMin practices that could reduce an additional 1,251 m3 of LLW

INTRODUCTION

The Pollution Prevention Program Office (P3O) at the Los Alamos National Laboratory (the Laboratory) pilot tested an effort to integrate individuals with pollution prevention and waste minimization (PP/WMin) expertise into environmental restoration (ER) and decommissioning programs. The objective of the integration was to demonstrate that PP/WMin techniques can be (and have been) applied to the Laboratory's ER and decommissioning activities to minimize the volume of waste that requires subsequent treatment, storage, or disposal and to reduce waste management costs. In addition, the effort identified potential tools and systematic approaches for reducing wastes from ER and decommissioning activities.

As part of this effort, a Waste Minimization Value Engineering Workshop was conducted to focus on reducing wastes and recycling materials from the planned decommissioning of the Laboratory's Omega West Reactor (OWR). The Value Engineering (VE) workshop was sponsored by the Laboratory's P3O and the Department of Energy (DOE) Office of Environmental Management, in cooperation with the Laboratory's Environmental Restoration Program, Decommissioning Project. Benchmark Environmental Corporation planned and participated as team leader in the workshop; and the workshop was lead by an independent, certified VE facilitator.

The VE workshop evaluated the preliminary project plans for the OWR Decommissioning Project and recommended over 13 specific PP/WMin work practices that can be incorporated into the project plans and specifications to reduce the volume of generated waste. This paper presents the results of the Waste Minimization Value Engineering Workshop and discusses the use of VE as a tool to minimize the waste generated from future DOE decommissioning projects.

BACKGROUND

The Laboratory's Environmental Restoration Program assesses and cleans up sites and facilities that have been contaminated from past DOE activities, including the safe decontamination and decommissioning of inactive nuclear facilities. ER activities, in general, and decommissioning activities in particular, have the potential to result in large quantities of low-level radioactive waste (LLW), low-level mixed waste (LLMW), and hazardous waste (HW). There is a challenge to reduce (or avoid) waste that will require subsequent treatment, storage, and disposal for several reasons, including limited on-site and off-site capacities for waste treatment, storage, and disposal and the high cost of radioactive and hazardous waste management.

In 1995, the P3O began integrating with the Laboratory's Decommissioning Project Office (DPO) to incorporate waste reduction practices, where appropriate, and to evaluate existing tools or systematic approaches for incorporating PP/WMin as a standard practice in future projects. Jointly, the P3O and the DPO identified VE as a potentially effective tool for

decommissioning projects to identify and incorporate PP/WMin practices. The OWR Decommissioning Project was selected as a test case for the VE process.

The OWR Decommissioning Project was selected because it was in the planning and preliminary assessment stages and the project was expected to generate large quantities of waste, including radioactively contaminated soil, equipment, building debris, concrete, scrap metal, and personal protective equipment (PPE). It was then proposed to conduct a VE workshop on the OWR Decommissioning Project that would focus on recycling and waste minimization practices that could be incorporated into the project plans and specifications.

VE WORKSHOP OBJECTIVES

VE is an organized, systematic approach for evaluating a process or design to identify cost saving opportunities. The process uses functional analysis to challenge the designers, engineers, and project planners to consider alternative approaches that provide the same function. VE has traditionally been applied to engineering design projects in the conceptual, Title I, and Title II phases, and it has provided significant return-on-investment (ROI) of the costs spent to conduct the study and implement the VE recommendations.

The VE team included decommissioning representatives, PP/WMin representatives, cost estimators, construction specialists, decontamination specialists, waste management, recycling specialists, and other appropriate technical personnel.

The primary objective of the VE workshop was to identify and recommend specific work practices (e.g.,segregation) and PP/WMin practices that could be incorporated into the Decommissioning Project plans and operations to reduce the volume of generated waste and increase material recovery opportunities. The specific objectives included the following:

- Identify and prioritize specific decommissioning activities that are expected to generate waste
- Identify recycling opportunities and work practices that will reduce waste or increase material recycling
- Identify available technology or services needed to implement the PP/WMin practice
- Develop draft language for specifications or procedures to guide implementation of the waste minimization or waste reduction ideas
- Develop preliminary cost estimates for the practices with the highest potential and evaluate, by ROI or other cost analysis, which are the most feasible
- Recommend PP/WMin practices that the OWR Decommissioning Project should implement

The workshop focused on minimization, decontamination, and recycling concrete, steel, metals, and soil, which are expected to be the highest waste volumes. The VE team considered recycling, decontamination, and volume reduction techniques and technologies that are currently available at the Laboratory, within the DOE, or in the commercial private sector. Emerging technologies or research and development technologies were not considered.

Preliminary documents reviewed by the VE team included the Preliminary Project Plan (1), the Preliminary Characterization Report (2), and the Preliminary Waste Management Plan (3). Preliminary waste projections were estimated and the Laboratory's Solid Radioactive Waste Management group established a maximum volume of waste that will be accepted from the OWR.

Waste acceptance may be contingent on the expansion of the current on-site disposal area.

OWR DECOMMISSIONING PROJECT SUMMARY

The OWR Facility was originally constructed in 1943 and housed five nuclear reactors between 1943 and 1995. The first nuclear reactor was a low-power, water-boiler reactor and was the first reactor in which enriched uranium-235 was used as fuel to achieve a self-sustaining nuclear reaction. After several conversions, the final water-boiler reactor was completely shutdown in 1974 and dismantled in 1989. A 25 kW, fast-neutron research reactor, which used plutonium fuel surrounded by mercury coolant, was constructed on-site and brought to full power in 1949. This reactor was dismantled in 1954, after discovery of a fuel element failure leading to plutonium contamination of the mercury coolant (1, 4).

The final reactor, the OWR, was built on top of one of the previous water boiler reactor foundations and is still present. The OWR is a tank-type reactor, which has a full power rating of 8 megawatts (MW) thermal. The reactor is light-water moderated and cooled, and uses aluminum clad, MTR-type fuel elements. A coolant leak in an underground pipe was identified in 1992 and the OWR was shut down. That pipe and another pipe have since been removed. The remaining pipe stubs connected to the reactor are welded shut or capped. The OWR has not operated since discovery of the leak (1).

In 1994 the fuel and all control blades were removed from the OWR, and the OWR was placed in a safe shutdown mode. The reactor vessel has been drained of all coolant. During defueling operations, the fuel elements were inspected and no damage had occurred.

Decommissioning activities will address the ancillary buildings, facilities, and equipment; remove the reactor vessel and biological shielding; and survey, decontaminate, and demolish the main building. Decommissioning activities will include asbestos abatement; detailed radiation surveys; decommissioning of contaminated components and separation of radioactive and nonradioactive materials; detailed radiation survey for material or equipment release; demolition or disposal of material or equipment; and backfill to grade. After decommissioning activities are complete, the site will be turned over for Resource Conservation and Recovery Act facility investigation and corrective measure activities (1).

A preliminary characterization was performed in early 1995. The OWR characterization is consistent with the operation of a nuclear reactor that used enriched uranium as a fuel. Activation products, radioactive contamination, and primary coolant isotopes include cobalt-60, nickel-59, nickel-63, strontium-90, and cesium-137. Radiological contamination includes high levels of radioactivity internally located at the fuel element grid plate and nearby ports in the reactor tank (i.e., 2600 R/hr, primarily activated cobalt-60 in experimental ports). Contaminated areas include the following: reactor tank tops; building roof; concrete-capped floor; painted areas (fixed contamination); ion exchange resin, pumps, filters, and piping; primary surge tank; three underground storage tanks; cooling tower heat exchanger air handling blower and exhaust stack; and building walls and floors (less than 5000 dpm/100 cm2). Contamination inside the primary system, lead shielding, and foundation concrete is unknown (2).

Projected waste generation from the Preliminary Decommissioning Project plans is shown in Table I. No recycling or waste minimization requirements were identified for the project specifications; however, the following, ten, very general, PP/WMin practices were identified in the preliminary plans (3):

Conduct routine briefings
Segregate wastes to avoid mixing and cross-contamination
Remove contamination and reuse equipment and supplies
Remove visible and radioactive contamination from disposable items before discarding
Avoid the use of organic solvents during decontamination
Use drip, spray, squirt bottles or tanks for decontamination rinses
Use impermeable materials such as plastic liners or mats and drip pallets to prevent the spread of contamination
Practice contamination avoidance
Reduce waste volume
Consider waste treatment and recycling operations

Table I

VE WORKSHOP ACTIVITIES AND RESULTS

The steps for planning and conducting the VE workshop are shown in Fig. 1. The VE workshop was conducted over three days and followed a typical VE job plan as outlined in Fig. 1.

Fig. 1

The VE team developed over 13 recommendations that have a high potential for waste reduction and should be included in the Decommissioning Project plans. The recommendations are discussed below.

Seven recommendations can be immediately implemented that will reduce waste generation by an estimated 2,399 m3 (778 m3 of LLW, 1,621 m3 of sanitary waste, and 0.4 m3 of Toxic Substance Control Act (TSCA) waste), with an estimated project savings of $1.17 million in avoided waste management costs.

Two recommendations for radioactively contaminated concrete could potentially avoid 501 to 1,251 m3 of LLW concrete (saving $0.79 to $1.98 million in waste management costs). These recommendations were considered to be technically feasible and have very high potential for LLW reduction; however, they required longer implementation and negotiation efforts to resolve potential regulatory acceptance barriers.

Four recommendations are applicable to reduce or avoid LLW, but are nonquantifiable from a waste avoided and cost savings perspective.

- Dedicate one person with authority to make waste management, minimization, and recycling decisions and to provide technology direction or assistance, as necessary.
- Build incentives into the contract for reaching specific source reduction, volume reduction, and recycling goals (i.e., contractor keeps proceeds from recycle; media-specific incentives and goals are established; bonuses for project managers are provided; and waste management dollars are given to project up front).
- Change Laboratory and DOE rules so that consistent interpretation of rules is possible throughout projects. For example, allow free release of waste below limits; allow wasteform averaging; allow the use of explosives; set realistic waste acceptance standards for disposal.
- Include specific requirements in project specifications in requests for proposal. For example, require project-specific waste minimization and salvage plans; disallow disposal of specific recyclables; require

volume reduction before waste packaging; specify how waste will be characterized to meet criteria for disposal.

The nine quantifiable recommendations are identified in Table II. Cost savings include only project- related waste disposal costs; the entire cost to the DOE complex was not quantified.

Table II

The VE workshop findings and recommendations were presented to Laboratory program managers (i.e.,ER, decommissioning, waste management) and DOE representatives. There were no major objections to the recommendations. Admittedly, some might be difficult to implement, but none were identified as entirely unreasonable or impracticable. One raised issue was that this workshop did not identify any new techniques, but only identified practices that currently existed at different locations and for different on-site projects. The VE team acknowledged that fact, but felt that the information might not have been as quickly and effectively shared with the OWR Decommissioning Project personnel if left to chance and not gained through a formal process like VE. Without such an organized formal approach, the pressure of a project schedule, other work load pressures, and personal agendas often combine to inhibit such sharing and application development.

CONCLUSIONS

On the basis of the results from the VE workshop and the management briefing, it was concluded that VE is an effective tool for incorporating PP/WMin into decommissioning projects. Further, the VE workshop was beneficial to the participants and to the OWR Decommissioning Project. However, improvements can be made to the VE workshop process. The process appears to be most effective for large projects (e.g., high waste generation, high cost).

As a PP/WMin tool, the VE process was effective in developing specific ideas for reducing waste, including LLW and sanitary waste, and reducing (or avoiding) waste management costs. In addition, the process increased awareness of PP/WMin requirements and practices, and it enhanced communication between the Laboratory's decommissioning project and the P3O because it provided an opportunity for the interested parties to influence practices that are integral to their day-to-day responsibilities.

Over 70 percent of the VE team reported that the workshop was valuable, and they would recommend using the process again on a larger-scale with respect to waste generation and cost. Over 60 percent believed waste would actually be reduced as a result of the workshop's recommendations, if implemented; the other participants believed it was too early to predict the actual benefits. The cost savings identified during the VE workshop far exceed the cost of planning and conducting the workshop; and the cost for implementing the VE recommendations is expected to provide a high ROI in terms of avoided waste management cost to the OWR Decommissioning Project.

Lessons learned from the workshop are summarized below and should be considered in the planning of future VE studies.

- The process can be streamlined to shorten the length of the workshop
- The process can be combined with existing project technical review milestones or project planning and design meetings
- Project management commitment is needed to ensure followup to the recommendations developed during the workshop

Active participation by all interested parties must be encouraged (or enforced)

More advance notice should be given to participants, and complete background data should be provided before the VE workshop

In conclusion, VE can and should be used for large waste volume decommissioning projects as a tool to ensure that the decommissioning activities are planned and evaluated with an eye toward PP/WMin and that decisions made with respect to PP/WMin are documented.

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14-3

POLLUTION PREVENTION AND WASTE MINIMIZATION AT THE WATERVLIET ARSENAL THREE SPECIFIC EXAMPLES

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ABSTRACT

MSE Technology Applications, Inc., (MSE) is a Department of Energy (DOE) contractor and operates the Western Environmental Technology Office (WETO) for DOE'S Office of Scientific Development (OSD).

In this capacity, MSE not only evaluates innovative technologies for application in environmental restoration and pollution prevention but also actively promotes the transfer of these technologies to a broad spectrum of users.

The Watervliet Arsenal, located outside Albany, New York, in the city of Watervliet, is a site where several technologies developed in the private and public sectors are being implemented to prevent pollution and reduce waste generation.

A description of the technologies will be provided with a discussion of the approach used to select and implement the appropriate technology. Technologies that will be discussed include: state-of-the-art Coolant Management Systems used to clean and recycle coolants used in large-scale machining operations, application of anaerobic bacteria to reduce toxicity of chromium-contaminated waters, and utilization of a systems approach combining technologies from two vendors to treat chromium-contaminated water.

In addition to brief examinations of the three specific technologies, the approaches used to bring together several diverse participants to form an effective team and implement technology transfer will be outlined. This work is being performed under the DOE Contract number DE-AC22-88ID12735

and is funded by the Department of Defense (DOD) under the supervision of the U.S. Army Corps of Engineers, Construction Engineering Research Laboratory (USA/CERL) with the cooperation and support of the Watervliet Arsenal.

INTRODUCTION

Background

In May 1994, MSE Technology Applications, Inc. met with U.S. Army Corps of Engineers' Construction Engineering Research Laboratory (CERL) representatives to discuss potential projects for industrial wastestream remediation and minimization at the Watervliet Arsenal in Watervliet, New York.

Ten separate pollution control or waste minimization projects have been initiated as a result. In the following paper a summary of three of the technology demonstrations currently in progress will be presented. These technologies are focused on two of the Arsenal's major waste streams: 1) contaminated machine tool coolants; and 2) chromium contaminated water form hard chrome plating operations.

PROJECTS

Machine Tool Coolant Recycle.

Oil based coolants used to cool machine tools frequently become contaminated with grease and dirt from the components being machined, and also with oils that may leak from the machine tools themselves. Additionally, the quality of the coolant can be degraded by replenishing evaporative losses with hard water resulting in poor quality coolant emulsion. The approach to solving this problem in one major manufacturing area that utilizes a 40,000 liter coolant sump to supply a number of large machine tools was to provide reverse osmosis water for coolant preparation and replenishment, and install a system to remove tramp oils from the coolant.

Chrome Plating Rinse Water Recycle

Gun tubes manufactured at the Watervliet Arsenal are chrome plated in large 12 meter deep tanks. Following the plating process the tubes are rinsed in tanks of fresh process water. When the chrome level in the rinse water exceeds approximately 100 mg/l, the concentration is lowered by dilution with the effluent (tank overflow) being sent tot the onsite industrial waste treatment plant for chrome removal prior to discharge. The pilot scale demonstration evaluated a system comprised of two vendors' technologies for recovery of hexavalent chrome from the rinse water. The system consisted of electrochemical extraction of hexavalent chromium, effluent particulate microfiltration and reverse osmosis final polishing.

Treatment of Chromium Wastestreams using Sulfate-Reducing Bacteria

The use of sulfate-reducing bacteria to mitigate hexavalent chromium-contaminated waste water at the Arsenal is also being evaluated. A bench-scale bioreactor system is being operated and results evaluated at the Department of Energy Western Environmental Technology Office (WETO) in Butte, Montana.

TECHNOLOGY EVALUATIONS

Machine Tool Coolant Recycle

Coolant in the large (40,000 liter) sump at the Arsenal's Flexible Manufacturing area typically becomes contaminated with tramp oils and bacteria. This coolant must be replaced at least annually or at times more frequently, and the contaminated coolant must be shipped offsite for disposal as a special waste. By supplying demineralized make up water and

providing a system for removing tramp oils and bacteria, an indefinite coolant life is anticipated.

A reverse osmosis unit was installed in the facility to supply demineralized make up water and a new batch of coolant has recently been prepared using 20 parts water to 1 part mineral oil. The demonstration is continuing to evaluate both the coolant quality and the effectiveness of the tramp oil and bacteria removal system.

Selection of a coolant recycle system was accomplished by developing performance specifications and soliciting proposals from a number of vendors. Proposals were evaluated on the basis of cost and effectiveness resulting in selection of a Sanborn Pioneer Coolant Recycle System.

The Sanborn Pioneer System was installed in the Flexible Manufacturing area of the Arsenal, and a four-week proof-of-process test was initiated on September 15, 1995. During the course of this test, coolant samples were collected and analyzed to evaluate the systems effectiveness in reducing the bacteria count and removing tramp oils from the coolant. Bacteria count was typically reduced from 5.0 X 107 colony forming units/ milliliter (cfu/ml) to less than 10 X 103 cfu/ml by the pasteurization process. The bacteria count in the sump was reduced to a minimum of 9.5 X 104 cfu/ml, but typically remained in a range of 2 X 105 to 4 X 106 cfu/ml.

Initial tramp oil content in the coolant was 4 to 5 percent by volume. After 2 weeks of operations, the tramp oil content was reduced to 0.5 percent. As Fig. 2 illustrates, the tramp oil content then oscillated as a series of major hydraulic leaks on machine tools leaked large volumes of oil into the sump.

Chrome Plating Rinse Water Recycle

The demonstration to treat chromium contaminated rinse water took place at the 120mm Chrome Plating Facility on the Watervliet Arsenal, Watervliet, New York.

The demonstration evaluated the effectiveness and cost of hexavalent chromium recovery and rinse water effluent treatment compared to current Watervliet Arsenal methods of handling chrome rinse water.

The two vendors involved with the demonstration were the Memtek Division of Wheelabrator and their Retec Chrome Recovery System and ZENON Environmental Incorporated and their Microfiltration/Reverse Osmosis Treatment System. The two technologies were connected in series to a 2800 gallon chrome rinse tank for hexavalent chromium recovery, filtration, and final polishing. (Fig. 2) The Eltech Retec chrome recovery system demonstrated electrochemical extraction of hexavalent chromium from the rinse water. The ZENON Microfiltration (MF) and Reverse Osmosis (RO) units demonstrated effluent particulate microfiltration and final polishing respectively.

Seven sample streams were analyzed during the demonstration. The seven sample streams were:

1) RTE (Rinse Tank Effluent): Process water prior to any treatment processes;

2) CO (Catholyte Outlet): Rinse Tank Effluent after electrochemical treatment in the Retec Cell;

3) Anolyte Outlet: Electrolyte solution for the Retec Cell. Anolyte Outlet is recirculated through the Retec Cell and acts as a heat sink and reservoir for the recovered chromium;

4) MFP (Microfiltration Permeate): Catholyte Outlet after microfiltration treatment;

5) MFC (Microfiltration Concentrate): Stream containing solids from Catholyte Outlet Stream;
6) ROP (Reverse Osmosis Permeate): Polished process water after reverse osmosis treatment; and
7) ROC (Reverse Osmosis Concentrate): Reject Reverse Osmosis process water.

These streams were sampled, when applicable, on a daily basis and analyzed for Cr, Fe, and Pb. Iron and lead were below instrument detection limits on virtually all samples. Bent Laboratories on the Watervliet Arsenal performed all analytical work associated with this project.

The three technologies were installed as an end-of-pipe (2) treatment "system". Since the treatment technologies represent two separate companies, each treatment was evaluated individually wherever possible. The effectiveness and maintainability of each unit was evaluated on an individual basis. The economic viability of each technology was evaluated separately and as a part of the entire "system".

The Retec Cell was successful at removing chromium from the rinse water when the power source, rectifier, was supplying a high DC amperage to the cell and/or the flow through the cell was low (0.5 - 1 gallons per minute). Single pass recovery rates of approximately 70-99% were observed under these conditions. However, these recovery rates could not be maintained for more than a few days. Cleaning the unit, a 1.5 hour procedure, brought single pass recovery rates back up to desirable levels. The Retec Cell was simple to use and easy to maintain. Total operational costs for one year are estimated at $7,500.

The ZENON Microfiltration Unit could not be evaluated for particulate microfiltration effectiveness due to delays in receiving analytical results. The microfiltration unit was easy to operate and maintain and total operational costs for the microfiltration unit for one year are estimated at $4,800.

The ZENON Reverse Osmosis Unit was successful at final polishing of the rinse tank effluent. Percent rejection rates of 99% and an overall decrease in the conductivity as illustrated by Fig. 3 of the 2800 gallon rinse tank were observed as a result of using this unit. Total operational costs for one year for the Reverse Osmosis unit are estimated at $65,000. Ninety three percent of the total operational costs of the unit are associated with cooling water (4.5 gpm) required for operation.

Fig. 1

Fig. 2

Fig. 3

TREATMENT OF CHROMIUM-CONTAINING WASTESTREAMS USING SULFATE-REDUCING BACTERIA

MSE is performing laboratory-scale experiments to determine the effectiveness of using Sulfate-Reducing Bacteria (SRB) to reduce Cr(VI) concentrations in industrial wastewater. This project is being conducted as part of a group of projects being performed at MSE called the Department of Defense Construction Engineering Research Laboratory Projects (Industrial Waste Stream Pollution Prevention Projects) which address concerns at the Watervliet Arsenal (WVA) in Albany, New York.

The acid wastestream produced at the WVA contains a significant concentration of Cr(VI) (29 mg/L on average). Largely because Cr(VI) is a carcinogenic substance, it is desirable to reduce the Cr(VI) to Cr(III), a much less toxic substance. Presently, WVA is reducing Cr(VI) to Cr(III)

in the wastewater by adding sulfuric acid and other reagents to promote the following chemical reaction:

$2CrO_4^{2-} + 3SO_2 + 2H_2SO_4 \rightarrow Cr(III) + 5SO_4^{2-} + 2H_2O$

The Cr(III) is subsequently precipitated as a hydroxide: however, sulfuric acid is an expensive, dangerous reagent.

Through the laboratory experiments, MSE is evaluating the use of SRB to treat chromium-containing wastewaters. SRB are naturally occurring, heterotrophic, anaerobic bacteria, which are found in nearly all natural environments on earth. Sulfate-reducing bacteria decompose simple organic compounds using sulfate as a terminal electron acceptor, thus producing hydrogen sulfide, H_2S. The testing is being performed within several batch bioreactors in a laboratory at the WETO facility located in Butte, Montana. The bacteria are cultured and injected into bioreactors containing Cr(VI)-containing wastewater and are nourished with a low-cost organic substrate. The H_2S needed for the reduction of Cr(VI) is produced by the SRB. The use of biogenically produced H_2S is safer and more cost effective than the traditional H_2S or sulfuric acid sources. Cr(VI) is reduced through the following chemical reaction:

$3HS^- + 6FeSO_4 + 4CrO_4^{2-} + 13H_2O + OH^- \rightarrow 3S + 6Fe(OH)_3 + 4Cr(OH)_3 + 6SO_4^{2-}$

The primary objective of the laboratory-scale testing is to determine if SRB are effective in reducing Cr(VI) concentrations in synthetic solutions (initial concentrations varying from 25 to 55 ppm) and in WVA waste streams (initial concentration approximately 25 ppm). The tests will also determine if SRB, once established, can withstand a sudden increase in Cr(VI) concentration (an intervention event).

Secondary objectives of this project are to evaluate the treatment system for technology effectiveness and cost effectiveness. This information will allow the project personnel to assess the applicability of the technology for treatment of the WVA wastewater at field scale and the manner in which the technology should be applied.

Preliminary experimental results indicate that SRB are capable of reducing Cr(VI) to Cr(III). Figure 4 presents Cr(VI) concentrations over time in a reactor with an initial Cr(VI) concentration of 25 ppm.

Fig. 4

CONCLUSIONS

Machine Tool Coolant Recycle

Initial test results indicate that machine tool coolant life can be extended to at least several years or more, in contrast to the past practice of annual or more frequent change out. This results in an annual savings of approximately $21,000 comprised of disposal costs ($10,000), coolant purchase ($5,000), and labor ($6,000). There are additional less tangible benefits including: 1) reduced odor which creates a better work environment; 2) reduced incidence of dermatitis among machine operators, 3) reduced machine maintenance, 4) increased cutter tool life, and 5) improved product quality. Although several years of documented operating experience will be required to fully validate the above benefits, all results to date strongly suggest significant cost, quality, and environmental benefits will be realized through installation and use of the coolant recycle system.

Chrome Plating and Rinse Water Recycle

The treatment system for recovering chrome and cleaning the chrome-plating rinse water wastestream was not cost effective. Although the Retec Cell was effective in removing chrome from the rinse water, there was not sufficient chromium available for recovery to offset the

electrical costs. The cell would be best utilized to treat a wastestream with significantly higher chromium concentrations than those in the final rinse tank. (<75ppm)
The microfiltration reverse osmosis units were employed to clean the chrome rinse water for either recycle to the final rinse tank or for discharge. However, the energy consumption by the two units exceeded the benefits derived from reduced make up water requirements and reduced waste water treatment costs at the existing on-site industrial waste treatment plant. This situation prevails even if the RO cooling water is recycled to the plant process water system.
Installation and operation of treatment systems that do not reduce costs relative to existing systems, is not justified under current conditions. In the event a plant were required to operate as a zero-discharge-facility (regardless of the cost), then use of microfiltration and reverse osmosis to clean process water for recycle could be justified.
Treatment of Chromium-contaminated Wastestreams Using Sulfate-Reducing Bacteria
At the time of this publication, testing and evaluation of SRB effectiveness is still at the laboratory scale, and a cost-benefit analysis has not been performed.

14-4
PROMOTING A CULTURE CHANGE TOWARD POLLUTION PREVENTION AT THE SAVANNAH RIVER SITE
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ABSTRACT
The U.S. Department of Energy has established a pilot team to implement pollution prevention activities at the Savannah River Site. Over the past year, this pilot team has worked to build a strong foundation for the site-wide pollution prevention program. This program covers over a thousand waste stream at more than one hundred facilities. The four primary building blocks used by the pilot team for initiating its program include: 1) developing a site-wide pollution prevention policy and manual; 2) establishing a site-wide pollution prevention council; 3) successfully integrating pollution prevention criteria into contracting mechanisms; and 4) developing a technical assessment guide. While these four building blocks represent a strong foundation, the pilot team has much work left to do while it constructs the site-wide pollution prevention program.
INTRODUCTION
The Savannah River Site (SRS) was established in 1950 by the U.S. Atomic Energy Commission (subsequently, the U.S. Department of Energy) to produce isotopes for national defense and peace-time applications. Many of the wastes that were generated as a by-product of performing its mission are now stored at a variety of facilities at SRS. Current activities at SRS include storing and treating high-level liquid radioactive waste, managing low-level radioactive solid waste,

investigating and remediating radioactive, hazardous and mixed-waste sites, producing radioactive isotopes for NASA, and maintaining nuclear weapons. These activities provide an on-going source for the generation of both radioactive and non-radioactive waste streams.

SAVANNAH RIVER SITE PILOT POLLUTION PREVENTION PILOT TEAM

Although waste minimization programs have been used at SRS for several years, the U.S. Department of Energy (DOE) has recently developed a comprehensive pollution prevention (P2) program for SRS. Over the past year, DOE has taken the responsibility for developing and implementing a P2 pilot program and creating a team of DOE personnel to manage this pilot program. The DOE P2 pilot team defines pollution prevention as the use of materials, processes, and practices that reduce or eliminate the generation and release of pollutants, contaminants, hazardous substances, and wastes into land, water, and air. P2 at SRS includes practices that reduce the use of hazardous materials, energy, water, and other resources along with practices that protect natural resources through conservation or more efficient use. The P2 pilot team is working toward establishing a comprehensive P2 program by implementing cost-effective P2 and waste minimization strategies.

One of the primary responsibilities for the P2 pilot team is addressing cross-cutting issues that affect all contractors working at SRS. This responsibility will become even more critical as SRS awards its next Management and Operations (M&O) contract as a multi-contractor award, which means that several contractors will be performing many of the tasks currently being performed by a single contractor. In addition, privatization efforts are continuing at SRS, thus allowing other DOE contractors to work at SRS outside of the existing M&O contract. The P2 pilot team is given the unenviable task of managing the SRS P2 program for all current and future contractors at SRS. The team members, however, are in the best position to coordinate and manage such a program. The structure of the P2 pilot team is presented in Fig. 1.

Building a Solid Foundation

The P2 pilot team's first task was building the framework for a site-wide P2 program. This framework is the foundation for the site's P2 program. The P2 pilot team recognized that a comprehensive P2 program should be strong enough to handle over a thousand waste streams from more than one hundred facilities, yet it also needs to be flexible enough for mission changes as well as changes in its primary operational contract. The following are four primary building blocks that the P2 pilot team used for initiating the SRS P2 program:

- Developing a P2 Policy and Manual
- Establishing a P2 Council
- Successfully Integrating P2 Criteria Into Contracting Mechanisms
- Developing a Technical Assessment Guide

Each of these building blocks will be discussed in the following four sections.

Developing a Pollution Prevention Policy and Manual

A key aspect of any successful P2 program is a commitment from top management. The P2 pilot team worked directly with the DOE-Savannah River (DOE-SR) Manager's office to develop a clear, concise policy statement that can be easily understood and a program manual that can be readily implemented by all on-site activities and by all personnel. The trick was making the policy and manual specific enough so that all DOE

organizations were aware of their responsibilities and SRS contractors would have a clear idea of DOE's expectations regarding P2, while leaving the details of how to meet these expectations up to the contractor. The primary purpose of the policy statement is to communicate the DOE-SR Manager's commitment to P2 by establishing a site-wide program. The policy further directs all SRS employees to implement a P2 program using the program manual as guidance while complying with applicable laws, regulations, statutes, executive orders, DOE Orders, and DOE policies. The program manual identifies roles and responsibilities for various DOE-SR organizations. P2 responsibilities apply to all DOE-SR organizations, but individual organizations may have different responsibilities. Some organizations have little or no operational responsibilities and do not generate hazardous waste. These organizations, however, still have P2 responsibilities for energy conservation and non-hazardous waste recycling. Other organizations have greater responsibilities due to their assigned operations. Table I shows specific responsibilities for each DOE-SR organization.

Fig. 1

Table I

The P2 pilot team developed ten specific objectives to help define and communicate its vision of the SRS P2 program. The following objectives are based on DOE-wide objectives established in the 1994 Pollution Prevention/Waste Minimization Crosscut Plan (1) and DOE Pollution Prevention Policy Memorandum (2):

Establish, under a central DOE-SR authority, a comprehensive P2 Program complete with the necessary infrastructure to reduce environmental releases to air, surface water, groundwater, and land and to conserve energy and water in a cost effective manner.

- Reduce total releases of toxic chemicals to the environment (including off-site transfers for treatment and disposal) to support DOE's compliance with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements" (3), which calls for a 50 percent reduction (DOE-wide) by December 31, 1999.
- Reduce the generation of low-level, high-level, transuranic, hazardous, and mixed waste streams each year. Establish quantitative, yet realistic, source reduction and recycling goals that are based on appropriate production levels or activity levels.
- Reduce low-level waste disposal through source reduction, recycling, and volume reduction to meet annual DOE-SR allocations.
- Reduce disposal of sanitary waste by using source reduction techniques and maximizing the use of existing recycling programs.

Incorporate P2 into detailed facility-level budgets and plans, including both project-specific and programmatic planning documents

Establish the necessary infrastructure to support and promote P2 activities. The infrastructure will include the following items:

- Waste generation and environmental release baselines
- Material inventory tracking system
- Standardized, real-time, computerized tracking system for waste generation and environmental releases
- Waste manifest system
- Waste generator fee system
- Life-cycle cost determinations

Develop prioritized short- and long-term goals. Achievements are facilitated by using a computer-based waste tracking system that uses standardized units and annual milestones.

Incorporate P2 into planning, operations, and design activities.

Perform P2 opportunity assessments. Prioritize potential options on the basis of the most significant P2 potential, technical feasibility, and highest return on investment.

Modify procurement practices to promote P2; reduce the purchase of toxic and hazardous materials to the lowest practical level; incorporate the requirements of Executive Order 12843, "Procurement Requirements and Policies for Ozone-Depleting Substances" (4); and implement Executive Order 12873, "Federal Acquisition, Recycling, and Waste Prevention" (5) by using the Federal government's purchasing power to help stimulate the market for recycled materials.

Integrate P2 into the SRS employee work ethic through training, procedures, performance indicators, and incentive programs.

Develop and maintain a technical information exchange network for technologies, activities, practices, alternative processes, and chemical substitutions that is accessible by all SRS organizations and other DOE sites.

Incorporate P2 plans and activities into SRS public outreach and community involvement forums.

Establishing a Pollution Prevention Council

Each Assistant Manager has a designated point-of-contact representing his or her organization on the council. These individuals participate in council meetings to address P2 matters of SRS-wide significance. The DOE-SR P2 Council is empowered by the DOE-SR Manager to act as an advising body to facilitate the development, coordination, and implementation of the comprehensive SRS-wide P2 program and policy. The DOE-SR P2 Council was not established to exercise oversight on member organizations or to alter existing Assistant Manager authorities with respect to P2 activities under their direction. The ultimate responsibility for the SRS P2 program remains with the Assistant Manager for Environmental Restoration and Solid Waste.

The DOE-SR P2 Council ensures the timely completion of SRS-wide P2 activities that are recommended by the DOE-HQ Executive Board and such other direction as the DOE-HQ Executive Board may provide. Council members exercise their individual authorities collectively as an advising body to set P2 priorities, identify resources and expertise, monitor progress, and resolve issues to ensure the continuing effectiveness of the P2 program. Moreover, council members ensure that their respective organizations fully participate in the SRS P2 program. The DOE-SR P2 Council communicates with the DOE-HQ Executive Board through its chairman, the DOE-SR Manager, to achieve management objectives defined by the DOE pollution prevention crosscut plan (1). The DOE-SR P2 Council implements all further directions provided by the DOE-HQ Executive Board. Council meetings are called by the Chairman and are held on a regular basis. In addition, the Chairman calls special meetings to discuss and resolve stakeholder issues or if the DOE-HQ Executive Board requires actions on an expedited basis. The council establishes internal operating procedures for conducting meetings and schedules for holding meetings. The council extends invitations to representatives of other DOE Offices, DOE-HQ, other agencies, or industry to attend meetings, when appropriate.

Successfully Integrating Pollution Prevention Criteria Into

Contracting Mechanisms

Naturally, the P2 program would not be successful without contractor support. The P2 pilot team ensures that realistic and achievable goals are placed into award fee criteria for each evaluation period. The M&O contractor award fee is the contracting mechanism that allows the contractor to earn a profit if certain DOE-SR goals or criteria are met. An example of award fee criteria is the documented reduction of low-level radioactive solid waste disposal at SRS by ensuring that no more than 70 percent of the annually-allocated disposal space is actually used during the year. This award fee criteria will force generators to find ways to reduce the volume of low-level waste sent for disposal and it will increase the life span for SRS low-level waste disposal facilities.

The P2 pilot team modified the contract with the existing M&O contract to ensure that affirmative procurement requirements were being met. Affirmative procurement basically uses the buying power of the federal government to support industries that manufacture products that are made with recycled material. Used oil, as an example, can be purchased by a recycler, recovered, treated, and made available for purchase as recycled oil. Affirmative procurement authorizes purchasing agents for the federal government to gives a preference to those vendors that sell oil, for example, that has been recycled. Under normal circumstances, purchasing agents are not allowed to give a vendor any type of preference. By modifying the M&O contract, the P2 pilot team gave SRS contractors the ability to meet the requirements in Executive Order 12873 (5), which identifies a preference for purchasing certain products made from recycled materials. This award fee criteria is a way of closing the loop on the recycling industry because it creates a market for the materials that are being recycled in increasing amounts across the country.

In addition, DOE's expectations are further delineated in the SRS Annual Operating Plan (AOP) and the Program Execution Guidance (PEG). The award fee criteria are evaluated and updated twice per year, while the PEG and AOP undergo annual updates. Together, the award fee criteria, PEG, and AOP provide a mechanism for updating both DOE and the contractors on expectations and progress toward achieving SRS's comprehensive P2 program.

Developing a Technical Assessment Guide

DOE has developed a contractor assessment program to help DOE evaluate contractor progress in implementing the SRS P2 program. A three-year plan was developed to evaluate the contractor's implementation of the P2 program. In 1995, the P2 pilot team has performed over 30 assessments using expertise from several DOE-SR organizations and a technical support contractor. This teamwork approach is a primary reason for the successful implementation of the comprehensive SRS P2 program to date. DOE has identified numerous achievements, especially in waste reduction and treatment, and several areas for further development.

One of the best success stories at SRS is the Excess and Salvage Program. This program diverts waste materials to salvage vendors, government excess material exchanges, government donation programs, and public auctions. Funds obtained from the sale of salvaged items are provided to DOE. The Excess and Salvage Program has more than quadrupled the amount of salvaged waste from fiscal year 1992 to fiscal year 1995. Cooperation among many SRS organizations is a key element to the program's success. This program is well-known and widely accepted at SRS, so future growth is expected.

CONCLUSION

In under 12 months, the P2 pilot team has successfully built a strong foundation for the comprehensive SRS P2 program by developing a site-wide policy and manual, assisting with the creation of the SRS P2 Council, integrating P2 program goals and objectives into contracting mechanisms, and developing and implementing an assessment schedule. Significant activities that are currently being implemented or pursued include a waste manifest system, waste generator fees, a subcontractor waste treatment cost allocation system, and generator waste tracking program. The pilot team will build on this foundation as it continues to implement a comprehensive P2 program at SRS.

REFERENCES

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RECONDITIONING CONTAMINATED GRAVEL

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ABSTRACT

Lawrence Livermore National Laboratory (LLNL) has developed a portable screening system that will recondition radioactively contaminated gravel in the field. The separation technique employed by this system removes dirt, contaminated debris, and other fine particles from gravel. The separation process can be used on gravel or other comparable material ranging in size up to 2.5-cm (1-in.) in diameter. The particle size of dirt and debris removed is variable. For pea gravel, the particles removed can vary from 38 mm-1 cm (3/8 in.).

At LLNL, gravel is used in conjunction with the experimental testing of explosives to reduce shock wave propagation. The gravel surrounds the experimental device and buffers the energy generated from the explosion. During an explosion, some of the gravel is broken down into small particles and mixed with contaminants. Contaminants in the used gravel

originate from metal sheathing and other parts comprising the experimental device. These contaminants may consist of radionuclides (primarily depleted uranium) and metals (e.g., beryllium, copper, and zinc) that are considered hazardous by the State of California when disposed. The small particles generated during the explosion mix with the gravel and collect in the void space between the gravel. The removal of void spaces increases the aggregate material compressive strength and reduces the gravel's effectiveness for shock wave reduction. Reconditioning removes the small particles and some contaminants and restores the gravel's effectiveness for shock wave reduction.

The paper presents the process that conveys contaminated material into the screener system, sprays the material with recycled water or other mild cleaning chemicals, and separates the particles based on size. Particles greater than a specified size are discharged out of the screener separator and recycled back into use, thereby reducing the amount of mixed waste generated and minimizing the need for new gravel. An array of smaller particles are discharged into drums and, if desired, reused in other applications. The fines or silt are flushed out of the separator with the water and are removed from the water and consolidated into a drum with the use of a hydrocyclone separator and drum decant system. Because the water in the spray system is recycled, minimal makeup water is needed. The system monitors pH and total dissolved solids (TDS) and, when undesired levels of pH or TDS are reached or when suspended solids could result in clogs, the spray system can be purged and refilled.

INTRODUCTION

Lawrence Livermore National Laboratory performs experimental testing of explosives at designated remote locations in an area called Site 300. An experimental device is used for the explosive test. The construction of the experimental device varies, but it usually has a metal sheath and often contains depleted uranium, beryllium, copper, and zinc. Experimental devices can also contain lead. The experimental testing at Site 300 is conducted on gravel pads. The gravel on the pad is approximately 1.3 cm (0.5 in.) in diameter, with the smallest particles being 2 mm in diameter. The gravel is used to reduce shock wave propagation during explosive testing.

The experimental device is buried with gravel, which surrounds it and buffers the energy generated from the explosion. During the explosion, the gravel is broken down into smaller particles and mixes with contaminants. Contaminants in the used gravel originate from metal sheathing and other parts comprising the experimental device. These contaminants may consist of radionuclides (primarily depleted uranium) and metals (e.g., beryllium, copper, zinc) that the State of California considers hazardous to dispose of.

After an explosive test, a higher percentage of the gravel material is 2 mm or smaller. These small particles generated during the explosion mix with the gravel and reduce its effectiveness for shock wave reduction. With repeated use of the gravel, a buildup of contaminants and radioactivity is deposited on the gravel. When the contaminants are beryllium, copper, and zinc, the buildup results in the potential generation of low-level radioactive waste with California hazardous metals. When the contaminants are lead and chromium, the buildup results in the potential generation of Resource Conservation and Recovery Act (RCRA) mixed waste. See Table I for a listing of the state and federal

regulated hazardous metals and their regulatory levels. To avoid the possibility of generating mixed waste after it is used, the gravel must be removed from the pad and either discarded or reconditioned.

To determine whether or not a waste is hazardous, the State of California requires a leach test and/or a total waste analysis using the California Assessment Manual Wet Extraction Test (CAM-WET) for Soluble Threshold Limit Concentration (STLC) and for Total Threshold Limit Concentration (TTLC). The STLC is a extraction method that measures the amount of extractable substances in the material. The TTLC provides a total analysis of the material by determining which analytes are present and their concentrations. These tests are used instead of the federal Toxicity Characteristic Leaching Procedure (TCLP).

Before we developed the gravel reconditioning method, the gravel was removed from the gravel pad when it no longer reduced shock waves effectively and was placed into disposal containers, sampled, and analyzed. Depending on the analysis, the waste was disposed of as low-level radioactive waste or low-level radioactive waste with California hazardous metals. The contamination had not built-up enough to consider the waste RCRA mixed waste. The amount of gravel removed averaged around 4,536kg (10,000 lb) per explosive test, and about 4,536 kg (10,000 lb) of clean makeup gravel was added to replenish the pad.

Using our reconditioning method, the small particles (particles less than 3.7 mm) and some contamination are removed from the gravel. Now, up to 85% of the gravel (3,855 kg or 8,500 lb per test) is reconditioned and placed back into use. About 680 kg (1,500 lb) of clean makeup gravel is needed to replenish the pad after an explosive test.

Table I

TREATABILITY STUDIES

We performed small-scale treatability studies to determine if screening would be an effective way of reconditioning gravel. A multitiered bench-top sieve unit (or screener) with an assortment of screen mesh sizes was used in the experiments.

Dry Screening

The first experiment was performed on dry gravel to determine the particle distribution of the gravel so that we could determine the optimal screen size(s) for retaining undersized particles. Six screens were selected with the sieve mesh ranging from 8 to 400 (i.e., sieve openings ranging from 2.8 mm to 0.037 mm). Dirty gravel was added to the top tray and allowed to shake in the sieve unit for 10 minutes. After shaking, the amount of gravel in each tray and in the bottom of the pan was calculated. See Table II for results of the test.

Table II

The design for the gravel reconditioning process made use of two screens: one screen for removing coarse fines from the gravel and the other for removing silt and small fines from the coarse material. Table II indicates that particles less than 2 mm account for 4.1% of the total gravel. When using a screen with a larger opening (i.e., No. 8 mesh), only a small increase of particles was noted, so we determined that No. 10 mesh screen could adequately remove coarse fines from the gravel. Both the No. 200 mesh and No. 325 mesh screens could adequately remove silt and small fines from the coarse material; however, the No.325 mesh screen is constructed of fine wires and is very fragile. Because the No. 325 mesh screen tears easily and is expensive ($300 compared to $186) to replace, we decided to use the No. 200 mesh screen.

Wet Screening

In the second experiment, we tested wet gravel to determine the effectiveness of spraying the gravel with water while screening and calculated the moisture content of the wet gravel removed from the unit. The sieve unit was adapted with a recirculating water system. Only two trays (No. 10 mesh and No. 200 mesh screens) were added to the sieve unit. The dirty gravel was added into the top tray of the sieve unit, the water recirculation system was turned on, and the unit was allowed to shake for a specified period of time. After shaking, the amount of gravel was calculated in each tray and for the bottom of the pan. See Table III for the wet screening results.

Table III indicates that particles less than 2 mm account for 9.6% of the total gravel. Compared to dry screening (where particles less than 2 mm account for 4.1% of the total gravel), we determined that wet screening is more effective at removing smaller particles from the gravel. In addition, the amount of water removed from the system when the gravel is discharged is small. The water makeup rate for both the top tray and middle tray of the wet gravel is 3.8%.

Table III

Test for Cleaning Ability

We also studied how well wet screening could clean. We performed the wet screening operation described earlier several times using water and twice using a nitric acid solution (pH 2) on contaminated gravel. Samples of the gravel in the top tray, middle tray, and bottom pan were taken and analyzed. The test results for gravel washed with water are shown in Table IV.

Table IV

The analyses shown in Table IV are based on the State of California's leach test and total waste analysis. The differences between the Federal (TCLP) and the California State (STLC) leaching tests are subtle. The California State leaching test is more rigorous and, therefore, provides us with more conservative results. The differences in these tests are summarized in Table V.

Table V

The STLC test was performed on the larger pieces of gravel (particles >2.0 mm) to test the effectiveness for reducing leaching, and the TTLC test was performed on the sludge and silt (particles 2.0 mm) to determine the type and concentration of material that was removed by the screening process.

Table IV shows that some beryllium, chromium, copper, lead, and zinc was removed; however, when performing a mass balance on each contaminant, the exact amounts or percentages could not be calculated with the limited number of samples taken. The gravel is heterogeneous, which made it difficult to collect representative samples. Future samples will be taken of the gravel, sludge, silt, and fines. These sampling results may help us determine how well wet screening cleans the gravel.

DESIGN REQUIREMENTS

The full-scale Gravel Reconditioning Unit was designed to meet the following criteria:

- A gravel process rate of approximately 5,443 kg (12,000 lb) in a 6-hour day or 907kg/h (2,000 lb/h)
- Ability to feed gravel to the screener with a front-end loader if conveyors are not used

Hopper loading minimized to 2-3 times a day (i.e., gravel capacity of between 1.2-1.8 m3 (42-63 ft3) if a hopper is used)

Skid-mounted unit, transportable by a flatbed truck, so that it can be moved from one gravel pad to another

Ability to withstand an outside environment and outdoor location

Portable so that it can be operated in the field on the gravel pad at a distance of 30 m (100 ft) from any electrical or water source

Easy to operate and requiring minimal set-up, operating, and shutdown effort

No use of an air compressor in its operation

LLNL seismic criteria at all times

Design and fabrication cost of less than $100,000 for the unit

Design, procurement, and fabrication schedule of 7 months

Design Overview

The Gravel Reconditioning Unit is a portable skid-mounted unit used to recondition gravel. The gravel reconditioned by the unit is restored to its original size with its original dampening effectiveness and is placed back into use. The Gravel Reconditioning Unit contains a feed delivery system, screen separator unit, water reservoir, water recirculation system, rinsate separation system, and control panel. A layout of the gravel reconditioning process is shown in Fig. 1.

Fig. 1

Feed Delivery System

The feed delivery system contains a hopper, slide control valve, and flexible connector. The feed delivery system uses a front-end loader to place the gravel into the hopper. The hopper feeds to the screen separator unit that is mounted on the skid.

For ease in construction, the hopper was designed as an inverted pyramid with a rectangular opening and base. The hopper was constructed out of 304 stainless steel with all exposed seams welded. The overall dimensions of the Gravel Reconditioning Unit is 1.7 m wide x 2.4 m long x 2.7 m high (5.5 ft x 8 ft x 8.9 ft).

The bottom opening of the hopper is located directly over the inlet to the screen separator unit. The slide control valve is mounted under the hopper and regulates the amount of gravel entering the screen separator unit. The slide control valve is a manually operated slide valve. With a little effort, the manual valve can be shut against a full hopper of gravel.

The flexible connector is mounted at the bottom of the slide valve and is connected to the screen separator unit. The connector is flexible so that it can move with the screen separator when it vibrates and can be lifted off easily to change out screens.

Screen Separator Unit

The screen separator for the Gravel Reconditioning Unit is a commercial unit used for wet classification (i.e., solid classification in a liquid medium). The screen separator is cylindrical, has a screen diameter of 0.76 m (30 in.), and is constructed out of stainless steel. The screen separator has two screens and antiblinding features to dislodge small particles from the screen. The sieve mesh for the screens were originally No.10 mesh and No. 200 mesh, but were increased to No.6 (3.7 mm opening) mesh and No. 18 (1 mm opening) mesh to allow better drainage of liquid through the screen.

The screen separator has one inlet at the top of the screen separator, three discharge ports, and a spray system. The spray system washes the

gravel as it enters into the screener. To prevent incoming gravel from damaging the screen, a velocity breaker (strike plate) was installed on the screener lid. The top discharge port is for effluent gravel (particles >3.7 mm), the middle discharge port is for effluent sludge (particles 3.7 mm and >1 mm), and the bottom discharge port is for the effluent silt, fines, and water. The top discharge port discharges clean gravel in to a collection container, the middle discharge port discharges the effluent sludge in to a 208-L (55-gallon) waste drum, the bottom discharge port discharges the effluent silt, fines and water into the water reservoir.

The screen separator uses a three-dimensional inertial vibratory motion to separate particles by size. The screen separator vibrates horizontally, vertically, and tangentially. The control for gravel flow in the unit is adjustable by increasing and/or decreasing the mass of the top and bottom eccentric weights and the increasing or decreasing the lead angle of the bottom eccentric weight. Increasing the bottom eccentric weight increases the vertical component of motion, increasing the top eccentric weight increases the horizontal throw and cause oversized material to discharge at a faster rate, and increasing the lead angle of the bottom eccentric weight imparts a spiral motion of the particles on the screen. If gravel requires additional cleaning, the lead angle of the bottom eccentric weight is increased to keep the gravel on the screen longer.

Water Reservoir

The silt, fines, and water out of the bottom discharge flows into a water reservoir that is constructed out of stainless steel, has a total capacity of 566 L (150 gal), and an average operating volume of 330 L (87 gal). Makeup water is also introduced in the water reservoir. The discharge for the silt solution is at the bottom of the water reservoir. A hinged lid is mounted on top of the reservoir for easy cleanout. A mixer and instrumentation for monitoring pH, conductivity, high water level, low water level, and high-high water level are mounted to the water reservoir to monitor conditions of the water.

Water Recirculation System

The water recirculation system consists of a pump, valves, and piping. The system is designed to process the reservoir water through the rinsate separation system and recirculate it back into the screener separator. The system was designed to the maximum flow rate and pressure requirements of the spray nozzles. The pump is designed so that it doesn't pulsate because pulsating flows cause interferences with the rinsate separation process. The pump chosen is a multistage centrifugal pump.

The valves and piping are designed to meet high system working pressures and constructed out of material that is protected against outdoor environments (ultraviolet radiation). The valves and piping used is schedule 80 chlorinated polyvinyl chloride (CPVC).

In addition to the valves that direct and regulate flow, an over pressure relief valve and pump bypass valve was installed to prevent over pressuring the system. A flow meter was also installed on the water recirculation line to monitor the flow rate of the recirculated water.

Rinsate Separation System

The rinsate separation system consists of a hydrocyclone separator, motor-operated ball valve, purge diffuser, and drum decant system. The rinsate separation system is installed in the water recirculation system

to remove silt and fines from the recirculated water. The solid-free water is discharged out the top of the hydrocyclone separator and into the screen separator. The solids are discharged out the bottom of the hydrocyclone separator and into a 208-L (55-gal) drum. Liquid from the 208-L (55-gal) drum is decanted off and gravity fed into the water reservoir.

The solution pumped from the water reservoir enters the hydrocyclone separator tangentially, which sets up a circular flow. The solution is then drawn through tangential slots and accelerated into the separation chamber of the hydrocyclone separator. Centrifugal action tosses particles heavier than the water to the perimeter of the separation chamber. The particles drop along the perimeter of the cyclone separator and settle into the collection chamber. The solid-free water is drawn up the separator's vortex, up through the separator's outlet, and into the screen separator. Due to high system pressures, a purge diffuser was installed on the hydrocyclone separator discharge line to prevent inadvertent spraying of liquid.

The drum decant system, which consists of a drum shroud with baffle plate and a discharge line to the water reservoir, is attached to the 208-L (55-gal) drum. The gasketed drum shroud is clamped to the top of the drum and allows the water level to raise past the height of the drum without leaking out. As the discharged material (solids and water) fill the 208-L (55-gal) drum, the solid material tends to settle to the bottom of the container while the lighter material remains on the top. When the water reaches the discharge port, the water gravity flows into the water reservoir. The drum decant system minimizes the amount of makeup water to be added to the system, minimizes the amount of liquid waste to treat, and maximizes the solid holding capacity in the drum.

Control Panel

All controls for the Gravel Reconditioning Unit are located on a control panel. The frequency and duration for purging the hydrocyclone separator are also adjustable from within the panel. The panel is a NEMA 4 enclosure, and all controls are weather resistant and rated for outdoor use. The lights and controls on the control panel are visible from outside the enclosure. A crash button and main disconnect are also mounted on the control panel.

CONCLUSIONS

The Gravel Reconditioning Unit is an inexpensive, easy-to-use, low maintenance, portable, and effective way to recondition gravel. Applying the Gravel Reconditioning technique to the gravel on the gravel pads at Site 300 will reduce the amount of low-level radioactive waste, low-level radioactive waste with California hazardous metals, or RCRA mixed waste generated.

The Gravel Reconditioning Unit was designed and fabricated in 8 months for less than $100,000. Testing began in January 1995 and full production cleaning commenced May 1995. Prior to the construction of the Gravel Reconditioning Unit, gravel from past explosions were contained in disposal boxes to await shipment and disposal. Since the late 1980's, a backlog of 142 disposal boxes (644,000 kg) of used gravel have accumulated. To date, over 10% (64,400 kg) of the backlog gravel has been reconditioned.

In an 8-hour period approximately 4,535 kg (10,000 lb.) of gravel is processed with up to 85% by weight (or 3,855 kg) of the large gravel being recycled. The remaining 15% of the material is contained in either

a disposal box or 208 Liter (55-gallon) drum to await shipment and disposal. The water reservoir is changed out approximately once a week, due to silt build-up and higher level of contaminants in the water. The handling and treatment of the waste water generated by the reconditioning process is a routine practice for LLNL. The waste water can be treated at the Tank Farm and (when analytical results indicate that it meets acceptance criteria) emptied into the LLNL sewer (ultimately to reach the city water reclamation plant). If necessary, the silt and sludge waste generated by the reconditioning process can also be treated (stabilized) at LLNL.

This gravel reconditioning procedure results in a cost benefit of approximately $42,000 for processing and disposing of the backlog gravel. This includes a $1,800/day savings in disposal costs but, does not include savings in costs associated with the procurement and delivery of new gravel. Factoring in the cost of the unit, the savings in treating the backlog material is greater than 10%. After the cost of the unit has been realized, the savings increase to greater than 30%.

ACKNOWLEDGMENTS

Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract W-7405-ENG-48.

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Session 15 -- POSTER - MIXED WASTE
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15-1

THE SYSTEM COST MODEL: A TOOL FOR LIFE CYCLE COST AND RISK ANALYSIS

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ABSTRACT

In May of 1994, Lockheed Idaho Technologies Company (LITCO) in Idaho Falls, Idaho and subcontractors began development of the System Cost Model (SCM) application. The SCM estimates life cycle costs of the entire U.S. Department of Energy (DOE) complex for designing; constructing; operating; and decommissioning treatment, storage, and disposal (TSD) facilities for mixed low-level, low-level, and transuranic waste. The SCM uses parametric cost functions to estimate life cycle costs for various treatment, storage, and disposal modules which reflect planned and existing waste management facilities at DOE installations. In addition,

SCM can model new TSD facilities based on capacity needs over the program life cycle. The user can provide input data (default data is included in the SCM) including the volume and nature of waste to be managed, the time period over which the waste is to be managed, and the configuration of the waste management complex (i.e., where each installation's generated waste will be treated, stored, and disposed). Then the SCM uses parametric cost equations to estimate the costs of pre-operations (designing), construction, operations and maintenance, and decommissioning these waste management facilities. The SCM also provides transportation costs for DOE wastes. Transportation costs are provided for truck and rail and include transport of contact-handled, remote-handled, and alpha (transuranic) wastes.

A complement to the SCM is the System Cost Model - Risk (SCM-R) model, which provides relative Environmental, Safety, and Health (ES&H) risk information. A relative ES&H risk basis has been developed and applied by LITCO at the INEL. The risk basis is now being automated in the SCM-R to facilitate rapid risk analysis of system alternatives. The added risk functionality will allow combined cost and risk evaluation of EM alternatives.

INTRODUCTION

The System Cost Model (SCM) was designed based on the cost knowledge developed for the Waste Management Programmatic Environmental Impact Statement (WM PEIS). The WM PEIS demanded that a consistent life cycle cost system be developed and utilized for cost analysis and data input for risk assessments and socioeconomic analysis. The WM PEIS analyses were successfully performed on various waste management alternatives for low-level waste (LLW), mixed low-level waste (MLLW), and transuranic waste (TRU). The engineering basis for the SCM was derived from WM PEIS technical reports: Waste Management Facilities Cost Information for Low-Level Waste (1), Waste Management Facilities Cost Information for Mixed Low-Level Waste (2), Waste Management Facilities Cost Information for Transuranic Waste (3), and Waste Management Facilities Cost Information for Transportation of Radioactive and Hazardous Material (4). This cost information as programmed in the SCM provides DOE with a tool to perform waste management sensitivity analysis on the Baseline Environmental Management Report and Technology Development applications.

Sponsors

Lockheed Idaho Technologies Company's (LITCO's) Technical Support Program with contractor assistance from MK-Environmental Services (MKES) developed the System Cost Model (SCM) at the request of the Department of Energy's (DOE's) Office of Waste Management (EM-30). Further development and refinement of the SCM for technology development applications is sponsored by the DOE Office of Science and Technology.

What Is the SCM?

The SCM estimates life cycle costs of the entire U.S. Department of Energy (DOE) complex for designing; constructing; operating; and decommissioning treatment, storage, and disposal (TSD) facilities for LLW, MLLW, and TRU (including mixed TRU). The SCM uses parametric cost functions to estimate these life cycle costs. Parametric cost functions develop costs for various treatment, storage, and disposal modules which reflect planned and existing facilities at installations. In addition, SCM can model new facilities based on capacity needs over the program life cycle. The user can provide input data (default data is included in the SCM) including the volume and nature of waste to be managed, the time

period over which the waste is to be managed, and the configuration of the waste management complex (i.e., where each installation's generated waste will be treated, stored, disposed, and transported). Then the SCM uses parametric cost equations to estimate the costs of the following program life cycle phases of waste management facilities.

Pre-operations__Pre-operations pertain to the studies and bench scale test costs, demonstration costs, and operations budget funded activities (conceptual design, safety assurance documentation, permitting, preparation for operation, and project management).

Construction costs__Construction costs include title I and II design, inspection, project management, building construction (including indirect costs), equipment (including indirect costs), construction management, and contingency costs related to facility construction.

Operation and Maintenance__Operations and Maintenance relates to operating labor, utilities, material, maintenance, and other costs including reserve and contingency costs.

Decommissioning__Decommissioning pertains to manpower, surveillance and maintenance, assessment and characterization, environmental documentation review, operations, closure, and post-closure monitoring.

Examples of Current Uses for the SCM

Baseline Environmental Management Report. The SCM has been used in FY-95 and FY-96 to support the preparation of the Waste Management Portion of the Baseline Environmental Management Report (BEMR). In order to support BEMR modeling efforts, the SCM has been calibrated for each of the six major DOE sites. The calibration is done so that SCM can simulate the BEMR cost estimates for these sites and represent their existing and planned waste management facilities in the complex-wide BEMR modeling activities. DOE then uses the BEMR treatment option within the SCM to conduct sensitivity analysis.

To increase the accuracy of the model for BEMR analysis activities, the SCM was calibrated to the baseline plans at the six major DOE installations (i.e., Hanford Site, Idaho National Engineering Laboratory, Los Alamos National Laboratory, Oak Ridge Reservation, Rocky Flats Environmental Technology Site, and the Savannah River Site). These six sites represent over 80% of the waste management costs within the DOE complex. The calibration effort included site visits and intense data gathering to:

- Customize SCM with site-specific input data;
- Calibrate modeled estimated operation and maintenance costs to the sites cost for existing treatment, storage, and disposal facilities;
- Calibrate algorithms for studies and bench scale test, demonstration, construction, and operations to the site's assumptions for facility design, construction, and operations costs; and
- Develop cost relationships to derive a total site waste management cost (including program management and other site overheads) from modeled SCM facility costs.

Technology Development. The SCM will be used in FY-96 to perform technology assessments on thermal and non-thermal treatment systems. The SCM will aid Technology Development in estimating system and sub-system cost variations due to: changes in waste streams loading, schedules (e.g., long-term storage, varying operation periods), pre-treatment requirements (e.g., sorting, characterization, handling), waste form variations on disposal requirements, and transportation (e.g., containers, packaging).

Examples of the type of analysis that can be performed using the SCM for Technology Development include:

Evaluation and comparison of new thermal treatment technologies and non-thermal (washing) technologies;

Comparison of effects from final waste form (e.g., grout, glass) on transportation and disposal costs and risks;

Trade-offs between waste storage costs and risks versus improved technology performance;

Advantages of combined processing of similar wastes (e.g., alpha LLW, alpha MLLW, mixed TRU waste);

Optimize performance of treatment systems (e.g., reducing treatment effluents, reducing sorting and characterization).

SYSTEM COST MODEL FUNCTIONALITY

The SCM system architecture enables new features to be easily added as the product matures. A high-level, functional view of SCM architecture is shown in Fig. 1.

Fig. 1

The SCM architecture illustrates the partitioning of the model structure into three distinct components: Inputs, System, and Outputs. The Inputs define the model parameters that are required to execute calculations of the system. The System components consist of the calculation engines required to produce the desired outputs. The Output component provides the desired screens and reports that contain the results from the system calculations and provide a record of the input parameters. Each of these components are described in more detail in the following sections.

INPUT INFORMATION

Data from the five required categories (waste loads, TSD scenarios, site schedules, and facility profiles) must be completed for the program to function. It is up to the user's discretion whether to make any changes to the five optional categories (work breakdown structure (WBS) scale factors, other site costs, cost factors, charge backs, and inflation factors). Default data for these five optional categories has been pre-loaded into the SCM.

Waste Load Information

Matrix categories, also termed waste stream fields, are categories of waste that are distinguishable by their origin, physical state or form, composition, radioactivity, or a combination of these characteristics. The waste loads in the SCM are identified by 32 unique matrix categories which are consistent with the classification scheme used in the Mixed Waste Inventory Report. The categorization of wastes also allows summation of common wastes across waste types (i.e., LLW, MLLW, TRU). Waste loads provide the foundation for the SCM. Facility costs are calculated from algorithms relating cost to capacity, and capacity is derived from waste loads. Waste loads are required for a given waste type (low-level waste, mixed low-level waste, transuranic waste) in order to create a specific case. If waste loads are not defined for any of the waste types, calculations cannot be executed. Waste loads that are in inventory are termed legacy waste. Generated annual waste contributes to the overall waste loads for the SCM case scenario.

Treatment, Storage, and Disposal Scenarios

The user defines the destination of waste for treatment, storage, and disposal (TSD) locations by site (onsite, offsite). The waste can be "split" by the 32 waste matrix categories to go to different DOE sites for treatment, storage, or disposal. For onsite treatment, SCM offers

optional treatment schemes that make use of different technologies aimed at meeting various treatment objectives.

Site Schedules

Once the quantity of waste and the treatment, storage, and disposal scenarios have been entered, the SCM allows the user to edit the scheduling of new treatment, storage, and disposal facilities at a site. Site schedules can be manipulated for start and stop dates, and durations of major cost elements (e.g., preoperation, construction, O&M, D&D). Shipping schedules are used to establish when waste moves from storage to treatment. The scheduling information controls storage requirements and will affect the amount and scheduling of costs.

Facility Profiles

The SCM's database contains information about the known DOE site waste management facilities, based on the information available at the time of the release of the SCM application. This information includes capacities, operating periods, and any known upgrade costs or O&M costs. The SCM also contains information for the modules represented by the facilities and the waste type dedications (what kind of waste the module can process).

Optional User Input Data

In addition, the user is allowed to select, enter, or change the following data:

- Offsite DOE treatment unit costs
- Treatment options for each waste type (e.g., Base Case, BEMR Calibrated, Nonflame,)
- Transportation (rail/road)
- Commercial unit costs
- Existing or planned DOE facility cost information
- Site-specific cost factors and labor rates
- Cost escalation factors

User Input Options

The model provides a default set of parameters that SCM users may use to select the site, facility, waste type, etc. The system was designed so that very little input data is required from the user. The SCM contains and provides the following internal reference data:

- Cost data based on Waste Management Facility Cost Information reports;
- Generic schedule data;
- Existing and planned/approved facility capacity and operating parameters (based on latest BEMR);
- Minimum and maximum scaling factors for parametric cost/capacity equations;
- Transportation miles and costs/mile;
- Standard operating parameters (such as years of operations and maintenance);
- Module flow factors (site-specific processing schemes).

Case Changes Can Be Saved

If the user selects to change the data input elements as defined above, the modified data can be saved in scenarios called cases. An SCM user can access the saved cases in the future and perform additional modifications. The SCM cases can be copied to or from different personal computers (PCs) via floppy diskettes or a network media. The user can select different cases and merge these cases into a single case.

SYSTEM CAPABILITIES

The SCM provides the general capability to calculate the total life cycle costs by module for the LLW (alpha, non-alpha, and remote handled), MLLW

(alpha, non-alpha, and remote-handled), and TRU waste / MTRU waste (contact and remote handled). The SCM user can chose between different treatment options for each waste type. The following table provide the treatment options that SCM supports for the 3 major waste types.

Table I

System Calculations

The System calculations include the following:

- Full-time equivalent (FTE) manpower estimates for each module by WBS
- Cost and FTE annual and cumulative profiles for the three selected waste types
- The administrative and support modules are automatically sized and costed
- Cost by general module/facility type (pre-treatment storage, treatment, storage, and disposal)
- Cost by new facilities versus existing facilities
- Portable treatment or commercial treatment options
- Fixed minimum cost of facility modules for waste loads less than the defined module capacity range
- Summation of DOE complex cost roll-ups of all DOE sites
- Integrated transportation costs (combination of road and rail)
- Calculations to allow facility capacity versus operating period manipulations

Unique System Features

The System also has unique capabilities which allow waste shedding, which allows a site to distribute the same waste stream to several sites including a commercial designation. Disposal shedding also allows the generating site to distribute waste treated at one treatment location to multiple disposal sites. A user can also designate offsite processing for treatment and/or disposal of waste. In past versions, the SCM automatically built a new facility that would be decontaminated and decommissioned. To more accurately provide full life cycle costs, a finite lifetime constraint is placed on all new facilities (default of 30 years which can be configured by the user up to 50 years). The SCM automatically calculates pre-treatment storage based on user decisions of the earliest construction startdate of new facilities and waste scheduling. The system also automatically calculates post-treatment storage based on availability of disposal facilities. Waste storage or disposal costs can also be charged back to the generating sites. This charge is based on the quantity of waste and either a calculated unit rate or a user-provided unit rate.

The total life cycle costs for each module are organized based on the work breakdown structure (WBS) outlined in Table II.

Table II

OUTPUT CAPABILITIES

The SCM provides the following reporting capabilities:

- Prints user-selected case summary or site detail reports; case summary, site detail, or case comparison graphs; and case Gantt charts.
- Preview option for viewing reports on the screen before printing reports.
- Data exports into text database, spreadsheet, and word processing format.
- Present data in tabular and graphical formats (Fig. 2).

Fig. 2

RELATIVE RISK ANALYSIS CAPABILITIES

The System Cost Model - Risk (SCM-R), a prototype version of the SCM, will provide relative Environmental, Safety, and Health (ES&H) risk information. A relative ES&H risk basis has been developed and applied by LITCO at the INEL. The risk basis is now being automated in the SCM-R to facilitate rapid risk analysis of system alternatives. The added risk functionality will allow combined cost and risk evaluation of EM alternatives. The simplified methodology for ES&H relative risk is composed of the following five elements (5):

Element 1 - is the characteristics of the waste type which is composed of quantity of contaminants in the waste type and the specific radio toxicity of the radionuclides and/or the specific chemical toxicity of the hazardous chemicals.

Element 2 - expresses the ease with which the contaminants in the waste type could escape confinement as a result of events or conditions that breach the confinement.

Element 3 - expresses the likelihood, or probability, of loss of waste confinement.

Element 4 - how effectively the released contaminants could be moved by environmental transport processes (e.g., wind, groundwater transport, biotic transport) to receptors.

Element 5 - the presence of human receptors. How frequently workers would be located around the waste type, how many workers would be involved, and how closely they would be involved? How many members of the public are located near the waste type, and how close?

The data inputs for the SCM-R includes several parameters from the cost and FTE calculations. These include technology descriptions, waste forms, schedules, transition and rest states, facility capacities, and DOE site-specific information. Additional risk parameters are included in the SCM-R to define the waste characteristics (i.e., radiological and hazardous profiles), mobility, confinement, stresses, transport, worker and public proximity, and time in states. The relative risk will be output from the SCM-R on a comparative state basis, and on an annualized risk basis.

DELIVERABLE SOFTWARE AND DOCUMENTATION

Software distribution of the System Cost Model is under the control of LITCO and the DOE. System modifications and configuration control are closely maintained by the project. New releases of the SCM consist of the following:

1. SCM Software: The software consists of an executable version of FoxPro which runs stand-alone on an IBM PC (or compatible) under the Microsoft Windows 3.1 environment.
2. Product Description: The SCM Product Description (6) provides the current system description of the model capabilities. The Product Description will be maintained throughout the SCM development cycle to ensure that it is current and accurate. All changes to this document will be done in accordance with the change control procedures established by the WMFCI project manager and LITCO document control.
3. User's Manual: A user's manual will be developed to accompany major releases of the SCM. The manual will provide a basic overview of the software and user instructions. In addition to the user manual, user training and a training presentation may be required.

CONCLUSIONS

This paper has provided a description and overview of the capabilities of the System Cost Model and the System Cost Model - Risk. The SCM calculates life cycle waste management costs based on waste loads. The

SCM is loaded with default information that represents the latest available site-specific data collected in support of the 1996 BEMR. The SCM has been successfully applied to several EM-30 programs. The tool will be refined in the future by addition of the relative risk function (SCM-R), and capabilities to provide technology development analysis.

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15-5

USE OF CONSUMER WASTE PLASTICS FOR ENCAPSULATION OF MIXED WASTE

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ABSTRACT

A polymer encapsulation process is being developed at the Rocky Flats Environmental Technology Site (RFETS) for treatment of mixed wastes containing high concentrations of salts, which inhibit the solidification reaction of Portland cement based grouts. Encapsulation using polymers forms a suitable matrix for this type of waste by a micro encapsulation of the waste via a homogeneous mixing process of the polymer and waste. Solidification of actual radioactive and mixed waste in a pilot-scale system will be the first large-scale polymer encapsulation of significant quantities of actual mixed waste within the DOE complex. As part of this program, methods are being identified to collect consumer waste plastic for utilization in the polymer encapsulation of wastes.

INTRODUCTION

The need to treat mixed waste (containing both hazardous and radioactive constituents) to meet regulatory requirements is a problem that is common to many DOE sites. During their many years of operation, Rocky Flats and other DOE sites generated mixed wastes that are now subject to the Resource Conservation and Recovery Act Land Disposal Restrictions (RCRA LDR). LDR regulations require that mixed waste be treated by specific technologies or that final waste forms meet specific treatment standards before they can be considered for disposal at a permitted hazardous waste landfill. Mixed waste must also comply with Department of Transportation shipping standards and acceptance criteria specific to each permitted landfill.

Rocky Flats cannot ship the majority of its mixed waste off site for treatment or disposal because the waste does not meet the regulatory requirements. Also, Rocky Flats does not have adequate treatment facilities in place to produce LDR compliant final waste forms. For these reasons, polymer encapsulation, as one of a suite of treatment technologies, is being developed at Rocky Flats.

Although not as widely used as cementation, polymer encapsulation offers several advantages over cementation, including increased waste loading, increased waste form durability, and insensitivity to variations in waste stream chemistry. Polymer encapsulation is a simple, low temperature process that is relatively easy to permit and implement. Furthermore, waste plastics can be used as the encapsulating medium. This has cost benefits and provides an opportunity to put to beneficial use material that might otherwise be landfilled.

Since polymer encapsulation can be adapted to handle commercial low level and hazardous waste, Kaiser-Hill has entered into an agreement with Rust Federal Services to commercialize the technology. As part of the agreement, Rust will use resources available through its sister company, Waste Management, to investigate the economics of collecting appropriate post consumer plastics for mixed waste treatment.

TECHNOLOGY DESCRIPTION

Mixed waste stabilization using polymers is adapted from existing processes widely used in the polymer industry (1-3). Although the application of polymers to mixed waste differs from private sector applications in the areas of product acceptance criteria and operating conditions, the two applications are very similar. All of the equipment and supplies are commercially available "off the shelf."

Thermoplastic polymers, such as polyethylene, are heated above their melting temperature (110oC), mixed with powdered wastes, and poured into a disposal container, where solidification occurs as the melt cools. No significant chemical interaction occurs between the waste and the polymer. Waste requiring stabilization to meet Toxicity Characteristic Leaching Procedure (TCLP) standards is dried and then encapsulated in polyethylene using a commercially available compounding extruder. The appropriate extruder for this application uses two co-rotating intermeshing screws. Electrical resistance heaters in the extruder barrel and friction introduced by the rotation of the screws melt the polyethylene pellets. Dried waste is fed into the extruder using a down stream side feeder, where the waste encounters molten polyethylene. Kneading blocks and/or pin mixers downstream of the waste feed port mix the waste with the molten polyethylene. A vent port can be used to remove excess moisture and reduce the porosity of the final waste form. The

molten mixture is extruded into the final disposal container, where solidification occurs as the mixture cools.

Rocky Flats also uses thermoplastic extrusion to macroencapsulate radioactive lead and debris wastes. In this case, waste is placed in a basket sized slightly smaller than the disposal container. Molten polyethylene is extruded into the disposal container, filling the annular space between the basket and disposal container as well as voids within the waste.

The treatability studies completed at Rocky Flats indicate that waste loadings of 30 to 80 weight-percent are possible with polyethylene extrusion while still producing an LDR compliant waste form (4-6). Waste loadings of 50 weight-percent are typical for the majority of the mixed waste tested. Depending on the physical characteristics of the waste stream and the maximum achievable waste loading, volume expansion factors vary from -0.5 to +1.5.

Because polyethylene is resistant to most chemicals, polyethylene extrusion is a good option for mixed wastes that are difficult to stabilize using other methods such as cementation or vitrification. Sodium chloride salts, for example, are not soluble in glass and severely inhibit the cement hydration reaction. Since there is no reaction between the sodium chloride and the polyethylene, high waste loading of sodium chloride is achievable in polyethylene. Other salts that are difficult to stabilize in cement or glass but that are compatible with polyethylene extrusion include sodium sulfate, ammonium sulfate, and sodium nitrate.

Through its wide usage, the durability of polyethylene has been clearly demonstrated. Particularly significant is the fact that polyethylene does not degrade in municipal landfills. Of concern to radioactive waste forms, however, is the potential for degradation and hydrogen gas generation as a result of radiation-induced reactions. An evaluation conducted by Rocky Flats concluded that radiolysis effects for low level radioactive (<100 nCi/gram) waste streams encapsulated in polyethylene are insignificant (7). In terms of the mechanical properties, low level waste streams do not have a sufficiently high specific activity to deliver the 100 megarads required to significantly degrade the polyethylene. Hydrogen gas generation is also insignificant with these waste types. Polyethylene extrusion is not a feasible option for high specific activity waste streams due to the potential for radiolysis effects. This is not an issue, however, for low level waste forms.

Another area of concern evaluated by Rocky Flats was the issue of the thermal stability of sodium nitrate waste streams encapsulated in polyethylene (7). Encapsulating sodium nitrate, an oxidizer, with polyethylene, an organic, could potentially result in a chemically reactive mixture. Such a mixture of fuel and oxidizer could burn if exposed to sufficient thermal energy, and consequently release additional energy and gases. Experiments conducted to evaluate the issue of the thermal stability of the sodium nitrate/polyethylene waste form conclusively demonstrated that no exothermal reaction hazards exist. A wide variety of tests have been conducted, including calorimetry, ignition, time-to-explosion, thermal decomposition, gas evolution, detonation, and thermal runaway. None of the experimental results indicate any tendency to detonate or explode, even under wide variations in the waste-to-polyethylene ratio.

Favorable results have been achieved on bench-scale treatability studies and limited cold pilot-scale testing; however, no pilot-scale tests have

been conducted to date on polyethylene micro encapsulation of actual low level mixed wastes. This is critical to demonstrating the technology's performance. Rocky Flats has recently acquired a twin screw extruder for pilot scale testing of large quantities of actual low mixed salts, sludge, and ash. In conjunction with Rust Federal Services, Rocky Flats will be conducting extended hot pilot-scale tests using this extruder in mid-1996. In addition to supporting the pilot-scale tests, Rust will use resources available through its sister company, Waste Management, to investigate the economics of collecting appropriate plastics for mixed waste treatment.

USE OF RECYCLED PLASTICS FOR WASTE ENCAPSULATION

The ideal plastic for mixed waste treatment would be readily available, chemically resistant, and inexpensive. In order to simplify operations and produce an acceptable waste form, the plastic needs to have a low melt viscosity (melt index greater than 50) and cannot contain co-polymers or plasticizers that would offgas at extruder operating temperatures. Rocky Flats has had the most success with low density polyethylene (LDPE) and linear low density polyethylene(LLDPE). High density polyethylene (HDPE) is more widely used by industry, but it is not appropriate for most Rocky Flats applications because large monolithic pours of HDPE tend to shrink and crack upon cooling.

LDPE and LLDPE, with the exception of films containing copolymers or plasticizers, are not currently being recycled in Colorado. Common post-consumer sources of this material are food containers, especially lids. The challenge is to collect sufficient quantities of relatively clean material without burdening the recycle vendor with unwanted plastics. There are several options for accomplishing this.

One option is to add LDPE and LLDPE to current curbside collection programs. In recent years, recycling has become a significant part of waste collection systems throughout the country. "Waste Minimization" programs have increased the awareness of homeowners and corporations of the advantages of recycling. As a result, the addition of another "product" in curbside collection systems would not be extremely difficult. The concern of the recycle vendor is that, because Rocky Flats is the only market for these plastics, too much material will be collected. Targeting specific geographic areas, such as certain neighborhoods, is usually not feasible for large recycling companies. The burden of disposing of additional unwanted plastics significantly reduces the economic viability of this option.

Another option is to sort the desired plastics from the material currently collected through curbside recycling programs. Frequently, consumers inadvertently place LDPE or LLDPE containers in recycling bins specified for HDPE and polyethylene terephalate (PET) only. The recycler must then separate the LDPE, LLDPE, and other unwanted plastics from the HDPE and PET. Currently, plastics other than HDPE and PET are discarded. The recycle vendor can collect sufficient LDPE and LLDPE from the discarded plastics to augment, but not satisfy, Rocky Flats needs. The problem with this option is that processing and transportation of limited quantities of post consumer plastics is not economically viable.

Another option is to collect the required plastics from local companies willing to collect and/or provide waste plastics to Rocky Flats as a public service. One local grocery chain, for example, already provides bins for customers to drop-off recyclable materials. The company would be willing to collect LDPE and LLDPE for Rocky Flats This reduces overall

product costs to Rocky Flats by eliminating collection costs, but processing and transportation costs are still an issue. Processing and transportation of the material includes transportation from the source to a processing plant and from the processing plant to the point of use. Effective processing methods (such as pelletizing to reduce volume) may significantly reduce transportation costs, thereby reducing overall product costs to Rocky Flats.

One large manufacturing company in the region has offered to donate waste plastics to Rocky Flats for use in mixed waste treatment. This is obviously the best option for Rocky Flats, since it eliminates product costs and the volumes can be easily controlled. Unfortunately, the available plastics are not optimum for Rocky Flats applications, although they may be usable. Materials testing still needs to be completed to determine whether the available plastics are suitable for Rocky Flats needs.

If this material is deemed unsuitable to Rocky Flats, then the required plastics may be collected from companies manufacturing products such as food containers or food lids. This material is sold as "regrind" and is typically clean and readily available. It costs significantly less than virgin polyethylenes, but it must be shipped to Colorado from other states. Although Rocky Flats is currently using this material to satisfy existing needs, the preference is to utilize plastics that are being landfilled in Colorado.

CONCLUSIONS

Polymer encapsulation is a good treatment option for many mixed wastes, especially wastes that are difficult to stabilize using cementation or vitrification. Some of the advantages include high waste loading, excellent waste form durability, and insensitivity to variations in waste stream chemistry. Public acceptance of the technology is enhanced because waste plastics can be used to treat wastes.

The appropriate plastics for mixed waste treatment are not currently being recycled in Colorado. While this provides the DOE an opportunity to put to beneficial use materials that would otherwise end up in a municipal landfill, it also presents some challenges. The main challenge is to identify collection, processing, and transportation methods that are economically viable.

Four options for collecting the appropriate waste plastics were presented in this paper. One of the options, collection of "regrind" from companies that manufacture plastic products, is currently used at Rocky Flats. This option costs significantly less than procurement of virgin polyethylenes, but Colorado does not directly benefit since the waste plastics are shipped from other states.

The best overall option is to collect waste plastics from local companies willing to collect and/or provide plastics to Rocky Flats as a public service. This option utilizes plastics that are currently being landfilled in Colorado while reducing (or eliminating) product costs to Rocky Flats.

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15-6

STABILIZATION OF INORGANIC MIXED WASTE TO PASS THE TCLP AND STLC USING CLAY AND pH INSENSITIVE ADDITIVES

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ABSTRACT

Stabilization is a "Best Demonstrated Available Technology" or "BDAT" as defined by the Environmental Protection Agency (EPA) in Title 40, part 268, of the Code of Federal Regulations (40 CFR 268). This technology traps toxic contaminants (usually both chemically and physically) in a matrix so that they do not leach into the environment. The typical contaminants that are trapped by stabilization are metals (mostly transition metals) that exhibit the characteristic of toxicity as defined by 40 CFR part 261. Pozzolanic materials have been routinely used in stabilization. Portland cement, fly ash-lime mixes, gypsum cements, and clays are some of the most common materials. These materials are used because they are inexpensive, easy to use, and effective for wastes containing low concentrations of toxic materials.

In many instances, passing the Toxicity Characteristic Leaching Procedure, TCLP, (i.e., Federal leach test) or the Soluble Toxicity Leachate Concentration, STLC, (i.e., CA leach test) requires high concentrations of lime or other caustic material because of the nature of the leaching media. Both leaching media, California's and EPA's, have a pH of 5.0. The State uses citric acid and sodium citrate. The EPA uses acetic acid and sodium acetate. These media also can form ligands that provide excellent metal leaching. In several circumstances, due to the aggressive nature of the leaching medium, stabilized wastes will not pass. The concentration in the leachate is approximately 10 times higher for the STLC procedure than the TCLP.

Additives such as dithiocarbamates and thiocarbonates, that are pH insensitive and provide resistance to ligand formation, are used in this stabilization process. Attapulgite, montmorillonite, and sepiolite clays are used because they are forgiving (recipe can be adjusted before the

matrix hardens) when formulating a stabilization matrix and they are neutral in pH.

Passing the TCLP and STLC has been achieved for highly concentrated wastewater treatment sludges by using these clays and additives. The most frequently used stabilization process consists of a customized recipe involving waste sludge, clay and dithiocarbamate salt, mixed with a double planetary mixer into a pasty consistency. TCLP and STLC data on this waste matrix have shown that the process matrix meets land disposal requirements.

INTRODUCTION

Stabilization is considered a Best Demonstrated Available Technology (BDAT) by the Environmental Protection Department and is the technology of choice for passing regulatory leachate tests such as the Toxicity Characteristic Leaching Procedure (TCLP) and the California Assessment Manual Wet Extraction Test for Soluble Toxicity Leachate Concentration (CAM-WET STLC).

The primary purpose for solidification/stabilization technologies for low-level mixed waste is to meet land disposal restrictions under the federal regulations (40 CFR 268).

To assure that the solidification/stabilization technology is acceptable, the processed material must pass the regulatory leach tests. The TCLP extract concentrations and the STLCs are applied to California waste when the waste is disposed at the Nevada Test Site and are applied to waste constituents which are given in the Constituent Concentrations In Waste Extract (CCWE) table (40 CFR 268.41) when the Universal Treatment Standards (40 CFR 268.48) apply. The CAM-WET in California is a much more aggressive test and applies to many more constituents such that frequent customization and verification of success is required.

EPA recognizes that Stabilization is a BDAT. Therefore stabilized wastes do not need to be sampled to verify they meet land disposal restrictions. However it is good practice to verify that the wastes pass STLC and verification is required by the Nevada Test Site for 10% of our wastes. Lawrence Livermore National Laboratory's (LLNL's) current work applies to wastes exhibiting the characteristic for toxicity and bear one or more of the EPA codes D004 through D011. California constituent radioactive wastes must also be treated, as a requirement of the Nevada Test Site, prior to disposal. Treatment of California constituent wastes, that are radioactive also, allows LLNL to develop techniques necessary to meet the universal treatment standards when promulgation requires treatment of underlying constituents for metal characteristic codes.

Currently, mixed wastes that exhibit the characteristics of corrosivity (D002), low total organic carbon (TOC) ignitability (D001), or toxicity for pesticides (D012 through D017) must be treated for all underlying constituents to meet land disposal requirements. Table I summarizes threshold limits for metals, EPA codes, and underlying constituents.

The waste requiring stabilization at LLNL is filter-aid sludge from processing wastewaters generated through various systems within the LLNL Site. This filter-aid sludge does not exhibit the characteristics which require LLNL to treat underlying constituents. However, to dispose of the waste at the Nevada Test Site, the State of California regulations apply which require LLNL to pass the STLC. The metals analyzed in the STLC are the same as the underlying constituents in EPA's Universal Treatment Standard. Table II gives the metals analyzed and the CAM-WET threshold limit for hazardous waste.

The differences between the Federal (TCLP) and the California State (STLC) leaching tests are subtle but provide substantial differences in test results. The differences in these tests for the stabilized wastes are summarized in Table III.

Examination of Table III shows that the California State leaching test is more rigorous in all categories but pH and Extraction Fluid Weight Ratio. The citrate buffer has greater soluble ligand formation properties than does the acetate buffer. Both anions form soluble complexes with metals but, citrate has much larger formation constants and can form bidentates and tridentates with metals in the presence of hydrogen (as in a pH of 5.0). Both buffer strengths are the same for this type of waste. Wastes, that are not pasty, are ground up to a certain particle size. The particle size for the STLC is 5 times smaller than for the TCLP. This provides the STLC with a steeper internal diffusion gradient. The leachate time for STLC is longer so that more contaminants leach out. There is twice as much extraction fluid in the Federal test but this may not be as significant. The more fluid does provide for a larger diffusion driving force but, this is not significant for lower concentrations (parts per million range).

Table I

Table II

Table III

RAW WASTE CHARACTERIZATION

The waste to be stabilized originates as aqueous waste. The metal constituents in the wastes are precipitated predominantly with hydroxide ion. The waste is then filtered through a rotary-drum vacuum-filter to remove the hydroxide precipitate. The rotary-drum vacuum-filter uses diatomaceous earth (filter-aid) to trap the precipitate. This spent sludge is periodically cut from the rotary drum during the aqueous waste treatment process. The filter-aid sludge collected is sent to our processing building for stabilization. The filter-aid sludge contains about 60% water, with the balance of the material being diatomaceous earth, metal contamination, and often organics, such as, oil and carbon. The metal contamination in the sludge varies widely from batch to batch, since the waste streams processed are widely varied. Typically one 5 cubic meter aqueous waste batch will yield one to two 0.2 cubic meter drums of diatomaceous earth waste. A typical example of the filter-aid sludge in a drum that requires stabilization is given in Table IV.

Table IV

The diatomaceous earth waste itself has little resistance against either the Federal or State leaching test. This is to be expected since hydroxide precipitates cannot hold up against any mild acid buffer regardless of its ability to form soluble complexes with metals. Metal hydroxide solubilities can be calculated from first principles using hydroxide formation constants, solubility products, and assuming unity for activity coefficients. Cadmium, lead, and zinc appear to be completely soluble at a pH of 5.

Table IV demonstrates the wide variety of metal constituents found in the filter aid sludge. The highest in the subset shown is nickel at 2076 mg/Kg. Usually the largest concentration of metal contamination in the waste sludge does not exceed 5,000 mg/Kg. The more concentrated sludges at LLNL are from spent plating baths (electro- and electroless plating). The waste from spent plating baths seldom contains arsenic, antimony, and selenium. These metals usually have to be precipitated as anion complex

and ion exchanged prior to filtration because these metals will not hydroxide precipitate. Fortunately, LLNL does not see much of this type of metal contamination.

THE CLAY MATRIX USED IN STABILIZATION

The primary clays used in LLNL's stabilization process are sepiolite, montmorillonite, and attapulgite. These clays have defined alumina or magnesium oxide-silica layers upon hydration. These clays were chosen because LLNL has worked with these materials in the past and these materials have demonstrated effectiveness in stabilization against TCLP. These clays tend not to increase the total volume of waste to the extent that other clays (e.g., Bentonite clay) do.

These clays possess the ability to hydrate and adsorb hazardous constituents. They form thixotropic fluids when hydrated and have minimal compression strength, but are considered solids from a regulatory standpoint (they pass EPA SW846 9095, Paint Filter Liquids Test).

Montmorillonites are impure forms of Al2O34SiO2H2O. The impurities are magnesium, potassium, calcium, titanium, and iron. The clay is an expanding type which forms a smectite when hydrated. The structure consists of an aluminum hydroxide octahedral in between two sheets of silica tetrahedral. The impurities in the structure are what gives this and other clays the ability to hold cations in place or "sorb" them. The metal impurities which replace silica (and in some instances the alumina octahedral) make the clay become deficient in positive charge. This causes the lattice to have a net negative charge and thus have the ability to "sorb" cations. During the formation of the clay, cationic impurities disrupt the clay matrix by replacing the alumina ions. This is most profound when the two aligning tetrahedral (above and below the octahedral) have substituted alumina for other metal oxides. In the case of this clay. the replacement cations have a less positive charge than the alumina. This results in a net negative charge in the clay lattice. The hydration reaction in its simplest view is given below.

Al2O34SiO2H2O + 2H2O 2[SiO2Al(OH3)SiO2]

Attapulgites and Sepiolites have a similar behavior to Montmorillonite except that the structures are different. Attapulgite and Sepiolite are not just alumina sandwiched between silica. They have ribbon-like structure in which alternating twists in the ribbon are silica and metal oxides. The structure is stable and replacement of the metal oxides is less evident than in montmorillonite. This gives less adsorption capacity but does provide a lower expansion upon hydration. The chemical composition of Attapulgite and Sepiolite is given below.

Attapulgite: (Mg)5Si8O20(OH)2(OH2)44H2O

Sepiolite: (Mg)9Si12O30(OH)6(OH2)46H2O

These formula are based on Nagy and Bradley model and are discussed in "The Chemistry of Clay Minerals", Charles E. Weaver, Elsevier, New York, 1975. These clays do contain aluminum as impurity but at a much lower composition than magnesium.

pH INSENSITIVE ADDITIVES

Since pH, complex formation, and diffusion are the primary driving forces for leaching in these two tests (TCLP, STLC), it is important to focus as much as one can on effective limitation to these phenomena. Molecular diffusion, although modeled in many situations and scenarios (sometimes very complexly), is small. Arguably, the diffusion of interest is between two solid phases, a successfully precipitated metal that is physisorbed to an active site, diffusing through a layer of clay. This is orders of

magnitude smaller than solid-liquid phase diffusion. Complex formation between the citrate and metal is relatively small. The formation constants of the bidentates and tridentates are orders of magnitude smaller that most inorganic solubility products. Acetate salt formation is even smaller still. Hydroxide precipitates do not hold up well against these mild acid buffers. The use of pH insensitive additives is required when high concentrations of metals are found in the sludge. There are many marketed precipitating agents under a variety of names. They are usually sold for waste water treatment. The primary non-hydroxide chemicals sold for precipitation are iron and sodium sulfide, thiocarbamate, and thiocarbonate. These chemicals are all relatively pH insensitive but still work better in alkaline solutions. The solubility products of sulfide or sulfur bearing organic salts are all much lower than hydroxides. Care must be taken in using these materials because they are toxic in their own right.

THE MIXING EQUIPMENT

The mixer used in this process is a double planetary open paddle mixer. It is a "change-can" mixing device which uses a standard 55-gallon (0.2 cubic meter) drum as the mixing vessel or change-can. Its power source is a 15 horsepower (11200 Watt) totally enclosed fan cooled (TEFC) motor operating at 1800 revolutions per minute. The motor shaft is connected to a worm that reduces speed by 40:1. The worm gear is attached to a shaft that is itself attached to a flat circular gear. This gear engages two gears attached to each other, floating freely about the center shaft. These two gears each have a shaft attached to an open paddle. The central shaft turns clockwise, spinning the two opposing gears and their paddles counterclockwise. As the two paddles spin counterclockwise about their own axis, they spin together clockwise about the center gear. Thus each paddle behaves as a planet spinning on axis, rotating around the sun (center gear). All three gears are approximately the same size and thus spin at about 45 revolutions per minute. This would appear slow (less than 1 revolution per second) if you did not see three motions occurring at once.

This mixing action is needed because the clay and diatomaceous earth mixture is a thixotropic pseudoplastic. It has extremely high viscosity until a high shear is applied and, it is a free standing monolith when it is not being forced to move. Low speed high shear mixing is more appropriate for this type of material than is high speed low shear mixing.

PROCESS DESCRIPTION

The process has been implemented using water that is already contained in the diatomaceous earth. The amount of stabilizer added was calculated to solidify the water in the sludge, not the entire weight of the sludge. Much of the preliminary product was too soupy due to water additions or due to not knowing how much water was in the sludge. Now, the first step before stabilization begins is a moisture determination. This is performed in a standard laboratory oven at approximately 600C and 600 grams of waste. The average moisture content is about 55% but it varies between 33% and 80% depending on the aqueous waste treated. The amount of clay added to the waste is determined by using the following formula:

$M = (0.6)(X)(Z)$

M = mass of clay to add

X = mass fraction of water in the waste

Z = mass of waste sludge

This formula gives a consistent, stiff matrix. It allows for easy cleaning of the equipment because the clay matrix sticks to itself much more than to the open paddles. It also proves not to readily dehydrate when sealed in a drum to cure.

Additives are used when metal concentrations are too high to just use the clay. In many instances clay alone provides little fixation. Table V shows the original raw material concentrations of metals with the STLC values after stabilization. A fair comparison can be made if one takes the STLC value, multiplies the value by 10 (this is the dilution caused by the citrate buffer), then multiples by the ratio of stabilized net mass to mass balance net mass. As one can see, fixation does not occur in many cases and dilution is the main effect (values for zinc demonstrate this in Table V). Figure 1 shows a good way to present data to demonstrate the fixation of a metal constituent. The total constituent mass is calculated, then the percentage of the amount of metal leached is calculated. A direct comparison of the amount leached in the original waste form and the stabilized waste form can then be made.

Table V

PROCESS ANALYSIS

Figure 1 shows the results of stabilization with clay only. For many metals at moderate concentrations, fixation with clay is adequate to perform the fixation. Cadmium and cobalt show an order of magnitude reduction in leachate concentration upon stabilization with clay only. Nickel shows greater than one order of magnitude reduction and copper shows varying reduction in leachate concentration up to two orders of magnitude.

If the waste sludge contains greater than a few hundred grams of the metals mentioned above or contains metals such as chromium, molybdenum, vanadium, and zinc, additives are required to perform the stabilization successfully. Additives were first added at stoichiometric ratios plus 10% excess. This proved unsuccessful. Often stabilization was not successful unless 100 times stoichiometry was used. Currently, 12% by weight of waste is used and only dithiocarbamate (DTC) has been successful at this concentration.

Fig. 1

LLNL found that the sequence of stabilization is very important and that two mixing stages are needed. First, the pH insensitive precipitating agent is added, the waste sludge and additive are allowed to mix. The mixing continues for two to five minutes. The waste and additive are well mixed at this point and the reprecipitation has occurred. Next the clay is added and mixed. This locks the reprecipitated metals in the media and physisorbs it and any other materials still free in the matrix.

Figure 2 below shows a similar chart for stabilized wastes using dithiocarbamate. In every case except for barium, fixation has occurred. There was essentially no detection of nickel or silver in the leachate of stabilized wastes. This clay and dithiocarbamate showed very low leachate fractions for metals accept for arsenic which does not readily precipitate with DTC and usually exists in anionic form in wastewaters. Nevertheless, some reduction in leaching was observed.

Figure 3 shows the decrease in leachate concentration with the addition of DTC. The decrease is dramatic when the DTC additive is between 3% and 12% by weight of raw waste. These wastes originally had 3200 mg/Kg Nickel, 745 mg/Kg Copper, 429 mg/Kg Zinc. Leachate concentrations for

these wastes after stabilization with clay and dithiocarbamate were less than 1 mg/L; all stabilized wastes passed the TCLP and STLC.
Fig. 2
Fig. 3
CONCLUSION
In many cases the use of clay alone can fix metal constituents in a waste form. In other cases, fixation must be augmented. Using clay alone when metal concentrations are high will not fixate metals enough to pass the STLC. The pH insensitive additives, such as dithiocarbamate or other sulfide bearing compounds, may need to be used. Without the use of pH insensitive additives, metal salts become available to the leaching fluid because the interaction of the clay and the metal hydroxide is weak. With the salts available and at a pH of 5.0, metal hydroxide dissolves into the bulk extractant in the form of citrate or acetate salts and salt complexes. The addition of DTC at concentrations between 6 and 12% of the raw waste precipitate metals so that the clay matrix passes the STLC.

15-8
SURVEY OF COMMERCIAL FIRMS WITH MIXED-WASTE TREATABILITY STUDY CAPABILITY*
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ABSTRACT
According to the data developed for the Proposed Site Treatment Plans, the U.S.Department of Energy (DOE) mixed low-level and mixed transuranic waste inventory was estimated at 230,000m3 and embodied in approximately 2,000 waste streams. Many of these streams are unique and may require new technologies to facilitate compliance with Resource Conservation and Recovery Act disposal requirements. Because most waste streams are unique, a demonstration of the selected technologies is justified. Evaluation of commercially available or innovative technologies in a treatability study is a cost-effective method of providing a demonstration of the technology and supporting decisions on technology selection.
This paper summarizes a document being prepared by the Mixed Waste Focus Area of the DOE Office of Science and Technology (EM-50). The document will provide DOE waste managers with a list of commercial firms (and universities) that have mixed-waste treatability study capabilities and with the specifics regarding the technologies available at those facilities. In addition, the document will provide a short summary of key points of the relevant regulations affecting treatability studies and will compile recommendations for successfully conducting an off-site treatability study.
Interim results of the supplier survey are tabulated in this paper. The tabulation demonstrates that treatment technologies in 17 of the U.S. Environmental Protection Agency's technology categories are available at

commercial facilities. These technologies include straightforward application of standard technologies, such as pyrolysis, as well as proprietary technologies developed specifically for mixed waste. The paper also discusses the key points of the management of commercial mixed-waste treatability studies.

INTRODUCTION

Treatability studies represent a ready vehicle for U.S.Department of Energy (DOE) waste managers to evaluate and demonstrate both available and innovative treatment technologies on mixed-waste streams. The U.S.Environmental Protection Agency's (EPA) Resource Conservation and Recovery Act (RCRA) regulations (Title 40, Code of Federal Regulations, Part 261 and following) exempt limited volumes of waste from treatment in fully permitted facilities for the purposes of technology screening and process data compilation. In most states, up to 1,000 kg of hazardous waste and up to 10,000 kg of media contaminated with hazardous waste can be treated under the RCRA treatability study exemption. The Toxic Substances Control Act (TSCA) regulations are less specific on this point but recognize the need.

COMMERCIAL MIXED-WASTE TREATABILITY STUDY SUPPLIERS

To provide a service to DOE's waste managers, the Mixed Waste Focus Area (MWFA) of the DOE Office of Science and Technology (EM-50) is undertaking a survey to identify firms with current capability in the treatment of mixed waste at the treatability study scale. The survey has compiled candidate firms from three sources of information. The EPA's Vendor Information System for Innovative Treatment Technologies (VISITT) database was queried for firms with a background in the treatment of both hazardous and radioactive waste. The DOE Tech-Con database, resident at Argonne National Laboratory, was also reviewed. Finally, Lockheed Martin Idaho Technologies published an announcement in the Commerce Business Daily to identify any new sources.

Over 100 firms were identified from these three sources, and a representative from the MWFA contacted all firms by phone to determine whether they have current capability and interest in participating in the survey. Subsequently, the interested firms were sent questionnaires to characterize their capability and history in mixed-waste treatability studies. Contact information of some firms was not correct, and those leads were deleted from further consideration. To identify and confirm each firm's specific capability, the questionnaire asked for the following:

- Name of facility and facility contact
- Technologies available through that firm
- Waste matrices/hazardous constituents treated by that firm
- Copies of radioactive material licenses (to identify curie limits in acceptance)
- Relevant other limits of their waste acceptance criteria
- Demonstration unit capacity
- Treatability study history
- Lists of publications or technical papers presented on the results of that firm's treatability studies.

Information from the survey forms has been received, and pertinent information has been compiled into a database. At this time, the survey results are considered "work in progress," as a number of firms failed to respond and some responses raise questions. In addition, the MWFA intends

to categorize the companies into several groups after detailed review of the questionnaires.

Table I presents the information received (to date) according to the general treatment technology category available from the firms. It also lists the firms in each technology category along with their contact names and phone numbers. This table was compiled directly from the questionnaire results with limited validation.

Table Ia

Table Ib

Table Ic

Table Id

Table Ie

DOE facilities interested in complete questionnaire responses should contact the MWFA at the Idaho National Engineering Laboratory in Idaho Falls, Idaho.

The following conclusions result from the data presented:

Eighty-two processes are available in 17 of the EPA's general technology categories.

Responses have included both waste management and environmental restoration technologies.

At least three different categories of firms responded to the questionnaire. A limited number have current capability for mixed-waste treatability studies in existing processing systems with U.S. Nuclear Regulatory Commission (NRC) (or state) permits for radioactive materials and have either treatability study capability or full RCRA operating permits. A second group of respondents has technologies available, but mixed-waste treatability studies are available at permitted facilities owned by others. A third group identified technologies but made no indication of mixed-waste treatability study capability.

Several firms expect an NRC or state radioactive materials license very soon.

Many firms responded positively to the telephone screen but did not return the completed questionnaire.

Radioactive license limits are typically radionuclide-specific, but total curie possession as high as 10 is available.

Two commercial suppliers identified the ability to accept plutonium in gram quantities.

Prior to issuance of the report from the MWFA, several additional checks will be made. Data will be presented to identify the firms by category: those with fully permitted facilities, those with a technology and access to fully permitted facilities, and those with a technology only. Note that the MWFA requested copies of relevant permits and licenses. This information will be used to confirm capabilities, where appropriate. In addition, each firm will have an opportunity to correct or expand on information presented. After these data checks, the supplier data are expected to be considered complete for publishing.

TREATABILITY STUDY MANAGEMENT ISSUES

The MWFA document also provides recommendations for conducting mixed-waste treatability studies. This information was compiled from several EPA guidance manuals and discussions with DOE waste managers who have contracted for off-site treatability studies as well as from reports generated within DOE. The document discusses the issues in detail. The following are some key points taken from the discussion:

The document generally advises that each of DOE's waste streams is in some fashion unique, and a quick confirmation of the applicability of a technology is excellent insurance against process failure.

Several of the EPA guidance documents (1, 2, 3, 4, 5) address the development of the overall project objectives. Generally speaking, the level of data quality should be adjusted to reflect whether the treatability study is a simple screening test or one that is performed to develop complete process design data.

The availability of commercial suppliers of a technology and treatability study can significantly affect cost. In particular, the availability of an existing pilot plant can offer a significant cost and schedule advantage.

Residues disposition should be carefully considered. Generally, the residues from a treatability study are subjected to a hazardous waste determination, and it is important that the original generator not be required to perform additional treatment on the residues.

Jolley et al. (6) recommends a readiness review to ensure that all contractual, regulatory, and health and safety issues be considered prior to conducting the test.

At least one run of a treatability study should be observed to ensure compliance with the test plan.

A management process for approving quick changes in the treatability study should be implemented. Frequently, project changes may be warranted based on interim results, and a rapid change approval expedites the treatability study completion.

TREATABILITY STUDY REGULATORY FRAMEWORK

DOE mixed-waste treatability studies, conducted in accordance with the RCRA, are potentially subject to numerous other national and state regulations, as well as to various DOE Orders. Treatability studies dealing with environmental restoration wastes generated at sites regulated under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) will be subject to CERCLA requirements. If the treatability study involves polychlorinated biphenyls, the requirements of the TSCA will apply. If the study is conducted at a commercial treatability study laboratory or treatment facility, the facility must have a received radioactive materials license from the NRC. Waste samples transported to off-site facilities by common carriers must meet packaging requirements of RCRA, the NRC, and the U.S. Department of Transportation. DOE Orders on the management of a commercial facility will also apply. The DOE is also responsible for complying with the provisions of the National Environmental Policy Act.

The MWFA document will discuss each of these regulatory frameworks separately and will summarize pertinent issues to be addressed.

CONCLUSION

Treatability studies represent a viable option for demonstrating available and innovative technologies on DOE mixed wastes. The MWFA will continue to collect information from treatability study firms, analyze the data, and verify capabilities before publishing the results. Interim results from the survey indicate that there are a number of commercial and university suppliers who can currently demonstrate technologies in their own licensed facilities, and there are many other technology owners with access to facilities with the appropriate permits and licenses.

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15-10

PRELIMINARY TREATABILITY STUDIES FOR POLYETHYLENE ENCAPSULATION OF INEL LOW-LEVEL MIXED WASTES

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ABSTRACT

Preliminary treatability studies were performed during fiscal year 1995 (FY95) at Brookhaven National Laboratory (BNL) for polyethylene encapsulation of low-level mixed wastes from Idaho National Engineering Laboratory (INEL). Treatability work included thermal screening and/or processibility testing to determine the applicability of polyethylene encapsulation for solidification of priority wastes identified by INEL. Seven candidate wastes were selected and investigated during this study. However, for purposes of this paper and to provide a thorough review of the experimental method, processing results will only be presented for the highest priority INEL waste; Eutectic Salts (INEL ID#SMC-507). Thermal screening was conducted by Differential Scanning Calorimetry for this waste to assess the thermal stability under pretreatment and polyethylene processing conditions. Processibility testing to determine whether the waste is amenable to extrusion processing included monitoring feed consistency, extruder output consistency, waste product homogeneity, and waste form performance. A nominal waste loading was selected and successfully processed for a eutectic salt surrogate as well as an actual eutectic salt mixed waste sample. Both surrogate and mixed waste sample waste forms showed marked improvement in lead leachability during Toxicity Characteristic Leach Procedure (TCLP) testing and were below

levels specified by the Environmental Protection Agency (EPA) under the Resource Conservation and Recovery Act (RCRA).

INTRODUCTION

Treatability studies for polyethylene encapsulation of Idaho National Engineering Laboratory (INEL) low-level mixed wastes were conducted at the Brookhaven National Laboratory (BNL) Environmental & Waste Technology Center.(1) The Technical Programs Unit, part of the Waste Management Division of Lockheed Idaho Technologies Company (LITCO), is investigating the use of polyethylene for waste encapsulation in an effort to develop mixed waste treatment alternatives. BNL has developed the extrusion-based polyethylene encapsulation process as an effective waste solidification technology and has demonstrated applicability to a wide range of hazardous, low-level radioactive and mixed wastes.

Treatability work was performed on seven INEL identified priority candidate wastes and included thermal screening and/or processibility testing. The processing results will be presented in this paper only for the highest priority INEL waste stream so that a thorough review of the process and the experimental method may be given. The seven wastes, generated during ongoing processing and maintenance operations or one-time cleanup operations, have been physically and chemically characterized at INEL. Based on the characterization data, surrogates were prepared for each waste stream that closely resembled the actual wastes in chemical and physical form. For three waste streams, including the one discussed in this paper, actual mixed waste samples were processed for confirmation and proof of technique.

The candidate wastes are classified as mixed wastes due to the presence of radionuclides and heavy metals or organics. The seven candidate wastes in order of priority are shown in Table I. The order of priority for this list was determined based on the volume of the waste and the currently available treatment options for each waste. The top three wastes are generated by ongoing activities and therefore are the highest priority. Of these, ion exchange resins and activated carbon may be treated at the INEL Waste Experimental Reduction Facility (WERF) mixed waste incinerator, but treatment alternatives for the eutectic salt waste stream are more limited.

Treatability work for each waste involved thermal screening by Thermal Gravimetric Analysis (TGA) or Differential Scanning Calorimetry (DSC) and processibility testing including any required waste pretreatment. Thermal screening was used to identify wastes containing compounds that may be volatile at either the thermal pretreatment or extrusion processing conditions (120-150C), or to identify wastes that may undergo a phase transition at thermal processing conditions. A surrogate was used during thermal screening testing. Processibility testing to determine whether the waste is amenable to extrusion processing included monitoring material feed, processing consistency, waste product homogeneity and waste form performance. A nominal waste loading was selected and processed successfully for a eutectic salt surrogate as well as an actual eutectic salt mixed waste sample. Processing parameters were not optimized within the scope of this study since the primary focus of this preliminary effort was to simply establish whether these wastes were amenable to polyethylene encapsulation.

Table I

BACKGROUND

Waste Description

Treatability work was conducted on a surrogate of the eutectic salt waste and on a sample of the actual INEL priority mixed waste. This waste stream is also known as Carbonate Eutectic Salt or more recently as Cartecsal. The waste description as provided by INEL is shown in Table II. It is a lead (Pb) contaminated salt mixture roughly comprised of 60 weight percent potassium carbonate, 30 weight percent lithium carbonate, and 10 weight percent sodium carbonate. Characterization conducted at INEL indicated that the salt mixture was a fine powder with TCLP analysis indicating a lead concentration between 4.8-5.8 mg/l. There are currently approximately 29 drums of this waste being stored at INEL.

Table II

Polyethylene Encapsulation Process Description

Polyethylene is an inert, low permeability, thermoplastic material that is highly resistant to chemical attack, microbial degradation and radiation damage.(2) The polyethylene micro-encapsulation process selected for eutectic salts is a solidification technology utilizing polyethylene to solidify waste particles within a polymer matrix. This is accomplished by heating polyethylene above its melting point (120-150C), then mixing the molten polyethylene with the waste. Solidification of the encapsulated matrix is assured upon cooling and is independent of the chemical nature of the waste materials. The BNL polyethylene encapsulation process uses a versatile, industry-tested single-screw plastics extruder. A schematic of the microencapsulation process is shown in Fig. 1. Dry waste and polyethylene are continuously fed to the extruder by individual dynamic feeders and accurately metered to maintain a constant waste loading. The feeders are either volumetric, designed to meter a constant volume of material at a given speed setting, or volumetric feeders retrofitted to a computer-controlled loss-in-weight system. Within the extruder, a variable speed screw mixes the molten polyethylene with the waste material and pumps the homogeneous mixture through a die directly into a waste container. The processing temperature is controlled through five independent extruder zones and a separate die zone. For optimization, the temperature profile can be tailored for different polymer and waste combinations.

Pretreatment is often required of wastes to make them amenable to extrusion processing with polyethylene. The degree of pretreatment is specified by the chemical and physical characteristics of the untreated waste. For successful microencapsulation and to achieve optimum mixing and product/processing consistency, wastes should be dry (less than 2% moisture) and have a nominal particle size distribution of 50-2000 mm. The particle size range may exceed these limits but successful processibility is dependent on the material feeders, extruder and type of screw(s) employed, extruder die configuration, and waste product performance considerations (e.g., leachability). Larger particles that may be processed from an engineering standpoint can reduce the effectiveness of microencapsulation which relies on the coating of individual waste particles. Thus, for any given waste, the larger the particle size, the greater the surface area of unencapsulated waste, and the greater the potential for leaching.

Fig. 1

TREATABILITY STUDY

Surrogate Preparation

The eutectic salt surrogate was prepared according to the composition data provided by INEL. As mentioned, the waste is primarily comprised of

the salts potassium carbonate, lithium carbonate and sodium carbonate. Potassium and sodium carbonate are available as hydrated compounds. Potassium carbonate can be in an anhydrous or sesquihydrate (K2CO31H2O) form. Sodium carbonate can be anhydrous or monohydrate (Na2CO3H2O). Lithium carbonate (Li2CO3) does not complex with water. The compound form of the salts in the actual eutectic salt mixed waste is not known. For thermal screening, the surrogate was prepared without lead using reagent grade hydrated salts. Technical grade anhydrous carbonate salts were used for surrogate preparation for processibility confirmation testing. For treatability testing, the surrogates were spiked with lead at a concentration prescribed by the INEL characterization data, as shown in Table II, that indicted a maximum lead concentration of 5.8 mg/l from TCLP results. Accounting for a 20:1 dilution, a TCLP concentration of 5.8 mg/l is equivalent to 116 ppm in the waste. Thus, the surrogate was spiked with 120 ppm lead in the form of lead chloride.

Thermal Screening

Thermal screening of the eutectic salt surrogate was conducted using a Shimadzu DSC-50 Differential Scanning Calorimeter to determine if waters of hydrations would evolve from the salts at the thermal processing conditions expected during polyethylene encapsulation (120-150C). For DSC analysis, samples weighing approximately 20 mg were heated in the presence of nitrogen from ambient to 150C at 3C/min and held for 40 minutes. A slow heating rate (1-5C/min) is suggested by the manufacturer for observing phase changes due to free or bound water evolution. Each individual carbonate salt and the homogeneous mixture were tested under these conditions. The DSC results, plots of energy (mW) versus time (min), revealed endothermic peaks representing the evolution of bound water for potassium carbonate sesquihydrate and sodium carbonate monohydrate. Figure 2 is a DSC plot that clearly shows the evolution of the water of hydration from sodium carbonate monohydrate. The heating profile can also be discerned on this plot. Similar plots were obtained for potassium carbonate sesquihydrate and the eutectic salt surrogate. The eutectic salt surrogate showed an endothermic peak since it is comprised of the hydrated potassium and sodium carbonate salts. However, lithium carbonate (nonhydrous) did not reveal any phase change for the same heating profile, as expected.

To support and verify DSC results as well as determine the moisture content of the salts, three replicate samples of the eutectic salt surrogate were dried using a Sartorius Moisture Analyzer. Each test was conducted for 30 minutes at 150C. This is the extreme temperature that would be encountered during encapsulation processing and is longer than the approximate 10 minutes residence time in an extruder barrel that the waste would be exposed to during processing. A mass/mole balance was performed in order to determine if the weight loss corresponded to the quantity of bound water in the surrogate containing the hydrated salts. The mean weight loss from the three replicate samples of the eutectic salt surrogate was 11.6 0.27 percent. On a molar basis, hydrated water represents 16.4 weight percent of the compound potassium carbonate sesquihydrate (K2CO31H2O) and 14.5 weight percent of the compound sodium carbonate monohydrate (Na2CO3H2O). The weight percent of water in the eutectic salt surrogate is calculated by multiplying these percent numbers by their respective composition percents in the surrogate (i.e., 0.6 * 16.4 + 0.10 * 14.5) and is equal to 11.3 percent. This number is in close agreement with the weight loss observed on the moisture analyzer.

Based on these results in conjunction with the DSC results, it can be assumed that if the salts comprising the actual eutectic salt mixed waste are hydrated, all waters of hydration will evolve at the temperature required for extrusion processing. Therefore, pretreatment of the waste by drying should be performed prior to extrusion processing.
Fig. 2
Processibility
Processibility testing of the non-radioactive eutectic salt waste surrogate was conducted using a vented, 38 mm (1.5 in.) single-screw extruder. The vent port used along with a carbon trap and vacuum pump removed any volatiles (e.g., moisture) that may have evolved during processing. The INEL mixed waste eutectic salt was processed using a 32 mm (1.25 in.) single-screw extruder contained in an enclosed HEPA filtered process facility. Feeding of the mixed wastes was accomplished through dry material volumetric feeders, and through volumetric feeders retrofitted to a loss-in-weight computer controlled system for the waste surrogates. The standard volumetric feeders were calibrated with the eutectic salt mixed waste and with low-density polyethylene pellets (Chevron 1409). Calibration curves were generated by recording the feeder output (in weight) at different feeder screw speed settings (percent of full speed). Five replicates were taken at a minimum of three separate feeder speed settings. The loss-in-weight feed system consisted of a master feed controller and two slave controllers (one for waste and one for polyethylene). Feeding was accomplished by simply entering the desired total feed rate as well as a recipe for the waste loading (percentage of waste compared to percentage of polyethylene) into the computer controllers.
Extrusion processing for the surrogate and for the mixed waste sample was conducted at a waste loading of 50 weight percent and at a screw speed of 25 rpm. These are not optimized nor maximum processing parameters but rather a conservative starting point that, based on previous experience with waste with similar physical properties, can be readily achieved. Processibility testing with the surrogate included fabricating right cylindrical specimens for compressive strength testing, taking periodic grab samples for density analysis (to monitor waste product homogeneity), and timed one minute samples for processing consistency. Ten replicates were made of each sample for statistical analysis. The densities of the grab samples were determined by weighing each grab sample and using a multipycnometer to determine their volumes. Right cylindrical pellets with a maximum dimension of nine millimeters were also formulated with the lead-spiked surrogate to monitor lead leachability from waste forms by TCLP testing. Processibility testing with the actual eutectic salt mixed waste was limited to confirmation processing while fabricating waste form pellets for TCLP testing. Nearly all of the mixed waste sample received from INEL was used for this task.
Processing of the eutectic salt surrogate was successful. Table III summarizes the extruder output rate and is representative of the processing consistency. The mean output rate at a screw speed of 25 rpm was 77.30 2.00 g/min (95% confidence interval). The small percent error of 2.58% is an excellent reflection on how well the material processed. Generally, materials that vary in output rate as much as 15-20% can still be successfully processed. Feeding and metering of the eutectic salt surrogate was also smooth and consistent. The waste form densities which were calculated from the grab samples taken during processibility testing

are shown in Table IV. The mean density for waste forms containing 50 wt% eutectic salt surrogate was 1.35 0.01 g/cm3. The small density deviations between replicate samples and the low 1.1% error are an indication of the high degree of homogeneity in the waste product.
Table III
Table IV
Waste Form Performance
Eutectic salt waste form specimens were compression tested in accordance with ASTM D-695, "Compressive Properties of Rigid Plastics."(3) The waste form specimens, nominal 5.1 cm (2 in.) by 10.2 cm (4 in.) right cylinders, were turned on a lathe to ensure flat surfaces on the top and bottom then compressed at a constant loading rate of 1.0 0.5 mm/min until the waste form failed or the load within the samples no longer increased with continued loading. The recorded compressive strengths for the ten replicates are shown in Table V. The maximum compressive strength of the waste forms averaged 1.56x104 248 kPa (2260 36 psi). The maximum compressive strength for these waste forms is similar to other polyethylene final waste forms and is well above the NRC waste form compressive strength requirement of 414 kPa (60 psi).
Table V
The leachability of lead from eutectic salt surrogate and actual mixed waste eutectic salt waste forms was conducted in accordance with 40 CFR 261--Identification and Listing of Hazardous Wastes, Appendix II--Toxicity Characteristic Leaching Procedure.(4) The TCLP is an 18 hour extraction of 100 g of sample in 2000 g of a buffered leachant. The leachant or extraction fluid is selected based on a pre-test which determines the pH and the buffering capacity of the sample. An extraction fluid with a pH of 2.88 is used for samples with good buffering capacity and pHs above five otherwise an extraction fluid with a pH of 4.93 is used. Test samples are agitated in an end-over-end fashion for 30 2 rpm for the test duration. On completion, the solutions are filtered into approximately 100 ml aliquots using a 0.6-0.8 micron glass fiber filter. During this study, the concentration of lead in the aliquots was determined by Inductively Coupled Plasma Spectroscopy (ICP) for the nonradioactive surrogate (spiked with 120 ppm lead) and by Hach UV-Spectroscopy for the mixed waste sample. For both TCLP tests, encapsulated waste forms were compared with the untreated samples which served as baselines.
TCLP results are summarized in Table VI. All tested samples were within the RCRA limit of 5 mg/l for lead concentration. The concentration of lead in the encapsulated surrogate (spiked with 120 ppm lead) leachate was undetectable based on two replicate ICP Spectroscopy analyses, compared with concentrations of 4.1 and 4.2 mg/l for the untreated surrogate. The detection limit of the ICP Spectrometer for lead is 0.14 mg/l. A marked improvement in leachability was also seen in samples of the encapsulated eutectic salt mixed waste. Based on HACH UV-Spectroscopy, the baseline or untreated mixed waste sample leached to a lead concentration of 4.1 mg/l compared to 1.8 mg/l for the encapsulated mixed waste samples. Although the current maximum allowable TCLP concentration for lead is 5.0 mg/l, the EPA has recently proposed reducing the allowable concentrations for most Toxicity Characteristic (TC) metals. The new proposed standard for lead is 0.37 mg/l. The polyethylene encapsulated surrogate would pass these new limits but the encapsulated mixed waste sample with a value of 1.8 mg/l would be above

the proposed acceptable limit. To meet the new lower limit, process optimization would need to be performed as well as possibly adding chemical stabilizers during solidification.

Table VI

CONCLUSIONS

Processing of the eutectic salts was successful. This waste is amenable to stabilization by polyethylene microencapsulation. Due to the limited scope of this study, processing parameters were not optimized. At a waste loading of 50 weight percent the leachability of encapsulated waste forms was effectively reduced below current RCRA limits. Higher waste loadings should be achievable from a processing standpoint and based on the successful waste form testing results achieved to date. Thermal screening showed that potassium and sodium carbonate may contain waters of hydration that would require pretreatment drying to prevent volatilization during extrusion processing. Additional processibility work on this waste stream would enable optimization of processing variables and maximize waste loading potential.

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15-12

BENCH- AND PILOT-SCALE DEMONSTRATION OF TREATMENT OF MIXED WASTE SEDIMENTS BY THERMAL DESORPTION AND SOIL WASHING

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ABSTRACT

The 1992 Oak Ridge Reservation FFCA listed a number of mixed wastes, subject to LDR, for which no treatment method had been identified, and required DOE to develop strategies for treatment and ultimate disposal of those wastes. Thermal Desorption and soil washing are two emerging commercial technologies that have typically been used for removing

hazardous organics and heavy metals from contaminated soils, sediments and sludges that are similar to many of these wastes.

This paper presents the results of a program to demonstrate that these technologies can remove both organics, uranium and mercury from a mixed waste from the DOE Y-12 facility in Oak Ridge. TN. This mixed waste sediment from the plant storm sewer system contained mercury, uranium and PCBs. Leachable mercury exceeded TCLP and LDR criteria. This program included bench- and pilot-scale testing of thermal desorption technology. This study also included testing of a soil washing process using physical separation and carbonate extraction to remove uranium. The pilot-scale testing was supported by DOE and EPA through the SITE Emerging Technology Program.

Results of the bench-scale testing of the thermal desorption technology showed that total mercury could be reduced to 120 ppm by treatment at 600oC, which is at the high end of the temperature range for typical thermal desorption systems. Leachable (TCLP) mercury was less than 50 ppb and PCBs were below 2 ppm. Mercury removal in the pilot-scale thermal desorption system was not as high, most likely due to kinetic effects of mercury oxide compound decomposition or inadequate flow of purge gas in the thermal desorber tube. Treated soil from the pilot test did pass TCLP for mercury. The pilot-scale thermal desorption treatment was effective in removing PCBs. The testing also provided information on the characteristics and quantities of residuals from the thermal desorption process. The soil washing process resulted in removal of roughly 50% of the uranium from the sediments and the carbonate extraction was only marginally effective, indicating that the thermal treatment converted uranyl compounds to less soluble species. This program demonstrated that thermal desorption was an effective technology for treating this material to meet waste acceptance criteria for ultimate disposal at a commercial facility.

INTRODUCTION

This project included both bench and pilot-scale testing of the following technologies; thermal desorption, soil washing using physical separation, and chelant extraction. While bench scale testing required less than 10 kilograms of soil, almost 1,000 kilograms were used for the pilot-scale work. The soil for these tests was obtained from the Department of Energy's Y-12 facility, located in Oak Ridge, TN. and operated by Martin Marietta Corporation. This material contained hazardous organics, heavy metals, and radionuclides.

The primary goal of the bench scale tests was to identify optimum operating conditions and performance for the various treatment technologies. The pilot scale tests then provided additional performance and scale-up data needed for evaluation of the potential for full scale application of these technologies. This paper presents the results of these studies.

Waste Characteristics

The Y-12 Storm Sewer Sediment waste was generated during a clean-up and upgrade of the storm sewer system of the DOE Y-12 facility in Oak Ridge, Tennessee. It consisted of sand and silt that had collected in the plant storm sewer lines and contained hazardous organics, heavy metals, and radionuclides. IT received four 55-gallon drums of the waste at the IT Environmental Technology Development Center in Oak Ridge, TN. The material was mixed and homogenized before analysis and testing. Average mercury content in the storm sewer sediment was 22,750 mg/kg. Total

activity of the sample was 9,000 pico-curies per gram for total uranium. The initial PCB concentration of the sample was 149 mg/kg.

Treatability Study Objectives

The primary purpose of the treatability study was to demonstrate that the thermal desorption and soil washing process could meet the performance requirements for removal/reduction of PCBs and mercury from the storm sewer sediments. The performance requirements or treatment criteria for the waste were 12 mg/kg total mercury, 2 mg/kg PCBs and 0.2 mg/liter leachable mercury, as measured by the TCLP. The 12 mg/kg treatment standard for total mercury was a risk based level for disposal in an on-site landfill. Alternate treatment criteria, based on disposal at an off-site permitted radioactive mixed waste facility, required that the material meet the LDR total mercury limit of 260 mg/kg and also meet the TCLP and 2 mg/kg PCB limits.

A secondary treatment objective was to remove the uranium from the sediments so that they could be managed as a non-radioactive material. This study was also designed to generate limited data on the fate of the mercury, PCBs and uranium during thermal treatment and on other contaminants in the residuals from the off-gas treatment system.

BENCH-SCALE THERMAL DESORPTION TESTING

Bench-scale thermal tests were performed to confirm the effect of primary treatment variables (time and temperature) on the reduction of mercury and hazardous organics. The tests also enabled an assessment of the characteristics of the residuals collected from the off-gas, an important factor for the preparation of the pilot tests. The bench thermal treatment program included screening tests in the tube furnace apparatus and tests in the rotary thermal apparatus (RTA).

The tube furnace tests were conducted in a 1-inch diameter by 16-inch quartz tube placed in an electric furnace. The purge gas was treated with a chilled condenser and sulfur impregnated carbon. Three different purge gases (air, nitrogen, and steam) were tested for mercury removal. Based on IT's previous experience, two separate treatment temperatures of 450C and 550C were selected with treatment times of 10, 15, and 20 minutes. Additional screening tests were conducted using the tray test apparatus and a muffle furnace. These tests were conducted to rapidly screen an expanded range of treatment conditions. Based on the results of the tube furnace analysis, temperatures of 550C, 600C, 650C, and 700C were tested at a treatment time of 10 minutes. In addition to these tests, treatment times of 10, 20, 40, 60, and 90 minutes were tested at a 550C temperature.

The results of the tube tests show that the total mercury concentrations were readily reduced to 50 to 100 mg/kg. These results also show that different atmospheres (the three different purge gases) had no significant effect on the mercury levels in the treated soil. The TCLP analyses show leachable mercury was reduced to below 0.011 mg/Liter. In the system's purge gas condenser, elemental mercury was collected. As for the organics, PCB concentrations were typically below 1 mg/kg. These results met the LDR criteria of 260 mg/kg total mercury, but not the risk based level of 12 mg/kg necessary for an on-site landfill.

A series of tray tests at higher temperatures and known treatment times were conducted in order to try and meet the 12 mg/kg total mercury criteria. The treated soil from the tray tests contains concentrations of total mercury between 70 and 115 mg/kg. The more aggressive treatment conditions did not consistently result in lower mercury levels. As the

test temperature was increased, the mercury concentrations were reduced in the treated soil. The test at the highest temperature (700C, 10 minutes RT) did produce the lowest mercury number at 72 mg/kg. However, the results of the varying treatment times were not consistent. The longer contact times at temperature did not provide any significant benefit.

The third phase of the bench, thermal treatment program was the RTA analysis. The RTA is a bench-scale system, basically a small, batch kiln, capable of treating up to 1 kg of soil. The RTA provides scaleable data for the pilot test conditions. The off-gas was treated by chilled impinger scrubbers and sulfur impregnated carbon. Two duplicate runs were conducted at two sets of treatment conditions. The first set of conditions required a temperature of 600C, with a time of 10 minutes at temperature, and an air purge at a rate 1.8 liters per minute. The second set of conditions had a temperature of 350C, with a time of 10 minutes at temperature, and a steam purge at a rate 3.0 liters per minute.

The first set of conditions (600C, 10 minutes RT, air purge) reduced the PCBs and leachable (TCLP) mercury levels to below detection limits. The total mercury concentrations of the treated soil were 141 and 82 mg/kg for the duplicate runs. The second set of conditions (350C, 10 minutes RT, steam purge) resulted in residual PCB concentrations of 2.5 and 3.2 mg/kg for the duplicate runs. The total mercury in the treated soil was 3700 and 4100 mg/kg, but the leachable mercury was below the detection limit of 0.002 mg/liter.

PILOT-SCALE THERMAL DESORPTION TESTING PROGRAM

After the bench-scale desorption test results had been evaluated, approximately 1000 lbs. of contaminated soil were thermally treated at 600C with a 10 minute retention time. The primary goal for the pilot-scale operations was to provide performance and scale-up data for full scale application and to provide sufficient material for soil washing and water treatment studies

Pilot System Description

IT's pilot-scale thermal desorber is an indirectly heated, rotary tube calciner (desorber) unit employing a continuous feed and a multistage off-gas treatment system. The pilot system has in past studies given comparable results to bench-scale tests. The thermal desorber is a pilot calciner that consists of a continuously rotating tube partially enclosed with a gas-fired furnace shell. The tube, constructed of Castalloy H-H, has a 16.5-cm (6 and 1/2-inch) internal diameter and is 4.3 m (14 feet) long; the heated section is 2.3 m (6 feet, 8 inches) long. Small flights are placed at intervals within this tube to provide soil agitation. A stationary thermowell extends from the discharge end into the tube with six thermocouples to monitor the soil temperature and three to monitor the gas temperature along the tube length. The tube can be rotated at speeds from 1.5 to 16 rpm and also can be inclined to adjust the flow rate of solids. Typically, a 2- to 3-degree slope is used.

The soil bed volume in the desorber depends on the material flow characteristics, the desorber inclination and rotational speed, and, particularly on the diameter of the "dam" or retaining ring at the discharge end. The soil residence time is a function of the soil bed volume and the soil feed rate. The soil feed rate is controlled by the rotational speed of a feed screw conveyor. Retention time tests were conducted to determine the slope and rotational speed of the desorber tube required to promote flow of the soil through the tube at the feed

rate. Solids discharging from the separator while operating at steady-state are weighed on a digital electronic scale to enable determination of the soil feed rate.

The furnace is a refractory-lined chamber with 14 equally spaced burners controlled by a standard burner control system with appropriate safety features. Propane was used as the fuel. Burner firing rate is manually controlled to give the desired soil temperature profile. The desorber is rated at 337,000 kilojoules per hour (kJ/hr) (320,000 British thermal units per hour [Btu/hr]) maximum heat duty. The maximum heat that can be transferred to material in the desorber tube is estimated to be 105,000 kJ/hr (100,000 Btu/hr). This value depends primarily on the temperature gradient between the soil being processed and the furnace temperature. The discharge end section of the desorber tube is enclosed by the air cooler casing. Cooling of solids within the tube at the discharge end is increased by forced air circulation through the air cooler casing. Soil exits the desorber tube through a gas plenum/transition section and a rotary air-lock valve into a metal receiver can. A sight glass on the discharge plenum of the desorber allows the soil flow to be visually monitored.

Purge gas is introduced at a low flow rate (0.057 cubic meters per minute [m3/min] [2 actual cubic feet per minute, at the soil discharge end to help flush desorbed contaminant vapors and to maintain the proper atmosphere for the treatment process. The off-gas which flows from a plenum at the soil feed end of the desorber containing the purge gas, volatilized contaminants and water, and some particulates, was transferred through a short electrically-heated, insulated duct to the off-gas treatment system.

The off-gas treatment system was comprised of the following components:

A hot cyclone removes a majority of entrained particulates

A quench, spray scrubber is 15.4 cm (6 inches) in diameter and 1.2 m (4 feet) tall. A demister pad is located at the top of the scrubber to minimize entrainment of water droplets. The scrubber liquid is collected in a tank and recirculated by a pump through a set of bag filters and through a heat exchanger cooled by a packaged refrigeration system. Caustic is added to the scrubber liquid prior to operation to control/maintain the pH to alkaline or neutral.

A wet electrostatic precipitator (WESP) captures very fine particulates and aerosols.

A chilled, noncontact condenser reduces the moisture content of the scrubbed gas and removes condensable organic vapors. The condensate is collected in a receiver.

A mist eliminator (Brinks) removes any remaining entrained droplets.

A HEPA filter removes any residual particles.

A two-stage carbon adsorber removes residual organic vapors and mercury vapors.

An induced-draft fan draws off-gas from the desorber and creates a slight negative pressure throughout the system.

The outlet (vent) gas from the off-gas treatment system is discharged to the atmosphere through a roof vent. The desorber system operates at a slight negative pressure to prevent fugitive emissions. The negative pressure is maintained by using a small blower to pull the gases through the desorber and off-gas treatment system. A dry test meter and rotometer measure the vent gas flowrate and volume during a test period.

Pilot-Scale Thermal Desorption Results

The treatment raised the soil pH from 8.8 to 9.1 while the treated solids contained no moisture. The following data provides the results for PCBs removal.

Table I

Approximately 99.6 percent of the Aroclor-1254 and 99.9 percent of the Aroclor-1260 compounds were removed. Concentrations of 220,000 mg/kg and 390,000 mg/kg for the two respective Aroclors were present in the bag filter solids.

The table below provides mercury results for the thermal treatment.

Table II

The feed soil composite sample had a total concentration of 22,750 mg/kg and a TCLP value of 3.1 mg/kg. The treated soil composite contained 1,175 mg/kg, providing a 95 percent reduction, with a TCLP value of 0.032 mg/kg. Mercury was also recovered in the various process residual streams, with the bag filter solids having the highest concentration at 57,300 mg/kg and the scrubber liquid containing 171 mg/kg.

The following table lists the semi-volatiles found in the residual scrubber liquid solution.

Table III

Finally, the next table provides isotopic uranium and gross alpha/beta results for the feed soil, the treated soil, and the scrubber liquid.

Table IV

SOIL WASHING

Soil Washing is an ex-situ, water-based process that employs chemical and physical extraction and separation processes to remove organic, inorganic, and radioactive contaminants from soil. It is usually employed as a pretreatment process in the reduction of the volume of feedstock for other remediation processes. The contaminated soil is excavated and staged, pretreated to remove oversized material, and washed with water and, possibly, other cleaning agents to separate and segregate the contaminants. The process recovers a clean soil fraction and concentrates the contaminants in another soil portion.

The principal advantage of soil washing lies in its ability to concentrate contaminants in a residual soil as a pretreatment step, facilitating the application of other remediation processes. In reducing the volume of soil that must be treated, soil washing can reduce the overall cost. Soil washing performance is highly sensitive to site conditions. The process is most effective when applied to soils and sediments containing large proportions of sand and gravel and is less ineffective, or more costly when applied to soils having a high silt and clay content.

Bench-Scale Soil Washing Operations

Soil washing operations were performed to remove radionuclides, specifically uranium and thorium, from the thermally treated soil. The soil washing methods tested included gravimetric separation (sieving, flotation, and heavy liquid analysis) and chelant extraction. This phase of the study was separated into four separate tasks; 1) an initial particle size characterization, 2) bench-scale physical treatment/gravimetric separation, 3) bench-scale chelant extraction, and 4) pilot-scale flotation/chelant extraction. The "as received" soil sample was initially separated into 5 different size fractions (plus 2mm, 2mm-1mm, 1mm- 150mm, 150mm- 53mm, and minus 53mm) and each of these fractions were individually analyzed for mercury, uranium, and thorium.

This provided a distribution of the contaminants for each particle size fraction.

Heavy liquid separation was then used to characterize the presence of radionuclides (material with a specific gravity higher than soil) from the soil in two different size fractions; the coarse sand fraction (plus 150 mm) and the fine sand/silt/clay fraction (minus 150 mm). These fractions were determined based on the results of the characterization. Thegravimetric analysis uses a tetrabromoethane/methanol liquid media with a density of 2.8, which is between that of soil and heavy metal fragments. The soil floats on the liquid and is separated from the contaminated sink fraction.

A series of bench-scale chelant extractions were then performed trying carbonate/bi-carbonate, citric acid, and EDTA solutions, at varying concentrations, pH values, temperatures, and reaction times. Peroxide was also added to some of the extractions to try and change the uranium to the hexavalent form, thereby making the uranium more amenable to chelation. Approximately 50-g aliquots of coarse soil were reacted with each extractant in a 10:1 weight to weight (wt:wt) ratio of extracting solution to soil. At the conclusion of the reaction time, each soil slurry was filtered to separate the spent extracting solution from the soil solids. The spent extractant was collected for analysis. The soil solids on the filter (filter cake) were reslurried with 200 mL of deionized water and filtered. This was to reduce the residual extractant concentration and remove additional uranium remaining in the filter cake. The rinsate solution and extracted soil solids were collected separately for analysis.

Bench-Scale Soil Washing Results

The results of the initial particle size characterization for the soil showed that each of the five size fractions were analyzed for mercury, uranium, and thorium. These results clearly show that a majority of the contamination is present in the soil fines (minus #150). This fraction only represents approximately 28.4 percent of the total mass. By removing the soil fines, the average concentration of uranium can be decreased from 18,000 mg/kg to 12,000 mg/kg, a 33 percent reduction. The thermal treatment had previously removed 95 percent of the mercury concentration and lowered the PCBs level to below 100 mg/kg. Gravimetric separation provided a 48 percent reduction in activity for the coarse solids (plus 150 mm) while creating a highly contaminated fraction which was only 6.4 percent of the original mass.

The following data presents a series of bench-scale, chelant extractions performed using sodium carbonate, sodium bicarbonate, EDTA, citric acid, and peroxide. A single sulfuric acid extraction was also tested.

Table V

The 0.2 M EDTA, 4 hrs., pH 8.5 extraction provided the best results for a chelant removing approximately 58 percent of the uranium. The sulfuric acid removed over 90 percent, leaving 737 ppm of uranium in the soil. The 0.8 M carbonate/bi-carbonate extraction removed approximately 42 percent of the uranium. This result was close to the results provided by the EDTA extractions. Residual water treatment is much simpler with carbonate fluids, therefore, the carbonate extractions were preferred for the pilot study. The experiments with the peroxide did not seem to significantly improve the extraction.

Pilot-Scale Soil Washing Operations

From the characterization data and the heavy liquid analysis, a separation based on flotation was determined to be the best approach for pilot-scale treatment. A flotation column was constructed to separate the fine fraction (minus 150 mm) from the coarse fraction (plus 150 mm). A 4-inch diameter by 10 ft PVC, schedule #40 pipe was used. A 3-inch thick porous plug was placed in the bottom of the column with a PVC cap and served as a distribution plate. A 3/4-inch water hose fitting was then placed on the side of the column, 1-inch below the top of the porous plug. A second 3/4-inch hose fitting was placed on the side of the column, 8-feet above the porous plug. The soil was charged through the top of the column and the appropriate flow rate for the water was activated.

The flotation column acted basically as a hindered settling classifier. Hindered settling uses a rising current of water, introduced at the bottom of the vessel to expand the soil into a state of teeter. In this teetered state, the soil grains will classify themselves so the coarse grains will report to the bottom of the column and the finer particles will be dispersed to the top of the column. The flow rate for the column was determined from Stokes settling law formulas. The column flotation provided two process residuals, a fine soil slurry and a coarse soil fraction. The fines were allowed to settle and the water was decanted. The fines were labeled settled fines, and the water, flotation water. The coarse soil fraction was treated further by chelant extraction.

Using the data gathered from the bench-scale chelant extractions, a pilot-scale extraction was performed on 12 lbs. of the coarse soil in a 20 gallon reactor. The 20-gallon reactor was constructed of Carpenter 20, a high chrome, high nickel, stainless steel. The reactor was equipped with a with a 1 Hp, 1800 rpm Pfaudler drive and a custom fabricated agitator. The impeller supplied by Chemineer was designed to ensure proper mixing of the soil slurry. Soil was charged through a 4-inch nozzle on the reactor cover. After the extraction, the soil was allowed to settle, and the aqueous layers were removed using an M-2 diaphragm pump and a 1/4-inch teflon, liquid withdrawal tube.

It was determined that a series of two 0.1 M carbonate/bi-carbonate extractions would be performed for the pilot-scale, chelant extraction. The coarse soil was charged to a 20 gallon reactor with 15 gallons of the 0.1 M solution. The soil was reacted for 2 hours, the extractant was removed, another 0.1 M solution was added, the soil was reacted for another 2 hours, and then the extract solution was removed. The soil was then rinsed and slurried with wash water twice, and the washed solids were collected. This wash water contained a large number of fines. These washed fines were filtered and analyzed separately.

Pilot-Scale Soil Washing Results

At the conclusion of the pilot-scale thermal treatment and soil washing stages, a total of five process streams were present. Their descriptions are listed below:

- Flotation Water : water used to remove the fines from the soil
- Settled Fines : fines decanted from the flotation water
- Extract Solution : the combined carbonate/bi-carbonate solutions
- Washed Solids : coarse, extracted and washed solids
- Washed Fines : fine solids decanted from the extraction wash water

All five of the residual process streams, as well as the chemically extracted, non-washed solids, were analyzed for the following parameters:

Table VI

CONCLUSIONS

The data from the thermal desorption testing showed several unexpected results. Thermal treatment, at temperatures readily achievable by commercial thermal desorption systems, could not meet the 12 mg/kg treatment goal for onsite landfill of the Y-12 storm sewer sediments. The results of the bench-scale testing showed that treatment at 600C resulted in a residual mercury level of 80 to 140 mg/kg. This does meet the LDR criteria of 260 mg/kg. Unfortunately the pilot run did not duplicate this performance. The results of the RTA and the pilot-scale desorber have been very comparable in previous testing on PCBs, PAHs and on soil containing lower levels (100 to 200 mg/kg) of mercury. On this waste, while the PCB results of the RTA and pilot desorber were comparable, the treated sediments from the pilot run contained 1175 mg/kg of total mercury instead of the 100 to 150 mg/kg that were expected. This difference is probably due to inadequate purge gas flow in the pilot desorber. The treated sediments contained 0.032 mg/liter of TCLP mercury which is high compared to the RTA results, even when compared to the 350C test. The 350C RTA test showed less than 0.002 mg/liter of TCLP mercury in a treated sediment that contained over 3500 mg/kg of mercury. This implies that leachable mercury species (elemental mercury or mercury oxide) are left in the treated sediments from the pilot run at a ratio higher than in the RTA run. Since these species are volatile at the treatment temperatures, they are probably left in the soil due to inadequate purge gas flow. In the RTA, which operates on a batch cycle, the purge gas is a little hotter than in the pilot desorber and there is no potential for 'reflux' of mercury from the cold (feed) end of the tube to the treated soil end.

The soil washing and soil characterization results were not unexpected. All soil fractions, fines to coarse, were contaminated to a high degree, although the fines did contain higher levels of uranium than the coarse fractions. While the physical treatment by attrition scrubbing and gravimetric separation did lower uranium concentrations, the initial activity of this waste was too high for these treatments to achieve the treatment goal of 15 pCi/g. The chelant extractions were also not fully effective on this waste. The thermal treatment probably converts the uranium to the less soluble reduced uranite form. The thermal desorption treatment is not incineration and the atmosphere in the desorber tube is oxygen starved or pyrolytic. The size separation or fines removal using the hindered settling or flotation column was effective in reducing the mercury concentration in the coarse fraction to below the LDR criteria. While the thermal desorption process alone was expected to provide adequate mercury removal, the test showed that the combined treatment did produce a treated coarse fraction that met criteria for disposal at a commercial radioactive waste facility. This treated waste met LDR and TCLP criteria for mercury and the TSCA PCB limit. Additional testing is needed to determine if thermal treatment operating conditions can be modified to result in a treated soil that passes LDR mercury criteria without the additional soil washing step.

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REPRESENTATIVE WASTE TYPES, ESTIMATED EMISSIONS, PERMITTING ISSUES, AND OPERATIONAL SAFETY FOR DEMONSTRATION OF DETOXSM WET OXIDATION ON HAZARDOUS AND MIXED WASTES

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ABSTRACT

DETOXSM is a cocatalyzed wet oxidation process being developed as an alternative to combustive processes for the treatment of hazardous and mixed wastes. An effort is now in progress to fabricate and test a modular prototype unit capable of oxidizing 25. kg/hr of organic material. The prototype unit will be demonstrated at Savannah River Site (SRS) and at Weldon Spring Site Remedial Action Project (WSSRAP).

There are many low level and mixed wastes in the US Department of Energy (DOE) complex, as well as the public sector, which are organic in nature, or which contain hazardous organic compounds, and can be treated using DETOXSM as an alternative to combustion and other high temperature processes. Waste types to be considered were obtained from the mixed waste data bases and other sources maintained by the Department of Energy, in consultation with the demonstration sites and the Mixed Waste Focus Area.

Once waste types and surrogates for the demonstration were chosen, estimates were made of the emissions from the prototype unit when processing these materials. National Environmental Policy Act (NEPA) documentation was prepared and submitted. Environmental approval for the demonstration effort at SRS was approached through a waste water treatment unit approval. For waste treatment at WSSRAP (a CERCLA site) the substantive requirements for a RCRA Part B permit must be met. A sampling schedule was prepared for emissions from the prototype unit and the oxidation solution, and reviewed by the primary stakeholders in the demonstration.

Operational safety has been approached through modeling studies, flammability studies, HAZOP of the prototype unit, a Process Hazards Review conducted by SRS, preparation of Operating and Safety Manuals, preparation of Safe Operating Procedures for site review, and specialized training of the personnel who will be operating the unit.

BACKGROUND

DETOXSM is a catalyzed wet oxidation process which uses iron ions and homogeneous cocatalysts in an acidic aqueous solution. (1) Organic components of waste materials are oxidized in a stirred tank reactor at temperatures of 400. - 473. K and corresponding pressures of 100. to 800. kPa (atmospheric pressure to approximately 100. psig). Organic compounds are oxidized to carbon dioxide, water, and other simple products, e.g., HCl for chlorinated compounds. Non-volatile inorganic materials are

either dissolved into the oxidation solution or left as inert and/or produced solids suspended in the oxidation solution.

The DETOXSM process is being developed as an alternative treatment for mixed, TRU, and low level wastes in the US Department of Energy (DOE) complex with funding from DOE EM-50 through the DOE Morgantown Energy Technology Center and from the DOE Mixed Waste Focus Area. Demonstration sites are the Savannah River Site (SRS) and the Weldon Spring Site Remedial Action Project (WSSRAP). A prototype unit capable of oxidizing 25. kg/hr (dry weight, organic material) is being fabricated. The prototype unit will be capable of feeding both liquid and solid wastes, and of removing inert and produced solids from the oxidation solution during operation. The oxidation solution will be reduced to residue at the conclusion of a treatment campaign and stabilized for disposal. Shakedown tests and treatment of hazardous wastes/surrogates will be conducted at SRS, after which the unit will be transported to WSSRAP for treatment of low level mixed wastes.

In order to obtain as much value as possible from the demonstration effort, it was highly desirable to include representatives of many waste types in the DOE complex which could be treated with the technology. It was also necessary to obtain exclusions, approvals, and/or permits to operate the prototype unit in compliance with all applicable environmental regulations. Lastly, regardless of whether the prototype unit is found to operate effectively, it must be operated safely during the demonstration effort. These three areas: application, environmental compliance, and safety, are crucial to the demonstration of any waste treatment technology.

REPRESENTATIVE WASTE TYPES

The choice of waste types or surrogates to be treated in the demonstration was not trivial, since they had to be suitable for treatment by the process, available, suitable for the environmental permitting requirements of the demonstration, and representative of wastes in the DOE complex and elsewhere.

Common wastes suitable for treatment by the DETOXSM process are contaminated aqueous solutions, oils, solvents, combustible solids, organic sludges, soft debris, and excess chemicals. Water solutions, although of large volume in many instances, have less concentration of contaminants than many pure organic wastes, thus it was felt that pure organic compounds would be a more rigorous test of the DETOXSM process. This did not, however, eliminate treatment of aqueous streams in the demonstration.

A variety of matrices and contaminants are listed in the DOE mixed waste data bases (2,3), but representatives of these wastes may or may not be available for technology demonstration purposes at a particular site. At SRS, the selection of actual wastes available for treatment studies is limited. In many instances, waste surrogates will be used at SRS instead of actual wastes. Standard surrogate formulations for DOE wastes have been defined. (4,6) There are some advantages to the use of waste surrogates, namely, they are more well characterized than typical wastes, they can be prepared as a standard mixture which can allow more direct comparison of different processes, process limits can be more accurately defined, and they are not considered hazardous wastes for regulatory purposes. Disadvantages of surrogates are that they must be prepared, they are not actual wastes, and their preparation costs can be high. At WSSRAP, a good selection of wastes are available for treatment, including

solvents, oils, sludges, combustible solids, aqueous solutions, soft debris, and excess chemicals.

Waste or surrogate types also had to be treatable within the permitting arena of the host sites. For example, it would be extremely difficult, if not impossible, to obtain permission to treat listed wastes or materials containing radionuclides during the demonstration at SRS. Surrogates will be used instead.

A final critical consideration is whether the wastes or waste surrogates are considered representative of waste types across the DOE complex. The most value is obtained from the demonstration when more representative materials are treated. Under guidance from METC, the MWFA, and representatives from SRS, WSSRAP, and other DOE sites, the list of wastes and waste surrogates proposed for the demonstration was reviewed and matched against lists of mixed wastes and "hard-to-treat" wastes in the DOE complex. As a result of the review, an aqueous waste stream was added at SRS and the surrogate solids treated at SRS will be spiked with hazardous organic compounds.

Table I lists the waste surrogates and wastes to be treated in the demonstration.

Table I

ESTIMATED EMISSIONS

Delphi has estimated emissions from the prototype unit based on the waste surrogates and wastes to be treated, expected operational parameters for the prototype unit, literature values for vapor pressures, and the results of bench scale continuous oxidation studies.

HCl is the primary hazardous species in gaseous emissions from the prototype. Based on analyses of the product water from the bench scale continuous oxidation studies, there are also estimated to be principally three hazardous volatile organic compounds in the gaseous emissions from the prototype. Worst case calculations of emissions of these species under normal operating conditions gave the values shown in Table II. With an emissions stack height of 4.9 m (16. ft) or higher, none of the emissions values exceeded the regulatory requirements at either of the two demonstration sites. The organic compounds were well below regulatory concern levels. HCl emission was the controlling factor.

Table II

Levels of organic compounds in the product water from the prototype was the other primary environmental concern. Based on analyses of the product water from the bench scale continuous oxidation studies, estimates were made of the concentrations of organic compounds in the product water from each type of waste or waste surrogate to be treated at SRS. The full estimates are too extensive to be presented in this paper, but can be summarized by the volume of product water and the total amount of organic compounds estimated to be in the product water. These values are given in Table III.

Table III

The third type of emission from the DETOXSM unit is inert and produced solids. In the demonstration, some inert solids will be contained in the waste surrogates and wastes, but by far the largest volume of solids will be ferric phosphate produced from the oxidation of tributyl phosphate (TBP). TBP is an excellent simulant for wastes containing a large proportion of inert material, since for every weight of TBP oxidized approximately 60.% by weight ferric phosphate solids will be produced. Thousands of kilograms of inert/produced solids (mostly ferric phosphate)

will be filtered from the DETOXSM solution and rinsed during the demonstration, and this will provide valuable operational data on the unit's filtration system. The inert/produced solids are anticipated to be non-hazardous at SRS and low level radioactive at WSSRAP. At WSSRAP, the inert/produced solids will be treated by chemical solidification/stabilization before placement in the on-site disposal cell.

The other solid residue produced from the DETOXSM process is the DETOXSM solution residue. It is anticipated that the solution will be converted to solid form three times during the demonstration effort (once at SRS and twice at WSSRAP), each time resulting in 200. - 250. kg of ferric oxide solid containing the soluble inorganic compounds (including most of the toxic and radioactive metals) from the wastes which have been treated. An estimated 90,000. kg of wastes and waste surrogates will have been treated during the demonstration. The DETOXSM solution solid residue will be disposed of as hazardous waste at SRS, and treated by chemical solidification/stabilization for on-site disposal at WSSRAP.

PERMITTING ISSUES

Exclusions, approvals, and permits to operate the prototype unit in the demonstration are vital. Environmental regulations are complex and restricting in many instances for demonstration of new technologies. An example of this is the definition of the entire SRS as a "facility" for the purposes of the RCRA Treatability Studies Exemption. This ruling restricts the entire 30,000 acre site to a maximum of 250. kg/day waste treatment under TSE. One must be thorough in defining the regulatory situation at contemplated demonstration sites. As a project at a government facility, the demonstration must also be compliant with NEPA, which has required submitting information on the demonstration effort and its potential environmental impacts.

NEPA documentation was submitted to SRS as per the site Environmental Evaluation Checklist. WSSRAP determined that the demonstration was covered by existing NEPA documentation at their site. At each site the demonstration effort falls under a NEPA categorical exclusion.

At SRS, the regulatory path chosen was through the pre-treatment scenarios of the Clean Water Act, with approval of the prototype unit as an industrial wastewater treatment unit. It also would have been possible, although much more complex, to obtain a RCRA RD&D permit for the unit. Approval required submission of detailed water emissions estimates as described above, as well as a description of the process, the prototype unit, the demonstration site, spill containment measures, and the wastes and surrogates to be treated. Estimated air emissions from the prototype unit are well below State of South Carolina limits for toxic air emissions. Solid hazardous residues from the process will be disposed of in accordance with the requirements of the South Carolina hazardous waste management regulations.

At WSSRAP, a CERCLA site, the substantive requirements of a RCRA Part B permit must be met, although a formal permit is not required. Delphi has obtained a determination from EPA that the DETOXSM unit is permittable under RCRA as a miscellaneous thermal treatment system (Subpart X requirements). Estimated air emissions have been submitted to the State of Missouri Department of Natural Resources (MDNR) and air dispersion modeling of the emissions from the demonstration unit was conducted by Argonne National Laboratory for WSSRAP. A preliminary determination has been obtained from MDNR that no air emissions permit will be necessary

for the unit if it operates within estimated emissions parameters. Product water from the prototype unit will be pumped to WSSRAP's wastewater treatment facility, whose permit covers chemical treatment effluents. Solid residues from the process will be treated by chemical solidification/stabilization at WSSRAP's on-site facility and placed in an on-site disposal cell as per the WSSRAP Record of Decision (ROD).

OPERATIONAL SAFETY

Safe operation of the prototype is vital to the success of the demonstration effort, and various safety evaluation procedures are mandated by DOE.

The prototype unit design process has included a formal HAZOP with participation of representatives from the two demonstration sites. The HAZOP process first identifies the causes and consequences of non-standard operational conditions (high or low temperatures, pressures, flows, etc.), identifies existing design features which will alleviate the consequences, and then makes recommendations as to additional design features and operational procedures needed to further minimize consequences.

In addition to the HAZOP, SRS is conducting a Process Hazards Review (PHR) for the demonstration. The PHR is similar to a Safety Analysis Report (SAR), in that it will review all aspects of the demonstration effort from delivery and installation of the prototype unit, through shakedown and treatment operations, to final preparation, loading, and removal of the prototype unit from the site. The nature of the demonstration and size of the prototype unit do not warrant performance of a full SA.

Materials of construction have been an important component of prototype safety. It has been found, through materials compatibility testing and through operational testing in the bench scale continuous oxidation reactor, that tantalum has excellent compatibility with the DETOXSM process solution and its vapors. A tantalum-lined reaction vessel was operated for over 400. hours in the bench scale apparatus with no visible corrosion.(7) Materials tests performed at the Colorado School of Mines with the DETOXSM solution have shown very low corrosion rates for tantalum.(8) It is also known that titanium is somewhat compatible with the DETOXSM solution, but will not survive long exposures. Thus, parts of the prototype unit exposed to hot liquid DETOXSM solution are being constructed of tantalum-lined titanium. This provides, in effect, double containment for the process solution. Sensors between the tantalum lining and titanium shell will detect any breach of the tantalum lining. If a breach occurs, the titanium shell will contain the DETOXSM solution while it is cooled and removed so that the breach can be repaired.

The combination of organic compounds and oxygen in the DETOXSM reaction vessel results in flammability/explosivity safety concerns. A flammability limits study has been conducted by Delphi and Sandia National Laboratories to determine the flammable limits of organic compounds and oxygen under the conditions of the DETOXSM reaction vessel. An example of experimental runs is given in Fig. 1. The "nose" of the graph is the defined flammability limit. Numerous experimental runs have determined that, for volatile organic compounds, no flammability is possible below approximately 5.% by volume oxygen concentration in the headspace of the reaction vessel. This has defined a strict operating limit for oxygen concentration in the primary reaction vessel during operation of the unit.

Fig. 1

Dynamic simulations of the prototype unit have been conducted by the Chemical and Nuclear Engineering Department of the University of New Mexico. The simulations have identified the critical control aspects of the prototype unit. Key to safe operation of the process is control of the oxygen feed rate. Recommendations were made for various control interlocks, for positioning a fast-response oxygen sensor at the gas outlet from the primary reaction vessel with immediate oxygen feed shut off on a sensor reading of 1.% or higher, and for nitrogen blanketing the primary reaction vessel at start up. Also found to be a potential problem scenario was sudden loss of the cooling system. Loss of cooling would result in heat rise in the primary reaction vessel, reaching the design temperature of the vessel in about 13. minutes. From results of the bench scale continuous oxidation studies, shutting off the agitator will decrease the reaction rate of organic material in the reaction vessel by approximately an order of magnitude, which will slow resulting heat rise by approximately an order of magnitude and give approximately 2. hours before the reaction vessel's design temperature is reached. Stopping oxygen flow will also limit heat rise, since there is only about 1. hour's worth of ferric iron available to oxidize the organic in the vessel if it is not being regenerated. Thus, a combination of stopping the oxygen supply and the agitator should result in safe operation even in the event of total cooling system loss. As a emergency backup, the prototype has been designed with a quench tank of water which can be emptied into the primary reaction vessel to quench the oxidation reaction.

The Operating Manual for the prototype unit will consist of a description of the DETOXSM process and the prototype unit including mass balances for typical feeds, normal operating inventories, heat loads, and utilities, a description of the control system and control philosophy including alarm and automatic shut down conditions, safety features of the unit including personal protective equipment, routine precautions during unit operation, and location of safety equipment, the operating procedures including start up, steady state operation, upset conditions, and shut down, emergency procedures cross-referenced to alarm conditions with a "quick reference" chart, routine maintenance and calibration procedures, specifications of all parts, and an index to the manual.

An Environmental Safety and Health (ES&H) Manual is essential for any operations at the demonstration sites. The existing site ES&H document will be followed during the demonstration, which will provide compliance with DOE and OSHA rules and regulations. A Spill Prevention, Control, and Countermeasures (SPCC) procedure is required as per the requirements of 40CFR 112. Guidelines for preparing the SPCC are given in 40CFR 112.7. Some modification of the existing site Emergency Response Plan may be necessary to cover operation of the prototype unit. Existing radiation safety procedures will be followed at the WSSRAP. At WSSRAP, Safe Work Plans are required for on-site operations. The Safe Work Plan provides a description of the work to be done, the crew, equipment, a safety, health, and environmental risk assessment, preventive measures, any subtier contractors, a permit checklist, designation of Delphi's site safety representative, and the work crew signatures and date.

Training is an vital component for insuring safe operation of the prototype unit. In addition to basic training in safety awareness and proper use of personal protective equipment, all on-site operators will

have an accredited 40. hour HAZWOPR (emergency response) training course, and will be trained and certified in operation of the prototype unit as per OSHA Process Safety Management requirements. The process safety training will include typical operating procedures including normal start up and shut down, and emergency procedures including emergency shut down. The sites will provide operators with other necessary site-specific training. At SRS this will include RCRA and OSHA awareness, benzene training, mercury training, and hazardous energy safety training (including lock-out, tag-out procedures). At WSSRAP, radiation safety training will be added to the training regimen.

SUMMARY

Demonstration of the DETOXSM process for treatment of mixed wastes will be conducted with a modular prototype unit having an estimated throughput of 25. kg/hr (dry weight, organic material) of waste. The demonstration will take place at Savannah River Site and at the Weldon Spring Site Remedial Action Project. Wastes and waste surrogates for the demonstration have been chosen based on suitability, availability, regulatory requirements, and representativeness. Emissions and secondary wastes from the prototype unit have been estimated based on the wastes and waste surrogates to be treated. Environmental exclusions, approvals, and/or permits are being obtained. Safe operation of the prototype during the demonstration is being aided by safety reviews, ES&H procedures, emergency procedures, safe operating procedures, and operator training.

ACKNOWLEDGMENTS

Delphi Research, Inc. wishes to express our appreciation to DOE EM-50 (Office of Technology Development), DOE Morgantown Energy Technology Center, and the Rocky Flats Environmental Technology Site for funding the development of DETOXSM. We would also like to express our appreciation to Savannah River Site and the Weldon Spring Remedial Action Project for their help and support as demonstration sites for the DETOXSM technology.

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AN INNOVATIVE PROCESS FOR CONTAINING TOXIC EMISSIONS WHILE THERMALLY TREATING MIXED WASTE (patent pending)

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ABSTRACT

The proposed innovative process for containing toxic emissions while thermally treating mixed waste is also designed to minimize waste handling for a board spectrum of solid and liquid feedstocks being processed separately or commingled in various shapes and sizes. Any potential toxic emissions present in the feedstocks or generated during treatment will be contained or recycled for further treatment. Two types of products will be produced: 1) industrial-grade commodities, and 2) solid products suitable for proper packaging for transportation and disposition.

The system is comprised of a series of technologies currently used in industry for throughputs which exceed current U.S. Department of Energy site-specific processing requirements. The use of a rotary kiln operating in series with a high temperature vitrification unit eliminates the inherent weaknesses of each unit operation. Consequently, a wider array of feedstocks can be processed more effectively with less energy being consumed.

Three additional unit operations have been added to offset the recognized limitations of existing off-gas cleaning systems: condensation, absorption, and membrane separation. Economic and operating considerations provide the incentives for utilizing enriched oxygen rather than combustion air. Cooled off-gases are used to replace the cooling effects of the nitrogen in the combustion air. The utilization of enriched oxygen coupled with off-gas recycling reduces the off-gases subjected to further treatment by 85%.

REGULATORY BACKGROUND

The environmental statutes and policies which most directly influence the selection of the subject technologies emanate from three environmental acts passed by Congress in 1990. In sum these three actsthe Clean Air Act Amendments of 1990 (CAAA), the Federal Facilities Compliance Agreement (FFCA) [PL102-39(b)], and the Pollution Prevention Act of 1990 (PPA) (42 U.S.C. 13101)redirected a portion of the U.S. Environmental Protection Agency's (EPA's) regulatory activities, expanded the scope of the existing environmental statutes and policies, and required federal facilities to conform to all applicable environmental statutes and policies.

Prior to 1990, the National Emission Standards for Hazardous Air Pollutants (NESHAPs) were used to establish air emission standards. This scheme became inoperable during the late 1980s when emissions standards were being established for compounds for which no risk assessment data was available and which could not be identified or qualified in the effluents from hazardous waste incinerators. Consequently, analytic results were inconclusive, and it was not possible to establish emission standards which could be enforced. As a result, Congress amended the Clean Air Act of 1970 (CAA) by abandoning this regulatory scheme. The new regulatory scheme installed was based upon "technology alternatives." Congress directed EPA to establish maximum achievable control technology

(MACT) standards on an industrial sector basis schedule to be completed by 1998.

It is clear that the CAAA has no direct bearing on federal facilities. Historically, most federal facilities had sovereign immunity whereby federal agencies were not required to follow either the federal or state laws due to the potential conflict between mission and compliance. The concept of sovereign immunity was sufficiently broad to prohibit one federal agency from forcing another federal agency to comply with federal statutes and policies. During the 1980s, Congress continued to empower state and local environmental agencies to act as the primary control mechanisms for federal environmental programs which could not be enforced at federal facilities. In a specific attempt to curtail sovereign immunity, Congress passed the FFCA which requires federal facilities to comply with all federal, state, and local solid and hazardous waste management requirements in the same manner and to the same extent as any agency subject to the Resources Conservation and Recovery Act of 1976 (RCRA). Also in 1990, Congress passed the PPA which, among other things, modified and cross-referenced the CAA, Clean Water Act of 1972 (CWA), and RCRA to develop greater continuity among environmental statutes. Section 6501 (42 U.S.C. 13101)(b) states:

The Congress hereby declares it to be the national policy of the United States that pollution should be prevented or reduced at the source whenever feasible; pollution that cannot be prevented should be recycled in an environmentally safe manner, whenever feasible; pollution that cannot be prevented or recycled should be treated in an environmentally safe manner whenever feasible; and disposal or other release into the environment should be employed only as a last resort and should be conducted in an environmentally safe manner.

This policy statement is consistent with the MACT objectives defined in the CAAA and defines the parameters for the proposed processing specifications. The only technical data used in this assessment were the data acquired from EPA-sanctioned studies conducted on hazardous wastes thermal treatment configurations. The detailed results of this broad assessment of hazardous waste incinerators have been presented elsewhere (1,2).

TECHNICAL BACKGROUND

Thermal treatment hazardous waste process specifications are determined by waste characterization and applicable regulations. During the last several decades, waste characterization requisites have been relatively constant while specifications for emissions have increased thirtyfold. Most recently, Congress has directed EPA to define MACT standards for some 20 waste characterization categories [Standard Industrial Classification (SIC) sectors]. Owners/operators are responsible for implementing MACT even if EPA does not comply with the congressional mandate in a timely manner.

The theoretically achievable emission limits for MACT standards approach zero for all three general emission regulatory categories: gross particulate matter, dioxin-like particulate matter, and ash leachability. Rotary kiln configurations were designated by EPA as the best demonstrated available technology (BDAT) for the treatment of hazardous waste during the early 1980s. Many of the configurations will not be able to qualify as MACT due to the inability to satisfy proposed standards for the dioxin-like particulate matter and ash leachability categories. BDAT hazardous waste treatment configurations will not qualify for mixed waste

MACT in most situations primarily due to the anticipated ash leachability criteria. There are interlocking technical relationships between ash leachability and gross and dioxin-like particulate matter; therefore, caution should be used in technically addressing these regulatory classifications independently. Much of the innovative technology recommended is complementary to one another as well as to an existing BDAT hazardous waste processing configuration.

Recently, EPA released a draft version of the Combustion Emissions Technical Resource Document (CETRED), which contains the initial technical analysis concerning emissions of dioxin/furans and particulate matter from certain types of devices which burn hazardous wastes (3). CETRED represents the first preliminary step in the development of technical standards governing emissions from hazardous waste combustors. Herein, the recommended approach utilized is to identify the best 12% (or best five sources, whichever is greater) and then determine the technology utilized to achieve the low emissions. These data are relevant since they potentially represent the lower bounds for the new MACT standards. CETRED displays a cumulative distribution of the hazardous waste industry gross particulate matter emissions. The critical values for the top 12% and median selected as representative of MACT are present in Table I.

Table I

These data imply that if the new gross particulate matter standards were set at the maximum emission rate for the top 12%, it would be necessary for the median emitter to reduce its emission rate threefold to tenfold to meet the new standards. Also, these data suggest that the unit operation selected for gross particulate matter removal should be capable of obtaining the maximum emission value for the median emitter before a dioxin-like particulate matter removal system is employed downstream since higher gross particulate matter values will reduce the effectiveness of the dioxin-like particulate matter removal system.

PROCESS OVERVIEW

The development criteria selected for the conceptual design of a MACT process which contains toxic emissions while thermally treating mixed waste are as follows:

1. Reduce the volume of flue gas generated.
2. Reduce the quantity of toxic emissions present in the flue gases generated.
3. Release only clean inert ingredients (industrial-grade carbon dioxide and nitrogen) from the system.
4. Recycle any potentially toxic substances not destroyed in the treatment process along with a small fraction of the uncleaned inert ingredients, including the water produced during treatment.
5. Produce a solid waste form that will not exceed the Toxicity Characteristic Leaching Procedure specifications, will satisfy nuclear waste product consistency tests, and will be in a form suitable for transportation and final disposition.
6. Minimize waste characterization, material handling and energy consumption.
7. Maximize operating flexibility and reliability while processing a nominal 30,000 tons/year through proven technology.

The baseline process configuration was a BDAT hazardous waste processing system consisting of a two-stage combustion system and an air pollution control system (APCS) (i.e., the K-25 TSCA Incinerator). This basic

process configuration was enhanced with a vitrifier to satisfy ash leaching requirements and with an extended toxic emission control system which contains the toxic substances and cleans the inert gases. Figure 1, Extended Toxic Emission Containment System, depicts the integrated process schematic. A BDAT hazardous waste processing configuration is shown on the left side of the schematic. The complementary extended toxic emission control scheme is reflected on the right side of the schematic. The proposed process utilizes oxygen, rather than air, as an oxidant. Oxygen utilization coupled with flue gas recycling reduces the stack emissions by 85%; thus, the extended toxic emission containment system needs to process only 15% of the normal flow rate (See Ref. 2 for details).

EFFECTIVE THERMAL TREATMENT

EPA designated rotary kiln technology as BDAT in the early 1980s after conducting field tests on eight full-scale commercial incinerators to validate the proposed trial burn analytic procedures. By definition, they are pre-RCRA incinerators. These emission results were reported by Trenholm, Gorman and Jungclaus (4). The significant average emission results were 1) destruction and removal efficiency (DRE) = 99.99%; 2) hydrogen chloride (HCl) = 99% removed; and 3) particulate matter = 180 mg/dry standard cubic meter (dscm) [0.08 grains/dry standard cubic foot (dscf)] released. In 1990 Oberacker reported similar average emission results obtained from 14 of the early trial burns (5). There are at least four polychlorinated biphenyl (PCB) incinerators operating in the United States. These incinerators must obtain a DRE greater than 99.9999%. Oxygen-enriched burners provide similar DREs (6). At the time of Oberacker's report, organic DREs were the primary concern, and with DREs greater than 99.99% the overall flue gas risk related evaluations were better than an acceptable health related incidence ratio of 1:106. During the mid-1980s, metal emissions in the flue gas, the bottom ash and the fly ash became a concern. Also, organic emissions in the bottom and fly ash were subjected to a more critical surveillance. One industrywide survey sponsored by EPA produced typical results (7). Some 43 organic priority pollutants were tested for in the solids residues from hazardous waste incinerators. Levels of organic priority pollutants were found in all solid residues, with the total quantity present varying between 10 ppm and 2000 ppm. Similar organic results were found in other studies while conducting metal analysis on the same residues (7,8). In these studies, ten metals were tested for, and some combinations of metals were found in all residues. The total quantities of these metals varied between 365 ppm and 44,763 ppm. When these organic and metal emission levels were integrated into the requisite risk evaluations via ash leachate migrating to groundwater, the overall risk evaluations produced health related incidence ratios between 1:105 and 1:106. To many, these health related incidence ratios represented an unacceptable risk. Although EPA designated stabilization as BDAT for waste metal treatment, it was quickly demonstrated that vitrification produced a more homogeneous, nonleachable product. As a consequences, much development work was performed to demonstrate that the vitrifier should replace the rotary kiln as BDAT for hazardous waste thermal treatment. This mode of operation was not universally successful due to the increased material handling/preparation complexities introduced into the overall process, coupled with a severe reduction in the types of wastes which could be economically processed. The proposed process eliminates these processing

complexities and retains the desirable operating flexibility features associated with rotary kiln operations. This is accomplished by operating the rotary kiln and the vitrifier in tandem as shown in Fig. 2, Traditional vs. Novel Thermal Treatment Configuration.

Figure 2 shows that the modifications to a BDAT rotary kiln process required for tandem operations with a vitrifier are straightforward. Unless large quantities of relatively dry, low British thermal unit (Btu) solids are being used as feedstock, the vitrifier can be located in the space utilized by the ash conveyers in an existing BDAT configuration. The proposed innovative thermal treatment process involves modifying the operating philosophy because the thermal treatment process is being converted from a two-stage to a three-stage process. This transition requires the kiln to be sized and operated differently. Traditionally, in the two-stage thermal treatment process, a kiln has been sized to achieve maximum heat transfer, which in turn provides the maximum treatment volume and throughput at the expense of several parameters currently recognized as severe toxic emission generation and containment constraints. In the innovative three-stage thermal treatment process, the major function of the kiln is to separate the waste into two phases: solids and vapors. Most of the thermal treatment reactions will be conducted in the other two reactors. Further, the phase separation activity must be conducted in such a manner as to minimize particulate carryover because it has been demonstrated that the presence of mineral particulate matter interferes with completing desirable reactions downstream which are associated with the elimination and separation of dioxin-like and radioactive particulate matter. In sum, in the innovative three-stage thermal treatment process, the major purpose of the rotary kiln is twofold: 1) to prepare the feedstock for the other two thermal reactors and 2) to provide adequate operating flexibility to offset the inherent weakness in the existing waste characterization methodology. Both of these objectives can be achieved at lower operating temperatures. With the proposed three-stage thermal treatment configuration, the normal exit gas temperature will be in the 1200F-1400F temperature range.

Another important feature is the manner in which heat is introduced into the rotary kiln. Traditionally, heat was introduced through a burner on the front face of the kiln in a manner to obtain maximum heat transfer. Due to the basic kiln configuration, bed burning and high water content in the feedstock, temperature differentials greater than 800F were routinely observed. These extreme temperature differentials are an indication of uneven heat transfer, which in turn creates unsteady state thermal reactions. During such transient periods, excessive particulate matter is conveyed into the flue gases, and the solid residence time may not be sufficient to volatize all organics present. The proposed process utilizes a staged cyclonic heat transfer device to ensure effective heat transfer, to provide steady state thermal reactions, and to minimize gas conveyance of mineral particulate matter out of the kiln. Figure 3, Staged Cyclonic Heat Transfer Device, shows this apparatus. The staging concept provides operating flexibility to adjust for feedstock variations not detected in the waste characterization. The use of multiple burners allows the use of high turbulent velocities which are necessary to obtain the homogeneous mixtures required to obtain high DREs with minimal bed disruption. Minimal bed disruption produces minimal gas conveyance of mineral particulate matter. The utilization of the staged cyclonic heat transfer device allows most of the combustion reactions to occur near the

top of the kiln. By introducing the oxidant perpendicular to the direction of the natural draft, a slight cyclonic effect is introduced whereby the flue gases, nearly deplete of oxygen, transverse the top of the bed carrying a thin layer of exposed solid particles to the hot refractory wall opposite the solid bed which does not lie at the bottom of the rotating cylinder. This cyclonic action increases solid bed burnout efficiencies, yet it is not sufficiently severe to create significant increases in airborne particulate matter which could be a carrier of substances that interfere with the destruction of dioxin-like compounds.

From the previous discussion, it is possible to observe that combustion criteria are no longer prominent for sizing and operating the rotary kiln. Rather, the sizing and operating requisites emanate more from direct drying and calcination criteria. This mode of operation is preferred because it reduces particulate carryover, minimizes transient gas flows, allows for a less vigorous waste characterization, and provides a larger operating envelope.

When the hot solids are fed directly into the vitrifier, two forms of energy conservation are utilized: 1) the heat content residing in the solids is immediately utilized, and 2) the vapors are not unnecessarily elevated to superheated conditions before being cooled to ambient conditions. Further, since the hot solids are fed by gravity into the vitrifier, material handling equipment is not required, and workers are not exposed to hot intermediate hazardous waste products. All of these activities enhance the primary objective for utilizing the vitrifierto produce a nonleachable stable ash. The prospects for metal refining are not precluded. In all cases, the ensuing product(s) can be produced in a form suitable for transportation and final disposition.

EXTENDED TOXIC EMISSION CONTAINMENT SYSTEM

The primary purpose of the extended toxic emission containment system (ETECS) is to contain dioxin-like particulate emissions. The primary purpose of the BDAT APCS is to remove the gross particulate matter. There are numerous types of APCSs in service. Table I, Gross Particulate Matter, shows the spectrum of their on-stream performance. In part, APCS performance is a function of the particular configuration in service and the manner in which it is operated. Waste composition is a common component impacting the performance of all APCSs. The two most critical parameters are the amount of particulate to be removed (grain loading) and the particle size distribution. As indicated in the thermal treatment discussion, the manner in which the thermal treatment reactors are operated impacts both the amount of particulate matter conveyed in the off-gases and the corresponding particle size distribution as well. A more complete discussion of these relationships can be found in Ref. 9.

The most sensitive design and operating parameter is the amount of particulate matter present in the feedstock, which is less than 1 micron. All of the traditional APCS configurations begin to experience a loss in collection efficiency in this particle size range. Further, the collection efficiency drops dramatically when the particulate matter is less than 0.5 micron. most of the dioxin-like particles are less than 0.01 micron; therefore, different unit operations must be used to contain these particles. From a design viewpoint, there will be particles present in the ETECS influent in the range between 0.0001 and 1 micron which must be separated for reprocessing. It is imperative that an APCS is operating upstream of the ETECS to remove most of the particulate matter greater

than 0.5 micron. The primary reason for distinguishing between gross and dioxin-like particulate is to draw attention to the following facts: 1) different processing principles must be applied to separate and contain the particles in these two distinct particle size ranges, and 2) the larger particles must be removed before an attempt is made to remove the smaller particles.

Fig. 1

Fig. 2

Fig. 3

The right side of Fig. 1 shows the ETECS. Because oxygen and flue gas recycle are incorporated into the overall design, the gas flow to the ETECS is only 15% of the gas flow through the thermal treatment reactors and the APCS. The nominal composition of the gas flow to the ETECS of water vapor (36%), carbon dioxide (40%), nitrogen (16%), oxygen (8%), and sundry trace substances (less than 0.1%). The processing objective of the ETECS is to clean and release substantial quantities of industrial-grade nitrogen and carbon dioxide. All other substances will be recycled to the most appropriate location in the process. The ETECS is comprised of three different unit operations operating in series: condensation, absorption, and membrane separation. All of these unit operations are used throughout industry at greater throughputs and in more corrosive environments; however, they have not been used in the proposed sequence.

Condensation is utilized as the first stage in the ETECS to remove the water vapor, which represents over one-third of the influent to the ETECS. Equally important, a significant fraction of the particulate matter will be removed when the water is condensed since a portion of this particulate matter will be physically or chemically bound to this water. Another fraction of the particulate matter will become encapsulated during the condensation process. In addition to flow reduction, condensation is very effective in removing particle sizes in the 0.01-1 micron range. Periodic excessive quantities of particles in this size range could reduce the long-term effectiveness of the absorption and membrane separation processes. Such transient conditions have minimal impact on the condensation process.

Absorption is the second unit operation in the ETECS. Its primary use is to clean and remove the carbon dioxide, which represents nearly two-thirds of the remaining off-gases. An organic solvent is used to absorb the carbon dioxide. The organic solvent will also absorb and adsorb trace amounts of particulate matter which is removed during the regeneration step. This portion of the process is not reflected on the process schematic since only small periodic flows are involved. After the clean carbon dioxide is separated from the solvent, it is sampled and analyzed prior to release. Should the product not meet the required specifications, it will be reprocessed.

It should be noted that membrane separation technology can be used to clean carbon dioxide as well. Absorption was selected as the second cleaning stage after condensation because absorption is more effective in the presence of oversized particles, where particles greater than 0.005 micron are considered to be "oversized." Also, trace amounts of acid gases may be present periodically. Although some of these constituents have a particle size less than 0.005 micron, they could have a long-term deleterious effect on certain types of membranes. Over the long term, these constituents can be processed more effectively in an absorption system.

The third unit operation in the ETECS is membrane separation, which has several applications all with small throughputs of relatively clean material. The primary application is to separate the clean nitrogen from the absorber off-gases, which are then recycled. The absorber off-gases represent about 14% of the influent to the ETECS. About one-third of these off-gases will be recycled after the nitrogen is separated from these off-gases. The clean nitrogen will be sampled and analyzed prior to release. All nitrogen products not meeting specification will be recycled.

Membrane separation is a complementary technology which is sized according to particle inlet concentration, particle size distribution, and desired particle outlet concentration. Table II, Membrane Material vs Particle Size Distribution Range, shows the nominal particle size range for three common categories of membrane materials.

Table II

There is an overlap reflected in these particle size ranges because separation efficiency is not constant over the entire range.

Membrane separation units consist of a cascade of common components arranged to provide the desired effluent(s). The most basic element in a membrane separation unit is the bundle of membrane material wrapped around a tube in multilayers separated by spacers. Several of these tube bundles are placed within a steel tube and capped. The caps contain three openings: feed, permeate (the material passing through the membrane), and retentate (the material contained by the membrane). These cylinders are then arranged in a myriad of configurations to achieve the desired concentrations in the streams to be cleaned. Fig. 4, Typical Membrane Cylinder Configurations, shows examples of the manner in which these cylinders are commonly arranged. There can be multiple cylinders in a staged configuration. Normally, all cylinders in a multiple-cylinder stage contain the same type of membrane material. On occasion, different material may be used in different states. Figure 4c shows the configuration used to separate the nitrogen from the other recyclable substances. Multiple recyclable streams are shown since all substances may not go to the same treatment reactor.

Fig. 4

SUMMARY

The successful containment of toxic emissions while thermally treating mixed waste emanates from the utilization of three thermal reactors in series coupled with oxygen utilization and flue gas recycling. This mode of operation reduces the gaseous effluents by 85%. The ensuing reduction in the quantities of flue gases emitted makes it economically affordable to use proven technologies to clean these effluents to environmentally acceptable levels.

The incremental capital cost for making these proposed enhancements to a traditional BDAT configuration is estimated to be less than 20% of the traditional BDAT plant site cost. The major increase in operating cost will be reflected in the power cost required to vitrify the solids and pressurize the residual flue gases for processing in the extended toxic emission containment systems. A reasonable estimate of these operating costs is the cost of 1.25 MW/ton of solids processed.

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RCRA PERMIT MODIFICATIONS AND THE FUNCTIONAL EQUIVALENCY DEMONSTRATION: A CASE STUDY

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ABSTRACT

Hazardous waste operating permits issued under the Resource Conservation and Recovery Act (RCRA) often impose requirements, typically by reference to the original permit application, that specific components and equipment be used. Consequently, changing these items, even for the purpose of routine maintenance, may first require that the owner/operator request a potentially time-consuming and costly permit modification. However, the owner/operator may demonstrate that a modification is not required because the planned changes are "functionally equivalent," as defined by RCRA, to the original specifications embodied by the permit. The Controlled-Air Incinerator at Los Alamos National Laboratory is scheduled for maintenance and improvements that involve replacement of components. The incinerator's carbon adsorption unit/high efficiency particulate air filtration system, in particular, was redesigned to improve reliability and minimize maintenance. A study was performed to determine whether the redesigned unit would qualify as functionally equivalent to the original component. In performing this study, the

following steps were taken: a) the key performance factors were identified; b) performance data describing the existing unit were obtained; c) performance of both the existing and redesigned units was simulated; and d) the performance data were compared to ascertain whether the components could qualify as functionally equivalent.

In this case, the key performance data included gas residence time and distribution of flow over the activated carbon. Because both units were custom designed and fabricated, a simple comparison of manufacturers' specifications was impossible. Therefore, numerical simulation of each unit design was performed using the TEMPEST thermal-hydraulic computer code to model isothermal hydrodynamic performance under steady-state conditions. The results of residence time calculations from the model were coupled with flow proportion and sampled using a Monte Carlo-style simulation to derive distributions that describe the predicted residence times. The results showed that the redesigned unit, although physically different in many aspects, is equivalent in performance to the existing unit, thereby obviating the need to seek a permit modification. By using this approach to demonstrate the functional equivalency of the redesigned unit, it is estimated that $1 million allocated to perform a trial burn was saved.

INTRODUCTION

The Los Alamos National Laboratory's (LANL's) Controlled-Air Incinerator (CAI) has recently undergone maintenance and improvements that involved replacement of components. The CAI carbon adsorption unit, in particular, was redesigned to improve reliability and minimize maintenance requirements. This study compared the hydrodynamic performance of the existing activated carbon adsorption unit/high efficiency particulate air (HEPA) filtration system (the system) and the proposed upgrade of the system. The purpose was to determine whether the changes can be considered "functionally equivalent" pursuant to regulations promulgated under the Resource Conservation and Recovery Act (RCRA) at 40 CFR 270.2 (1).

The CAI was originally intended to process small amounts of waste for research purposes only, so the specifications for the materials of construction did not consider the rigors and stresses of continuous operation, particularly those relating to corrosion. The existing carbon adsorption unit is constructed of low-carbon steel, a corrosion-prone material. Also, the angle of the carbon bed is thought to inhibit efficient filling of the bed with carbon and allows carbon particles to escape through the screen, which may encourage bridging of the media.

In order to maintain the current operational parameters and minimize the potential for a major RCRA permit modification, as discussed below, the proposed unit design is based on the same primary design criterion as the existing unit, namely, a gas-carbon contact time of 0.5 second. This criterion is based on Nuclear Power Plant Air Cleaning Units/Components, Section 5.2.2, "Adsorber Design," which requires that for the removal of gaseous iodine the minimum residence time of a gas stream in an adsorbent be 0.25 second (2). The current unit is designed to a minimum residence time of 0.5 second because a safety factor of 2.0 was used in the original design.

REGULATORY CONSIDERATIONS

To burn hazardous and mixed radioactive waste, the CAI must operate under the provisions of LANL's Hazardous Waste Facility Permit issued by the New Mexico Environment Department (NMED) (3). Under RCRA, any proposed

changes to a permitted facility must be evaluated to determine whether the changes require a modification to the permit. The requirement for a permit modification and its magnitude, or "Class," are determined using specific criteria. The general criteria that define the modification class are specified in 40 CFR 270.42(d)(2) as follows:

i) Class 1 modifications apply to minor changes that keep the permit current with routine changes to the facility or its operation. These changes do not substantially alter the permit conditions or reduce the capacity of the facility to protect human health and the environment. In the case of Class 1 modifications, the Director may require prior approval.

ii) Class 2 modifications apply to changes that are necessary to enable a permittee to respond, in a timely manner, to,

A) Common variations in the types and quantities of the wastes managed under the facility permit,

B) Technological advancements, and

C) Changes necessary to comply with new regulations, where these changes can be implemented without substantially changing the design specifications or management practices in the permit.

iii) Class 3 modifications substantially alter the facility or its operation (1).

Each class requires a different level of response from the permittee and different levels of public involvement, Class 3 modifications being the most rigorous.The carbon adsorption unit is not specifically described in LANL's permit as it is currently written (3). Therefore, a physical change to this particular apparatus does not automatically require a modification to the permit unless the NMED determines that it causes or justifies a change to the permit conditions(40CFR 270.41(a)(1)) (1). The only permit condition that applies to the carbon adsorption unit is a requirement to replace the spent carbon; this requirement will still be met after the proposed unit is installed.

The primary concern of CAI staff was the potential for the upgrade to require a Class 3 permit modification, as specified in Appendix I to 270.42, item L(3).

Modification of an incinerator, boiler, or industrial furnace unit by changing the internal size or geometry of the primary or secondary combustion units, by adding a primary or secondary combustion unit, by substantially changing the design of any component used to remove HCl/Cl2, metals, or particulate from the combustion gases, or by changing other features of the incinerator, boiler, or industrial furnace that could affect its capability to meet the regulatory performance standards.... [This is a Class 3 modification]. (1) (emphasis added)

The carbon adsorption unit is designed to remove radioactive isotopes of iodine, which is neither a particulate nor a metal. Hydrogen chloride (HCl) and chlorine (Cl2) are removed in the venturi scrubber and the absorber column, respectively. The unit may contribute to the removal of organic compounds, but its capacity in this regard has never been assessed, except through the overall destruction and removal efficiency (DRE) established for the entire CAI during the trial burn. That is the root of the problem for the CAI, because the last part of the citation above describes any change that could affect the ability to meet performance standards.

However, if the facility can demonstrate that the upgraded equipment is functionally equivalent to the equipment that it replaces, there is no

longer a basis for assuming that the change would affect performance. Functionally equivalent components are defined in the regulations as follows:

> Functionally equivalent component means a component that performs the same function or measure and which meets or exceeds the performance specification of another component (40 CFR 270.2).

In fact, upgrading with functionally equivalent components is a specific type of Class 1 modification listed in Appendix I of Part 270:

> Equipment replacement or upgrading with functionally equivalent components [is a Class 1 modification] (1).

To better define the circumstances that constitute this type of modification, the EPA stated in the preamble to the revised 40 CFR Part 270 that:

> Under Item A(3), permittees are able to make routine equipment replacements that are necessary for the continued operation of the facility ... (however) some permit conditions may inadvertently create restrictions by incorporating portions of the Part B permit application by reference. For example, if a permit incorporates a design drawing by reference which specifies a particular piece of equipment ... then to replace the item with anything other than the original model might require a permit modification. Such an item may not be available ... or the permittee may prefer to replace it with an improved version ... Therefore, if it is necessary to include design drawings in permits, the permit condition should also allow minor deviations from the design without a permit modification (although the Director may want to have the permittee send the revised design to the Agency to maintain a current file on the facility) (53FR 37924-37925, September 28, 1988) (4).
>
> Therefore, Item A(3) in the Appendix provides that equipment replacement or upgrading with functionally equivalent components is a Class 1 change (53 FR 37924, September 28, 1988) (4). (emphasis added)

It is clear, then, that if the proposed change to the design of the carbon adsorption unit could be demonstrated to be a replacement with a functionally equivalent component, it would not require a modification to the existing permit and at worst would have to follow the procedures for completing a Class 1 modification.

For the CAI, the difficulty arises from the unique nature of each unit's design. Under normal circumstances, manufacturer's performance data or other specifications could be used to demonstrate functional equivalency. However, because each unit was custom designed for application in the CAI, this was not possible. Instead, the existing and proposed unit designs were modeled using numerical simulation to perform a relative comparison of performance.

NUMERICAL SIMULATION

The time that moving gas resides within a process unit, or any delineated portion thereof, can be referred to as the gas residence time. For the system being analyzed, treatment of the offgas occurs when it is in contact with the adsorption medium, which is granular activated carbon. This leads to the conclusion that gas residence time within the carbon bed is the primary criterion for comparison between competing designs. Also of importance is the relative distribution of gas flow or flow proportion over and across the carbon bed. For the purposes of this analysis, flow proportion was defined as the fraction of the total gas flow moving through the unit at any given time, and was calculated by dividing the flow in a unit cell by the total flow. Values of gas

residence time and flow proportion cannot be calculated directly. Rather, numerical simulations were performed that provided gas velocities and pressure drops throughout the internal volumes of both unit designs. The gas velocities and their distributions resulting from these simulations were then used to calculate gas residence time and flow proportion. These parameters were compared using a Monte Carlo-style sampling technique to determine the relative steady-state hydrodynamic performance of the two designs, which enabled the determination of functional equivalency.

The hydrodynamic behavior of the two unit designs was modeled using several numerical simulations developed with the TEMPEST thermal hydraulic computer code (5). The models simulated the physical configuration and operation of the two designs under actual operating conditions. Physical configuration data were taken from drawings of the two designs (6,7,8,9). Typical operating conditions and additional information on the operation of the CAI were taken from a report produced by T.K. Thompson, Inc. (10)and various data transmittals from CAI personnel. Operational data were also available from the CAI RCRA trial burn, conducted in 1989, which described operating conditions in the existing unit (10,11).

Several assumptions were necessary to complete the calculation of the gas residence times and flow proportions. The primary assumptions are listed below.

Gas residence time is equivalent to treatment capacity.

Gas residence time distributions correspond to the numerical simulation predictions of flow.

The effects of nonuniform flow distribution are accounted for by using flow proportion as the measure of probability that the gas will achieve a particular residence time.

Minimum residence time is equal to the minimum gas flow path length divided by the gas velocity component orthogonal to the carbon bed screen.

Minimum gas flow path length is defined by the shortest path across the carbon bed (i.e., the carbon bed thickness).

Carbon bed cells in the region of the mid-axial barrier in the proposed cylindrical design can be ignored because of the lack of significant radial flow and the subsequent overestimation of gas residence time.

The results of the numerical simulation of the hydrodynamic performance of the existing and proposed unit designs, in the form of flow and relative pressure distributions, are shown in Figs. 1 and 2.

Fig. 1

Fig. 2

FUNCTIONAL EQUIVALENCY DETERMINATION

As mentioned previously, a statistical technique was employed to determine the distribution of residence times for the proposed and existing unit designs. Volumetric flow proportion was used as a measure of probability to characterize the residence time distribution for each unit. Monte Carlo-style simulations (i.e., random sampling of the distributions) were used to statistically describe the distributions. Flow proportion was used in the analysis to describe the frequency of occurrence of residence times in each unit. The proportions of flow at a given residence time were summed across all spatial elements in the carbon bed models and used to describe a probability density function for the distribution of residence times for each unit design. Because of its cylindrical, double-pass design, the approach used to determine the

distribution of total residence time for the proposed unit was more complex than for the existing unit. The distributions of residence times for the top and bottom halves of the unit were sampled separately and then summed. This approach is predicated on the assumption that the residence times in the top and bottom halves of the proposed unit design are independent.

Figures 3 and 4 illustrate the distribution or probability of gas residence times for the existing and proposed unit designs. Analysis of these distributions indicated that the proposed unit has a significantly greater proportion of flow that exceeds the 0.5 second requirement than the existing unit. The distributions also indicated that the volumetric flow of gas was better distributed in the proposed unit, thus offering potentially more efficient usage of carbon and a larger margin of safety for contaminant breakthrough. Based on the numerical simulations and subsequent analysis, the proposed design was determined to be functionally equivalent to the existing carbon adsorber unit in the CAI.

Fig. 3

Fig. 4

CONCLUSION

Because of the unique design and application of the carbon adsorber, manufacturer's performance data or other specifications could not be used to demonstrate functional equivalency. This necessitated a numerical simulation of the relative hydrodynamic performance of each design. The results of this simulation coupled with a statistical simulation technique provided the data necessary to determine that the proposed unit design was indeed functionally equivalent to the existing design. The approach used in this case successfully avoided the need for a potentially time-consuming and costly modification of the facility's hazardous waste operating permit. It is estimated that approximately $1 million was saved by avoiding a trial burn to evaluate the proposed unit's performance. This approach, in effect, sets a precedence for application at other permitted units to meet the compliance requirements of RCRA while minimizing costs and potential delays.

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BRINGING RADWASTE MEASUREMENT CAPABILITY TO THE CUSTOMER

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ABSTRACT

The lease of a comprehensive suite of radiometric assay systems is proposed as an attractive option for meeting the monitoring requirements of short-term decommissioning or waste retrieval operations. The merits of leasing are that a cost effective, guaranteed, high quality service is provided using state-of-the-art technology operated by skilled technical personnel. A suite of instruments have been developed as a result of significant operational experience which are capable of meeting the full range of the monitoring needs of decommissioning operations from planning through execution to final removal and characterization of the waste. These instruments have several common design features such as transportability, ease of decontamination etc., which make short-term leasing a viable option.

INTRODUCTION

Meeting the increasingly stringent requirements for characterization and assay of radioactive waste materials (e.g. from decommissioning operations) requires the use of sophisticated and complex radiometric monitoring systems. In many cases the plant operator has a continuing, long-term need for such monitoring systems and this need is best met by the conventional approach of purchase and operation by his own personnel. Where the requirement for the monitoring system is short-term, however, the operator may see little value in ownership and still less in providing the trained manpower.

BNFL Instruments Ltd. (BIL) and Pajarito Scientific Corporation (PSC), two wholly owned subsidiaries of the BNFL group of companies, have developed a solution to this problem which involves the lease of a comprehensive suite of highly mobile monitoring systems supported by skilled technical personnel. This approach offers a high quality, low risk service to the customer which will help to plan decommissioning operations and will meet all the relevant regulatory requirements. The

radiometric systems are all based on state-of-the-art technology and are designed to meet all the customers monitoring needs from initial survey of gamma "hotspots" and in-situ measurement of residual fissile material within the plant through to accurate assay of the final packaged waste materials.

OPERATIONAL EXPERIENCE

The operational experience which has led to the ability to provide this service includes the decommissioning in recent years of three redundant plutonium (MOX) facilities at the Sellafield reprocessing plant in Cumbria, UK. The three facilities were a co-precipitation plant, a dry granulation plant and a fast reactor fuel fabrication plant. The MOX fuel from these facilities had plutonium enrichments of up to ~30% PuO_2. In total some 60 individual Pu contaminated gloveboxes and vessels have been removed from these facilities and processed as TRU waste. As part of this extensive programme, decommissioning managers and engineers have worked closely with members of BIL to formulate an integrated dismantling and NDA monitoring strategy. This has resulted in the development of a "family" of radiometric systems aimed at providing cost effective, simple and accurate measurements of TRU waste to meet the decommissioning manager's needs. The overall decommissioning/NDA strategy and the radiometric instruments used to make this strategy possible are shown in Fig. 1.

All of these instrument systems have several features in common which ensure that they will function correctly for prolonged periods in a plant environment and that after completing a programme of monitoring they can easily be relocated and used for subsequent projects. For example, where the entire instrument is not readily transportable, modular construction has been used to allow for easy relocation. All the instruments are plant ruggedised (i.e. designed to withstand a certain level of rough treatment), are sealed wherever possible and have surfaces which allow for easy decontamination. Specific attention has been paid to preventing electrical noise problems e.g. through the use of RFI-screened detector cables and automatic data quality checking. The instruments all incorporate periodic self check features (which typically involve measurement of a standard source) to ensure the correct functioning of the instrument. All the assay type instruments provide "most probable" Pu mass values which can be used for material accountancy purposes and "worst case" values which are used for criticality control and are calculated by adding the total measurement errors (i.e. statistical counting uncertainties and systematic error components) to the "most probable" value.

Fig. 1

PRE-DECOMMISSIONING PLANNING

The overall decommissioning strategy for Pu plants usually involves first removing waste from within the process gloveboxes and vessels, then removing the gloveboxes themselves and finally removing any remaining contaminated areas of floors and walls within the building. This strategy requires good planning prior to the commencement of decommissioning operations. Remote identification and quantification of radiation hot-spots in gloveboxes allows clean-up teams to effectively target these areas for early decontamination, resulting in an overall minimization in dose uptake throughout the decommissioning programme. BIL's easily transportable RADSCAN 600 system has been specifically designed to

achieve this task by combining real-time visual and radiometric surveys of individual gloveboxes or entire process lines.

The RADSCAN 600 system (Fig. 2) consists of a CsI (Tl) scintillation detector and photodiode, with a tungsten collimator (which gives angular resolutions between 2 and 9), which is mounted so as to have the same field of view as a high definition colour CCD camera. The scanning head in which the detector and camera are mounted is sealed to IP65 in order to aid decontamination using a water jet. The system has a remote workstation with a colour monitor and VCR for viewing and recording RADSCAN surveys and a PC which controls the operation of the system including the pan and tilt of the scanning head.

From its deployment tripod the RADSCAN system provides real time 2D mapping of the spatial distribution of radioactive material in vessels and gloveboxes. A 3D map of radiation hotspots within a glovebox can be obtained by applying a triangulation process to the results of scans taken from two different locations.

Fig. 2

The location of residual quantities of plutonium of the order of tens of grams have been identified within redundant gloveboxes using the RADSCAN system and the triangulation technique. Intrinsic photodiode noise means that the RADSCAN system is inappropriate for the detection of radiation below approximately 100keV. Development work is currently underway to improve the instrument's performance in this energy range hence allowing, for example, 60keV radiation from 241Am to be detected.

INITIAL ASSAY OF WASTE REMOVED FROM GLOVEBOXES

Following the location of significant residual plutonium hold-up, the decommissioning teams target these areas for clean-up and remove the residual plutonium via bagging ports. TRU waste removed from gloveboxes in this fashion can then be rapidly assayed for nuclear safety purposes by the TRU-D measurement system (Fig. 3), a mobile instrument which is specially designed to monitor small, packaged waste items. The TRU-D system is a self contained, stand alone instrument system in which all the components (e.g. assay compartment, detectors, cables, electronics and user interface) have been integrated into a single mobile unit (dimensions 1.65m high x 0.71m wide x 1.55m long) which moves under the power of its own internal battery.

The TRU-D system is normally configured to determine the Pu mass via a total neutron counting measurement so that assay times are minimised (5 minutes or less). However this technique may give inaccurate results if the isotopic or chemical composition of the Pu is not known. In such cases the TRU-D system can be configured to perform the Pu assay using the Neutron Coincidence Counting (NCC) technique and to incorporate High Resolution Gamma Spectroscopy (HRGS) using a Hyper Pure Ge detector. The Pu isotopic composition can be determined form the recorded gamma spectrum through the use of BIL developed and tested algorithms or through the use of the MGA code (1).

The combination of these two techniques (NCC & HRGS) results in a Pu assay which is independent of the plutonium isotopic or chemical composition. With the TRU-D system configured to perform a combined NCC and HRGS assay, total measurement errors of below 20% are usually achieved for samples containing a few grams of Pu.

Fig. 3

IN-SITU ASSAY OF GLOVEBOXES AND VESSELS

In-situ assay of the residual plutonium content of gloveboxes and vessels prior to removal and size reduction is necessary to satisfy criticality safety requirements and to provide projectplanning information. Considerable experience has been gained in performing such measurements (more than 60 in-situ glovebox measurements have been taken) and has led to the development of a unique and versatile instrument system called DISPIM (Decommissioning In-Situ Plutonium Inventory Monitoring) specifically for such tasks.

The DISPIM system (Fig. 4) incorporates of a number of individual neutron counting modules which consist of two 3He detectors encased in a block of polyethylene (approximate dimensions 1200mm x 200mm x 110mm). The modules are deployed around the accessible faces of the glovebox on mobile stands which have been designed for maximum stability (in order to satisfy plant safety requirements) and easy dismantling and relocation. The system electronics and data recording facilities are housed in a remote sealed cubicle to prevent contamination of these components.

The configuration of the DISPIM system (i.e. the number of detector modules used and their relative positions) will vary from item to item depending on size and accessibility. Objects ranging in size from 200 litre drums to large process vessels (say 3m x 3m x 3m) can be assayed. Once the system is in position a standardisation check (using a small neutron source) is performed on all the detector modules to ensure their correct functioning before the measurement is conducted.

Fig. 4

The total Pu content of each glovebox is obtained by combining the results of a Neutron Coincidence Count (NCC) from the DISPIM system with Pu isotopic composition data which is either known or measured using a HRGS system and Pu isotopic composition algorithms. This measurement of the plutonium content is independent of the Pu isotopic and chemical composition.

The usual method of calibrating the DISPIM system is based upon "posting" a 252Cf source of known activity into the glovebox once all the detection modules are in position. If entry into the glovebox is not possible then the calibration can be performed either by reproducing the measurement geometry in free space or through the use of radiation transport modelling. Corrections which are required in order to account for the contribution to the observed neutron coincidence count from spontaneouslyfissioning material in neighbouring gloveboxes are also determined at this stage.

Calibrations based on the most likely locations of any residual Pu (obtained from information from the RADSCAN 600 system or from process knowledge or operator experience etc.) are used to determine the "most probable" Pu mass, whereas the "worst case" or criticality control Pu mass value is obtained by assuming the most pessimistic measurement conditions.

Measurement times for the DISPIM system are typically of the order of a few hours and total measurement errors of below 50% (or significantly less if the residual Pu locations are known) are usually achieved for Pu masses of approximately 10 - 20 grams.

SIZE REDUCTION AND DRUM PACKING

Once the individual gloveboxes have been shown to be belowcriticality safety limits, they are usually size reduced and packed into 200 litre drums. An alternative option is simply to load the gloveboxes directly into a box (or crate) for interim storage.

If size reduction is the chosen option, it is performed using a variety of cutting techniques within a specially constructed working area known as an RMC (removable modular containment). To ensure compliance with the nuclear safety requirements of drum filling, a "piece" monitor is used to assay the size reduced waste "pieces" before they are loaded into storage drums. The "piece" monitor (Fig. 5) is designed around an instrument bulge in the RMC and therefore remains outside the high contamination area. Each glovebox "piece" is placed in the instrument bulge by personnel within the RMC for assay of the fissile material. A running total is kept of the fissile mass of all the waste items consigned to each individual waste drum. The "piece" monitor automatically requests a new drum if the addition of a waste piece would raise the total fissile content of the drum currently being filled above the defined nuclear safety limit.

Fig. 5

As with the TRU-D and DISPIM systems, the "piece" monitor determines Pu masses through the combination of results from a NCC system and an HRGS system. If required, the gamma spectrum recorded by the HRGS system can also be used to determine the Pu : U ratio and U enrichment figure. Another optional feature of the system will warn of the presence of PuF4, a factor which can cause a significant increase in measurement errors. Once the presence of PuF4 has been identified, the measured Pu mass and its associated error value can be reassessed.

The "piece" monitor uses a short assay time (typically 5 minutes) in order to maximise the efficiency of the size reduction and drum packing operations. Total measurement errors of less than 15% are usually obtained for Pu masses of the order of a few grams. Several such "piece" monitors have already been supplied to Decommissioning Unit at Sellafield and they have been usedextensively in decommissioning operations.

ACCURATE DRUM ASSAY PRIOR TO INTERIM STORAGE

The total fissile mass values provided by the "piece" monitor satisfy nuclear safety requirements for the movement of drums of TRU waste. This allows the drums to be transported from the building in which they were generated, for a final accurate assay before interim storage. This final drum assay is performed by a mobile system whose purpose is to provide a highly sensitive, versatile, non destructive measurement for the characterisation of radioactive constituents in low level and transuranic waste. The system consists of an Imaging Passive Active Neutron Counter (IPAN) supplemented by high resolution Gamma Energy Analysis (GEA). Active mode neutron analysis uses a pulsed Zetatron source (108 neutrons/second) and the Differential Die Away (DDA) measurement technique to assay fissionable materials in the waste such as 235U and 239Pu. Passive mode neutron analysis determines the content of spontaneously fissioning isotopes such as 240Pu or 244Cm. For both active and passive neutron measurements, 3He detectors arrangedin a 4p configuration are used for neutron detection. Passive and active mode neutron imaging analysis identifies and corrects for inhomogeneities in the spatial location of fissioning constituents. Automatic correction for matrix composition and neutron multiplicity analysis augment assay accuracy.

A single 20% efficient high purity germanium detector is used for the GEA measurement to identify and quantify gamma emitting radionuclides and to determine Pu isotopic composition. This analysis is equipped with peak search and strip algorithms, a library of peak centroids and activities,

statistical precision analysis and corrections for the energy dependence of the detector efficiency.

Each assay is accompanied by advanced systematic error correcting algorithms, gamma-ray and neutron diagnostic packages and automatic data quality checking. IPAN/GEA systems are capable of performing assays in the range between milligrams and hundreds of grams of 235U or total Pu in 55 gallon or 85 gallon (overpack) drums. This range permits discrimination at 10 nCi/g for nearly all low level waste matrix categories and assures safeguards measurement requirements are met for drums with higher fissile contents. IPAN/GEA systems are operationally proven and comply with applicable waste disposal regulations. An easy-to-use operator interface is managed from a Microsoft Windows platform.

The measurement system is housed in a custom built 13' x 10' wide x 40' long reinforced semi trailer (Fig. 6) equipped with shock and vibration isolators. When outfitted with drum loading and conveyor apparatus, the total weight of the trailer is 35,000 lbs. An air conditioned control room houses measurement electronic modules and computers. This room is isolated from the measurement system section to reduce radiological exposure to operators.

Fig. 6

CONCLUSION

As a result of substantial operational experience a wide range of radiometric monitoring equipment is now available to support the decommissioning of redundant Pu facilities. All of the monitoring systems have a proven track record of exceptional performance in an industrial plant environment. The development of these instruments which has occurred as a result this operational experience has led to a capability to offer the use of these systems as a service as well as a product. The instrument systems are all transportable, robust, versatile and designed for easy decontamination. BIL and PSC personnel have gained extensive knowledge and experience in the operation of instrumentation in support of decommissioning operations. These factors combine to make the lease of instrument systems, supported by skilled technical personnel, a guaranteed high quality, low risk service, which will be very attractive for short-term decommissioning/waste retrieval operations.

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CHARACTERIZATION OF MIXED WASTE FOR SHIPMENT TO TSD FACILITIES PROGRAM

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ABSTRACT

In compliance with the Federal Facilities Compliance Agreement, Los Alamos National Laboratory (LANL) is striving to ship its low-level mixed waste (LLMW) off site for treatment and disposal. In order to ship LLMW

off site to a commercial facility, LANL must request exemption from the Department of Energy Order 5820.2A, Radioactive Waste Management, requirement that LLMW be shipped only to Department of Energy facilities. Because the process of obtaining the required information and approvals for a mixed waste shipment campaign can be very expensive, time consuming, and frustrating, a well-planned program is necessary to ensure that the elements for the exemption request package are completed successfully the first time. LANL is developing a program that is cost-effective, quality-driven, and compliance-based. This program, partially described in LANL's Low-Level Mixed Waste Off-Site Treatment and Disposal Management Plan (currently in draft form), encompasses selecting a qualified analytical laboratory, developing a quality project-specific sampling plan, properly sampling liquid and solid wastes, validating analytical data, documenting the waste characterization and decision processes, and maintaining quality records. The products of the program are containers of waste that meet the off-site facility's waste acceptance criteria, a quality exemption request package, documentation supporting waste characterization, and overall quality assurance for the process. Documentation of the decision process provides substantiation for containers of waste remaining on site, thus demonstrating a good faith effort to comply with the Federal Facilities Compliance Agreement and state agreements.

The program consists of elements that are common to all shipment campaigns, can be customized from templates, and are unique for each individual shipment. The majority of the effort is accomplished with the first off-site shipment. This program provides guidance for achieving a successful shipment campaign, is flexible to allow for customization to each waste stream, and can be revised based on lessons learned with each campaign.

The primary goal of the program is to provide an avenue for documenting decisions, procedures, and data pertinent to characterizing waste and preparing it for off-site treatment or disposal. Each shipment may require that certain elements of the program be revised, but the overall quality and consistency of each shipment will be assured. As LANL continues to develop this program, each LLMW shipment campaign provides new avenues for improvement.

INTRODUCTION

Until recently, treatment technologies and disposal capacities for low-level mixed waste (LLMW) generated at a Department of Energy (DOE) facility did not exist, requiring that all LLMW generated atLos Alamos National Laboratory (LANL) be placed in storage. Because of the recent developments in treatment technologies and disposal capacities, LLMW can now be shipped off site from LANL for treatment and disposal. Each off-site treatment, storage, and disposal facility (TSDF) has its own waste acceptance criteria (WAC) that must be met before the waste is shipped. To ship LLMW to any off-site TSDF, LANL must ensure that the waste meets these criteria. This requires obtaining accurate characterization information for the waste and certifying that the waste meets the waste management requirements of DOE Order 5820.2A, Radioactive Waste Management (1). To ship to a commercial facility, LANL must obtain an exemption from the DOE Order 5820.2A requirement that LLMW be shipped only to DOE facilities.

The most accurate characterization information for a waste stream is usually provided by sampling the waste and obtaining analytical data for

specified analytes. Sampling operations require strict compliance with quality assurance (QA) criteria to ensure the following: each collected sample is representative of the original waste stream, the analytical data provide a high level of confidence regarding the precision and accuracy of the waste stream constituents, and data quality objectives (DQOs) are met. The draft Low-Level Mixed Waste Off-Site Treatment and Disposal Management Plan (the Plan) being developed by LANL is quality-driven and compliance-based and provides procedures that ensure a quality program for sampling, characterizing, and shipping LLMW off site (2). The Plan encompasses selecting a qualified analytical laboratory; developing a quality project-specific sampling plan; properly sampling liquid and solid wastes; validating analytical data; and maintaining quality records. Implementation of the Plan provides documentation of decision processes, sampling strategies and procedures, deviations from established protocols, and data pertinent to the quality of the program. The goal of the Plan is to provide the waste certification official with a complete project package that demonstrates a quality program and manifests a high confidence level for the waste characterization data required to meet DOE requirements and the TSDF's WAC.

Although DOE Order 5820.2A is the regulatory driver for the exemption package documentation, this program focuses on establishing a process for obtaining and documenting accurate and complete waste characterization information, which is beneficial to all waste management activities. Therefore, even waste management activities not affected by DOE Order 5820.2A benefit from this program.

ELEMENTS OF THE PROGRAM

This program consists of the following elements:

- The DOE facility's waste management database
- A project database for the data being compiled
- A project-specific sampling plan
- A TSDF WAC checklist
- An analytical laboratory evaluation form
- An analytical data validation form
- A checklist based on DOE Order 5820.2A exemption request instructions
- The DOE exemption request package (ERP)

The majority of these elements can be developed at the outset of the program and used each time a shipment of waste is being prepared. For those elements that are shipment-specific, templates can be developed initially and customized to each shipment as it is being prepared.

Common Elements

The following elements of the program are common to all off-site shipments of LLMW and can be developed at the outset of the program.

- The waste management database
- A project database
- A checklist of the TSDF WAC
- An analytical laboratory evaluation form
- A form to document the validation of analytical data
- A checklist based on DOE Order 5820.2A exemption request instructions

The waste management database is the database used by the DOE facility to track its waste. To support this program, the database should have the ability to generate reports based on a key word search, the key word being the waste stream identifier or description. The generated report should provide container numbers and characterization information for the waste identified for off-site shipment.

The project database provides the quality record for the entire program, including the rationale for decisions made. Once this database has been formatted, data entry is all that is required for each off-site shipment of LLMW. This database can consist of fields for the following information:

Unique record number identifying the shipment (each shipment of waste constitutes its own database record)

Name, address, and telephone number of the TSDF to which the waste is being shipped

Waste stream identifier (the basis for the waste selected for shipment)

Project-specific sampling plan document number

Analytical laboratory selected

Container numbers, with access to the following subfields

- Rationale for eliminating containers
- Containers selected for sampling and analysis
- Analytical results

Characterization information for each container

Compliance of each container with TSDF WAC, with access to a subfield for any details or explanations

Comments pertaining to the program process

When identifying a TSDF available for the selected waste stream, it is important to obtain the most current WAC document applicable to the specified waste stream. The WAC is developed into a checklist, which is used to determine if each waste stream meets the WAC. Each checklist for different TSDFs can be formatted into the project database as a subroutine that is called up as applicable. Once the WAC checklists are developed, they are used for each shipment of waste to the specified TSDF. Revisions to the checklists are necessary only as the TSDF revises its WAC.

Selecting an analytical laboratory for sample analysis requires an evaluation of the available laboratories to determine which one is best suited for and capable of performing the necessary analyses. An evaluation form is developed using the criteria established in 10 CFR 830.120, "Quality Assurance Requirements," (3) the Environmental Protection Agency's (EPA's) Test Methods for Evaluating Solid Waste, Physical/Chemical Methods (SW-846) (4), and the Handbook for Analytical Quality Control in Radioanalytical Laboratories (5). The TSDF may also mandate its own requirements for the analytical laboratory, which must be taken into account when developing the evaluation criteria. These requirements can be developed as an attachment to the template created from the regulatory criteria.

To ensure that analytical data is of acceptable quality, it must be validated against established acceptance criteria. A data evaluation form is developed once for all analytical data. Specific sections of the form that do not apply to the data being validated can be marked "not applicable;" thereby eliminating the need to create new forms for each type of data. The criteria for data validation are based on the analytical method used, the analyte of interest, acceptable ranges for quality control (QC) sample results, and established DQOs. Specific validation criteria are obtained from EPA's SW-846.

Finally, to facilitate the preparation of a DOE ERP, a checklist based on DOE Order 5820.2A exemption request instructions is developed. This checklist can be developed by one DOE facility and shared with others. A selection of the elements of this checklist is provided below. Each

element is identified as being either common to all shipments (common), able to be customized from a template (template), or needing to be developed for each unique shipment (unique).

- Waste characterization program description (common)
- Waste certification program description (common)
- Waste moratorium implementation program description (common)
- Method(s) used to characterize the waste (template)
- Comparison of DOE Order 5820.2A requirements with the criteria used by the agencies regulating the TSDF (template)
- Options considered and cost estimates for these options, indicating that off-site treatment or disposal is optimal (template)
- Description of waste types (unique)
- Detailed waste characterization of the waste to be shipped (unique)

Template Elements

Templates can be developed for the following elements:

- Project-specific sampling plan
- Certain DOE ERP information

Developing a template involves generating an outline of the information required, providing the common information, and providing guidelines for developing the waste-specific information. Developing and using template information minimizes the need to recreate information common to all shipments of LLMW and to research requirements for information specific to each shipment, thereby facilitating the rapid staging of waste for shipment and optimizing cost effectiveness.

A project-specific sampling plan documents the specific sampling operations involved with obtaining representative samples from the selected waste containers for analysis. An outline for a sampling plan consists of the following elements:

- Sampling strategy and method used to ensure that the samples collected represent the originating waste stream
- Specific sampling protocol
- QA/QC requirements specific to sampling operations
- Health and safety concerns unique to each waste stream
- Roles and responsibilities of the personnel involved with the sampling operation
- Sample container and volume requirements
- Sample preservation methods
- Sample packaging requirements
- Sample shipping requirements, if applicable
- DQOs

Most of these elements can be taken from template information and customized to the specific sampling operation based on the waste stream. For example, sampling strategies and preservation techniques are provided in EPA's SW-846, generic sampling procedures can be developed to encompass most types of waste, and QA/QC requirements and DQOs should be consistent for all sampling operations. However, each of these elements, especially the sampling strategy, must be adapted to the characteristics of the waste being sampled. The sample container and volume requirements are based on the analytes of interest and the TSDF WAC; therefore, they need to be developed specifically for each sampling plan. Finally, although certain health and safety concerns are common to all sampling operations, specific concerns based on the characteristics of the waste being sampled must also be addressed.

Certain information required in a DOE ERP can be developed once and inserted into each ERP. For example, a description of the DOE facility's waste characterization program and waste certification program would change little over time. Therefore, these descriptions can be prepared at the outset of the first DOE ERP and copied into each one after that. The information describing the TSDF and its licenses, permits, and operating record can also be developed once and inserted into other DOE ERPs for waste being shipped to that particular TSDF.

PROCESS OF THE PROGRAM

A simplified flow diagram illustrating the program is provided in Fig. 1, which demonstrates that documentation occurs at each step within the program. This figure illustrates the skeleton of the program, with the details being provided in this report and customized as necessary for each DOE facility.

Fig. 1

The development of the DOE ERP parallels the preparation of the actual waste for off-site shipment. Much of the DOE ERP, such as descriptions of the DOE facility's waste characterization and certification programs and a forecast of mixed waste shipment campaigns, can be developed from template information. Other information required for the ERP, such as the characterization of the specific waste being shipped, is obtained through implementation of the Plan. The checklist of DOE Order 5820.2A exemption request instructions facilitates preparing the ERP in parallel with characterizing the waste.

The characterization program involves the following steps. These steps assume that the common elements and templates have been developed before beginning the process. The DOE ERP is prepared parallel to this process.

1. Select the category of waste stream for off-site treatment or disposal and enter it into the project database.
2. Identify key words describing the waste stream selected (e.g., chemical names, regulatory classifications, and chemical categories) that can be used to search the waste management database.
3. Query the waste management database to identify the containers of waste applicable to the chosen waste stream.
4. Enter the selected container numbers into the project database.
5. If, at any time throughout this process, containers of waste are withdrawn from consideration, document the rationale in the project database.
6. Review all available characterization information (i.e., hard copy and electronic) for the selected containers.
7. Enter the characterization information into the project database for each container.
8. Using the information available, complete the applicable TSDF WAC checklist for each container and enter the information into the project database.
9. Identify the containers that cannot conform to the TSDF's WAC and enter the reason in the project database subfield. Remove these containers from consideration for shipment.
10. Identify the analytical parameters required by the TSDF's WAC and enter them into the project database.
11. Based on the project-specific sampling plan and TSDF requirements, and following a documented sampling protocol, collect samples of waste from selected containers. Enter the sampling plan and sampling protocol

document numbers and the containers selected for sampling in the project database.
12. Maintain chain of custody for the samples by complying with a documented chain-of-custody procedure.
13. Maintain all records and logbooks for the sampling activity in the project files according to a documented records management procedure.
14. Define the analytical parameters and analytes of interest for each sample for the analytical laboratory.
15. Validate the analytical results received, completing the data validation documentation and entering the results in the project database.
16. Complete the TSDF WAC checklist based on the validated analytical results and enter the results in the project database.
17. Complete the DOE ERP.
18. Ensure that the waste is packaged according to TSDF and Department of Transportation requirements.
19. Upon obtaining the TSDF's and DOE's approval, ship the waste to the TSDF.
20. Review the project file to ensure that it is complete.

SUCCESSFUL IMPLEMENTATION OF THE PROGRAM

LANL successfully shipped 15 55-gallon drums of liquid scintillation fluid to Diversified Scientific Services, Incorporated, (DSSI) in Tennessee for treatment and disposal. Preparation of this shipment provided guidance for and insight into developing the program described in this paper. The lessons learned from this shipment campaign are also serving to enhance the productivity, efficiency, and flawlessness of the LLMW off-site shipment program.

DOE has reviewed and approved without comment the ERP for 1602 cubic feet of solvent-contaminated soil. This waste stream is being reviewed by Envirocare of Utah, and LANL expects to receive Envirocare approval. LANL is also anticipating DOE approval to ship to DSSI over 100 55-gallon drums of isopropyl alcohol waste and scintillation vial fluid waste by March, 1996. Using this program as a guide to the thought process, LANL intends to complete the disposal of the following radioactively contaminated waste streams, pending sampling, characterization, DOE approval, and acceptance at Envirocare:

- Soil with heavy metals in June, 1996
- Lead blankets in July, 1996
- Activated and inseparable wastes in August, 1996

As lessons are learned from different waste streams, the program is adapted. Throughout the process, LANL is gaining a better understanding as to how to solve problems that arise with each shipment. The resolutions to these problems are incorporated into the program. Finally, this program allows each person responsible for preparing a waste stream for off-site shipment to understand the process and ensure quality management for each shipment.

The success of this program allows LANL to expeditiously and cost-consciously ship LLMW off site and to demonstrate a reasonable and good faith effort to comply with New Mexico Compliance Order and FFCA. Because of LANL's successful implementation of this program, DOE has recognized the value added of the program and has subsequently increased funding to all expeditious shipments of LLMW. This has allowed LANL to expand the number of personnel from one person at program inception to a current number of over 10 subject matter experts who comprise the program core

competency group. This core competency group also utilizes the expertise of other LANL groups in the areas of transportation, waste profiling, sampling, document and database research, and acknowledges the efforts of DOE to ensure overall program success.

COST ANALYSIS

Although a direct analysis of cost savings resulting from the implementation of this program cannot be conducted because of the infancy of the program, indirect savings in level of effort can be seen. LANL expended 16 months of effort to prepare the soils for shipment to Envirocare. Upon implementation of this program, however, it took only nine months of effort to ship the scintillation fluids off site for disposal. As LANL continues to improve the program, the effort will become streamlined and a routine operation.

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ANALYSIS OF GAS PHASE EFFLUENTS FROM A PILOT SCALE PLASMA TORCH

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ABSTRACT

Evaluations of gas stream compositions and products of incomplete waste conversion are valuable for screening and controlling remediation processes. In this work, the effluent from a pilot scale plasma torch processing a surrogate waste has been analyzed by conventional gas analyzers, gas chromatography, GC/MS and FTIR spectroscopy. Initial studies on the use of on-line Air ICP are discussed along with development efforts for an instrument capable of quantifying

polychlorinated organics (PCB's dioxins) on-line. Results from the surrogate processing studies indicate a reduced oxygen gas stream along with the presence of NOCI. The formation of NOCI (from the HCI arising from PVC destruction and the NO generated from the plasma torch) is accompanied by the liberation of chlorine, The HCl, NOCI and CI2 represent acorrosive gas stream and attention is needed in the design of APCD systems for plasma torch units. Results from the on-line Air ICP indicated the presence of copper which was not a component of the surrogate feed and has been traced to the torch electrode. GC/MS experiments using surrogate waste spiked with hexachlorobenzene indicated a DRE of at least 99.99%. The results obtained thus far indicate the instruments employed for the measurements should be suitable for routine analysis and may find application in torch facility controls.

INTRODUCTION

Plasma torches and joule-heated melters are two technologies which have been selected for mixed waste stream remediation demonstrations by the U. S. Department of Energy. The high temperatures generated in the furnaces of these units permit the formation of a glass which is expected to contain the radioactive material along with metals from the waste. The processes realize a volume reduction and provide a means for separation and packaging of the glass.

Some of the current efforts at the Diagnostic Instrumentation and Analysis Laboratory center on the analysis and further development of plasma torch technologies. This work includes the evaluation of operational factors related to torch electrode life, the establishment of instrumentation for characterization of process parameters (gas and surface temperatures, velocities and turbulence levels), and the characterization of gas stream effluents which could include permanent, inorganic, and organic gases, trace organics (PCB's, PCDF's and PCDD's), heavy metals, and particulates. Control schemes integrating the signals from the various monitors and process instrumentation are under development. Details of the integrated plasma torch development program are presented in a companion paper submitted to this conference (1).

The work described here concerns the characterization of gas streams arising from plasma torch processing of a surrogate waste. Factors affecting torch operation including torch gas, power, and stand-off have been investigated. Measurements were performed using conventional gas analyzers, gas chromatography (GC), GC/Mass Spectroscopy (GC/MS), on-line and extractive Fourier transform infrared (FTIR) spectroscopy, and inductively coupled plasma spectroscopy (Air-ICP). The instruments provide a detailed analysis of the gas stream composition at various locations of the DIAL plasma torch facility. In turn, this information is critical for selecting the proper operating envelopes of the torch and air pollution control devices and for understanding the reactions occurring within and downstream of the primary torch furnace. The instruments described in this work, when combined in an integrated approach, represent a comprehensive means for the rapid and on-line characterization of gases, organics, polychlorinated organics and heavy metals arising from plasma torch operations.

Two major instrumental development efforts are also discussed. One concerns the use of an advanced MS/MS technique for the rapid on-line determination of target organic compounds (PCB's, PCDF's, and PCDD's) while the other method will allow for the quantification of a number of

trace metals at, or below, current regulatory levels. The advanced ICP work may also find application as an air pollution control device.

EXPERIMENTAL

All of the experiments were performed on the DIAL plasma torch facility (1). The system consists of a 250 kW plasma torch operated either with air or nitrogen as the torch gas. Powers investigated ranged from 80-190 kW. The exhaust from the torch furnace was routed to the DIAL combustion test stand facility and then to a scrubber and stack. Details of the plasma torch system are discussed in Ref. 1.

Experiments were performed with and without a surrogate mixed waste feed which contained poly(vinyl chloride), glass beads, pecan shell flour, perlite (a volcanic glass) and iron powder each at a percentage of 12.8 % by weight. Alumina, aluminum, Portland cement, activated carbon, and metal chlorides made up the balance. Percentages by weight were 19.6 for carbon, 8.8 for oxygen, 7.14 for chlorine, and 1.67 for hydrogen. Inert material comprised 63% by weight of the feed with the balance metals. Preparation of the feed as briquettes has been described (2). A number of different molecules may result from the destruction of PVC. In order to estimate destruction and removal efficiencies by GC/MS, some of the briquettes were spiked with hexachlorobenzene (HCB).

Sampling and optical penetrations are located on the torch furnace, on the transition section 8 between the furnace and the test stand, and at various locations along the test channel. All measurements were taken upstream of the APCD systems. Samples for the conventional gas analyzers were taken from a location on the transition section meters downstream of the furnace. Samples for analysis by gas chromatography were collected at the same location as that for the conventional CO2, CO, NOx, SOx and O2 measurements. The gas chromatograph was configured to utilize both FID (Flame Ionization Detection) and TCD (Thermal Conductivity Detection) capabilities through use of two ten-port gas sampling valves. Permanent gases were analyzed using a Molsieve 5A column with the TCD while the halogenated hydrocarbons were analyzed using a DB-624 column with the FID.

FTIR measurements were performed either in an emission or an absorption mode. Emission spectra were collected at a port location approximately 1 m downstream of the furnace and gas samples were routed from the facility penetration 2 m downstream of the torch. Details of the FTIR measurements have been reported previously (3).

Samples for GC/MS analysis were collected within the test stand channel (approximately 12 m from the furnace). HCB spiking levels of 10 and 25 mg were employed. Samples were collected cryogenically with collection times ranging from ten to twenty minutes. The volumetric flow rate through the sampling train was calibrated using a mass flow meter and glass wool was used to retain particulates. Samples collected throughout the series of experiments were acidic (HCl and HNO3), and varied in appearance. Sample workup involved a residue quality extraction protocol. GC/MS data were obtained on two instruments; one configured with a quadrupole ion trapsuitable for MS/MS experiments and the second, a quadrupole MS, capable of both positive and negative chemical ionization (PICI and NICI).

On-line detection of metals present in the offgas stream from the plasma torch (and most exhaust stacks) is a difficult problem using a standard argon ICP as the emission or ion source. Argon plasmas are rapidly quenched by the introduction of a molecular gas aerosol flow. The

standard sampling procedure involves filtering the gas stream, followed by dissolution of the captured metals and analysis in the laboratory. The turn-around time may be in terms of days rather than seconds.

One solution to this problem is the use of a molecular gas ICP. The principle advantage of such a system includes the ability to provide real-time sampling of an offgas stream. In addition, the increased thermal conductivity within a molecular plasma results in improved decomposition of refractory particles and increased plasma tolerance of high sample loading. AES work performed by Barnes & Meyers (1,2) in 1985 demonstrated the technique's viability (4,5). More recently, with the advent of the Clean Air Act, a need to provide an online metal emission monitor for molecular gas streams has been established. Baldwin et al. re-evaluated the air-ICP and in the process, demonstrated markedly improved detection limits (6). The elimination of the requirement for argon significantly reduces the operating costs. The disadvantages of such a system include the need for higher input powers (2.5 to 3.5 kW @ 27.12 MHz), lower excitational energies (max. ~12 eV for air as compared to 15.7 eV for argon) and the presence of a much more complex molecular spectral background.

An evaluation and optimization of the Air ICP as an emission monitor is in progress in these laboratories. This work includes optimizing the system with respect to the ICP operating parameters (input power, flowrate, sample introduction) and investigating spectral line selection, spectral interferences and matrix effects. The work is performed using an argon ICP modified to operate on air with emission detection using monochromaters, photomultiplier tubes and/or a CCD array.

The sampling of the exhaust gases by the Air-ICP is designed to occur after the HEPA filters, large particles having already been removed. During the preliminary experiments, the aerosol flow is introduced into the Air-ICP using a heated-head diaphragm pump 15 m from the torch facility. Several experiments were undertaken by sampling directly using a slipstream from the offgas with a filter being used to remove larger particles. As mentioned above, the increased thermal conductivity within a molecular plasma yields improved decomposition of refractory particles and increases the plasma tolerance for high sample loading. To take advantage of this and explore the use of the Air-ICP as a secondary combustor, the operation of the Air-ICP under direct sample loading (prior to filtering) is being evaluated.

RESULTS

Plasma torch experiments were conducted using air and nitrogen as the plasma gas. The results presented below correspond to operation of the torch with air. Sampleswere collected at intervals corresponding to the injection of briquettes or to a change in torch operating power. A comparison of gas composition as measured through gas chromatography (GC) and through gas analyzers (GAS) is presented in Fig. 1. Detailed data are given in Table I. The samples for GC and GAS analysis were obtained at the same location downstream of the plasma torch. Discrete samples for GC analysis were collected over an interval of two minutes in 1 liter Tedlar sample bags. Collection was initiated as soon as the on-line gas analyzer began registering a change in composition. For the on-line gas analyzers, the composition was obtained at 13 second intervals. The results for each gas analyzer, obtained during the collection of the discrete samples, were averaged and these values are presented for comparison with GC results. Of interest are the oxygen concentrations, which are reduced as

compared to normal air, and the levels of carbon monoxide. The lack of detectable carbon monoxide indicates that conversion of the organics in the surrogate feed are very nearly complete. The reduction in oxygen concentrations correlates to formation of CO_2, and of NO, NO_2, and other species. Agreement between the carbon dioxide results for the two analysis techniques is excellent. For oxygen, a systematic deviation between the GC and gas analyzer results is noted. This discrepancy may be a result of acid gases present in the plasma effluent which affect the performance of the oxygen gas analyzer. The presence of PVC can result in the formation of hydrogen chloride and chlorine gas, (see below) both of which are very corrosive. Quantification of these species through gas chromatography is currently under investigation.

Fig. 1

Table I

FTIR studies have been performed across the duct leaving the plasma torch furnace and in an extractive configuration with a folded path cell configured for a path length of 75 m. Figure 2 represents an absorption spectrum from the extraction measurements. In addition to CO_2, NO, NO_2, and HCl, the molecules HNO_2 and NOCl (nitrosyl chloride) were also observed. NOCl was also found in the in-situ experiments and the source of this molecule has been traced to the reaction

$$4NO + 4HCl \rightarrow 2NOCl + 2H_2O + Cl_2 + N_2$$

where the HCl is the principle transformation product from the poly(vinyl chloride) contained in the surrogate feed. The formation of NOCl is accompanied by the liberation of chlorine and further efforts in these laboratories are being directed at the fate and removal of these corrosive materials. A preliminary identification of the levels of chlorine in the plasma effluent has been conducted using color detector tubes. The presence of chlorine has implications in the use of secondary combustion. Specifically, the operation of a fuel-fired burner with an added chlorine source may lead to the formation of additional chlorinated organics.

Fig. 2

GC/MS experiments have been performed on samples collected during briquette injection and with spiked briquettes containing hexachlorobenzene. At the operating conditions tested, analysis of the raw surrogate material did not reveal the presence of polychlorinated biphenyls, polychlorinated dibenzo furans and polychlorinated dibenzodioxins. Qualitative analysis of samples collected with the spiked feed revealed that not all of the hexachlorobenzene was destroyed. Figure 3 shows a typical full scan electron ionization (EI) spectrum. The base peak at m/z 284 and the isotopic pattern associated with this peak represent six chlorine atoms, therefore characterizing the presence of HCB in the off-gas stream.

Fig. 3

The intensity of the m/z peak at 284 in the reconstructed ion chromatogram indicated a low level of HCB in the sample. This peak intensity is consistent with that of a nanogram (or less) of HCB injected on a GC column. Assuming complete quantitative trapping and extraction of the material, a destruction efficiency of at least greater than 99.99% for HCB occurs. Sampling conditions are anticipated to be less than ideal owing to the complicated sample matrix; consequently, the destruction efficiency (relative to the amounts of HCB present) could be slightly greater than that mentioned above.

In several experiments a compound was observed with spectra consistent with pentachlorinated biphenyls; however, correlation to the torch conditions proved difficult. Potential problems include those due to sampling and subsequent sample workup along with memory effects in the torch and manifold. Future work will concentrate on improving quantification, correlation of operational parameters to destruction efficiencies and sampling at different locations to determine if the distribution of analytes is consistent in the waste stream.

Over twenty years effort have been expended on optimizing the argon ICP to reach its present sensitivity. Very little of this essential work has been done on the Air-ICP. An air plasma is not only chemically reactive but the background spectrum contains complex molecular contributions absent from noble gas plasmas. This later artifact requires a careful re-evaluation of the selection criteria used to optimize the detection limits of the elements of interest. At present, research is being directed at the calibration of the instrument for use as an emission monitor of toxic metals. For example, the DIAL torch facility is processing feed that contains cesium, cadmium, lead, chromium, cerium, and nickel. The Air-ICP is being configured for the analysis using a slipstream and system optimization is in progress. Preliminary data (Fig. 4) demonstrate the presence of sodium plus copper from the torch electrode. On line analysis would then permit the assessment of electrode wear. In addition, the molecular spectra are complicated with the addition of contributions from other molecules (i.e., CN). Future work will focus on spectral line selection, calibration and improved quantification.

Fig. 4

INSTRUMENTATION DEVELOPMENT

The determination of trace organic molecules and metals is currently possible only through the extraction of a sample from the gas stream followed by analysis in the laboratory. Such information, while an average description of the process emissions at a given point in time, may not reflect rapid changes in the effluent and correlation to facility operations is delayed. Two projects are currently underway at DIAL which are anticipated to provide low limits of detection for large chlorinated organics and formetals in an on-line and rapid (<4 minutes) way.

An advanced MS/MS system is under development which should allow the quantification of large organic molecules (polychlorinated biphenyls, dioxins, etc.). The instrument is comprised of a molecular beam interface, a laser ionization source, and an orthogonal quadrupole ion trap housed in a six way cross vacuum chamber. A capillary tube and skimmer cone are mounted in a differentially pumped region and used to produce a continuous defined molecular beam. SIMION6, an electrostatic modeling program, was used to design the source region and lens stack necessary to direct ions into the orthogonal trap (7). The fourth harmonic of a Nd:YAG laser (266 nm) is used for multiphoton ionization of target compounds. The ion trap is controlled by a commercial electronics unit which also controls data acquisition. Resonant MS/MS capabilities (available in the control unit) will be used to enhance analytical specificity and sensitivity.

Additional efforts with the Air-ICP include the development of a fieldable system and application of the unit as a downstream pollution control device. A major drawback with the use of an Air-ICP for on-line monitoring of heavy metals has been the capital cost as well as the bulk

of the power supply. Air-ICP power requirements are on the order of 2.5-3.5 kW and have typically been beyond the capabilities of solid-state RF generators. DIAL has funded the development of a solid-state 3.5 kW 27.12 MHz RF generator and matching system. The unit will be rack mountable and will weigh 60 kg which is approximately 1/10 the weight of a standard generator. Air is supplied by a portable compressor thereby freeing the system from the supply requirements of the standard ICP.

As mentioned, different detection methods are also being explored. For complicated matrices it is sometimes necessary to employ high resolution monochromators of up to 1.5 meters in focal length. Ideally, for a fieldable system, the resolution of the monochromator is desired but the size is cumbersome. Baldwin has developed a fiberoptic interferometer which is expected to provide the resolution of a 1.5 m unit in a 0.5 m package (8). A protoype system is being developed and will be interfaced to the Air-ICP.

Another aspect of the Air-ICP work is the potential use of this system as a secondary combustor for the off-gas from a plasma torch. Plasma torch units typically operate at very low flow rates as compared to combustion-driven technologies. As an example, the 250 kW plasma torch used in this work normally operates in a range from 225-500 liters/min. (8-18 cfm). The addition of the large flow rates from a conventional secondary combustor may have many drawbacks including the fact that considerable amounts of additional mass are placed into the system. Reactions of the burner fuel with molecular chlorine and HCl may actually increase the chlorinated organics in the effluent An additional thermal load and added costs are also associated with using burner fuel. An air plasma has significantly increased thermal conductivity over an argon plasma. This, together with the presence of dissociated oxygen and nitrogen, makes the plasma very chemically-reactive. The ICP can rapidly destroy anyhydrocarbons without the introduction of the large quantities of additional gas and energy supplied by the propane burners. In addition, it may be possible to, by the addition of selected gases (e.g. hydrogen), control the plasma chemistry thereby reducing further the reaction end products.

CONCLUSIONS

Analysis of the effluent from a pilot-scale plasma torch processing a surrogate mixed-waste have indicated that the principle products of destruction are CO_2, NO_2, NO, and HCl. In most cases only low or trace concentrations of CO were found indicating complete conversion of the organics in the feed to CO_2. FTIR emission and absorption experiments have indicated that the NO from the plasma can react with the HCl (liberated from the PVC) to produce Nitrosyl Chloride. Although NOCl can be easily scrubbed from the gas stream the generation of Cl_2 accompanies this reaction. Problems may be encountered with downstream systems which are not designed for removal and handling of the corrosive gas stream. GC/MS experiments with added hexachlorobenzene have indicated destruction and removal efficiencies close to 99.99 %. Further studies are planned concerning various torch operating envelopes and different feed compositions and rates. Preliminary results demonstrate that the Air-ICP has an excellent future as a toxic metal emission monitor. However, it must be noted that considerable work remains in characterizing and optimizing the system. The use of the Air-ICP as a secondary combustor may have advantages other than merely reducing the thermal loading and

similar associated costs. If so, this may substantially reduce problems in downstream treatment. Further studies are planned.

ACKNOWLEDGMENTS

This work was supported by the U. S. Department of Energy through grant number DE-FG02-93CH-10575. We also acknowledge J. Andol, P. Herring, and A. Hodnett for operational assistance with the plasma torch facility.

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15-25

REAL-TIME MONITORING OF THE POLYETHYLENE MICROENCAPSULATION OF LOW-LEVEL MIXED WASTE AND SURROGATES BY TRANSIENT INFRARED SPECTROSCOPY

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ABSTRACT

Rocky Flats Environmental Technology Site and other DOE sites have large volumes of low-level mixed waste (LLMW) in powdered forms, which must be processed into stable waste forms suitable for long-term storage. Polyethylene microencapsulation, in which the waste is thoroughly mixed with molten polyethylene and extruded into containers, is the first LLMW stabilization process that can be monitored in real-time to avoid processing errors and provide regulators with proof-of-proper-processing documentation. Monitoring compatibility and the smaller, lighter, more economical, inert polyethylene waste form results in significant advantages over cementation and grout based matrices, which are prone to

chemical interaction with the waste. The monitor is a noncontact device that acquires the midinfrared spectrum of the process stream via emission spectroscopy. The stream composition is then derived from the spectrum using partial-least-squares techniques. Monitor operation is insensitive to the thickness of the process stream and to its surface morphology. The monitor has been demonstrated on polyethylene microencapsulation lines processing nitrate-salt LLMW and its surrogate, molten salt oxidation LLMW and its surrogate, and flyash. Typically, the monitor achieves a standard error of prediction of less than 1% by weight with an analysis time of less than 20 seconds.

INTRODUCTION

The DOE complex has large volumes of low-level mixed waste (LLMW) in powdered forms, such as salts and flyash, which must be converted into stable waste forms prior to long-term storage. Polyethylene microencapsulation is a waste immobilization process that produces a smaller and lighter final waste form than the cementation processes presently in use.(1, 2) Microencapsulation also avoids the chemical interactions with the waste that create cure problems for cementation.(2) Polyethylene microencapsulation is well suited for on-line monitoring that ensures the processed waste meets specifications required by regulators without experiencing costly processing errors.

The on-line monitor being developed for microencapsulation processing provides a continuous, real-time, quantitative record of the waste-loading concentration in the polyethylene matrix. This allows the process operators to better maintain the waste loading and processing conditions at optimum. They can also use the monitor analysis in developing treatment protocols for specific wastes. The immediate analysis shows the operators how the processing parameters affect the properties of the final waste form. In addition, the monitor log can assist in waste certification and act as an archival record of the waste composition. The monitor uses transient infrared spectroscopy (TIRS) to perform its analysis. TIRS is a noncontact method for acquiring the midinfrared spectrum of a moving stream of solid or viscous-liquid material. The stream composition is then derived from the spectrum using partial least squares analysis. Typically, the monitor determines the waste loading every 15 to 20 seconds with an accuracy of less than 1% by weight.

The monitor has been demonstrated on process lines encapsulating nitrate-salt LLMW and its surrogate (sodium nitrate),(1, 3, 4) flyash, and molten-salt-oxidation LLMW and its surrogate (a mixture of sodium carbonate and sodium chloride). Demonstration analyses done at Rocky Flats Environmental Technology Site on nitrate-salt LLMW and on flyash are discussed in this paper.

TIRS TECHNOLOGY

TIRS has been developed as a general molecular-analysis on-line monitor for use on process streams of solid and viscous-liquid materials. It has been demonstrated on a wide variety of moving materials, both in the laboratory (5, 6) and on process lines.(3) Figure 1 schematically shows how the TIRS monitor is set up on a waste-processing line.(4) The molten stream of polyethylene-encapsulated waste flows through the field of view of an infrared spectrometer. The molten stream, by virtue of its elevated temperature, strongly radiates in the midinfrared part of the spectrum, which the spectrometer observes. This bulk emission is a blackbody spectrum; it consists of emission at all wavelengths, and its intensity varies slowly with wavelength, following Planck's Law. This blackbody

spectrum is indicative of the temperature of the stream, but it is not useful in analyzing the stream composition. Spectroscopic analysis relies on the fact that sufficiently thin or dilute samples will absorb, emit, or reflect only those wavelengths that strongly interact with the molecules of the sample. The spectrum resulting from this interaction consists of a set of sharp features, called bands, that are characteristic of the particular molecules in the sample. The blackbody spectrum from the stream lacks these sharp features because the stream is too thick and optically opaque.

Fig. 1

The TIRS monitor avoids this blackbody problem associated with a thick, opaque process stream by focusing on only a thin surface layer of the stream. The TIRS monitor trains a small jet of room-temperature air onto the surface of the process stream as it flows through the spectrometer field of view. This cools a thin layer of the stream at its surface. This surface layer no longer emits infrared radiation strongly because of its reduced temperature, but the infrared spectrometer still observes infrared radiation from the rest of the process stream. This radiation must pass through the cooled layer to reach the spectrometer, and the cooled layer absorbs some of it. The cooled layer is sufficiently thin that it absorbs only at its characteristic bands. The spectrometer therefore observes the smooth blackbody emission spectrum of the stream bulk with the sharply structured transmission spectrum of the cooled layer superimposed on it. The process-stream analysis is based on the transmission spectrum.

For the analyses discussed here, a Bomem MB 100 FTIR spectrometer fitted with a mercury-cadmium-telluride detector was the infrared spectrometer around which the TIRS monitor was built.(3, 4) The TIRS cooling-jet nozzle was a length of 0.6 mm inner-diameter stainless-steeltubing aimed at the process stream and carrying a 1.6 L/min flow of room-temperature air. The partial least squares analysis of the data was done with Spectra Calc PLSplus (Galactic Industries).A TIRS monitor can also be used when the process stream is not at an elevated temperature. For cool process streams, a TIRS monitor uses a hot-gas jet to produce a heated, thin, surface layer.(3, 5) The structured emission from the thin layer can then be analyzed in the same manner as the structured transmission spectrum of the cooled layer produced when a cooling jet is used. Early work on applying TIRS to polyethylene encapsulation examined both cooling-jet and heating-jet versions.(3, 4) Both versions gave good quality analyses, but the cooling version is simpler and slightly more stable, so it was selected for the applications discussed here.

DATA TREATMENT

Spectra acquired by the TIRS monitor are "single beam," that is, they are the raw signal intensities observed as a function of wavenumber. Usually in spectroscopy, such single-beam spectra are "normalized" by ratioing them against a reference single-beam spectrum so as to remove any wavenumber-dependent variation in the spectrometer response. In TIRS, this normalizing also removes the Planck's Law rise and fall of signal intensity with wavenumber that is characteristic of thermal emission. Figure 2, discussed below, shows both the single-beam and normalized versions of a spectrum. The analyses discussed here are based on normalized spectra. The reference spectra used were blackbody emission spectra observed from the process stream when the TIRS cooling jet was turned off. It should be noted that the analysis can be done based on

single-beam spectra, but normalization can often result in improved accuracy.

Partial least squares is used to derive the stream composition from the spectra.(7) The partial-least-squares method calibrates the monitor by building a model from a training set of spectra. For the work reported here, the monitor acquired the training-set spectra while the waste processor is held at a series of stream compositions with other process parameters (e.g., stream temperature and flow rate) held strictly fixed. More readily applied methods of acquiring the training-set data are being developed. A prediction as to how good the resulting calibration model will be is the Standard Error of Prediction (SEP) produced by single-elimination cross validation. In a single-elimination cross validation, one of the training-set spectra is removed and a calibration is developed using only the remaining members of the set. The composition of the excluded member is then determined from this calibration. The excluded member is then put back in, a second member is taken out, a new calibration is developed, and the second member is analyzed. This process is repeated until all members of the training set have been excluded and analyzed. SEP is the root mean square of the deviations of the analyses from the known sample compositions. Once the calibration model is built, analysis of an unknown spectrum is very fast (under 1 second), so partial least squares is very compatible with real-time analysis.

DEMONSTRATION ANALYSES

Figure 2 shows spectra from demonstrations at Rocky Flats on two different processed-waste streams. The spectra in the top panel of the figure are from a molten stream of nitrate-salt LLMW in polyethylene. The two spectra labeled With Jet and Without Jet are single-beam spectra of a stream containing 55% by weight LLMW. The Without Jet spectrum is the spontaneous blackbody-like thermal emission spectrum of the stream observed with the TIRS cooling jet turned off. The With Jet spectrum is the TIRS spectrum observed by aiming the jet onto the stream. The overall rise and fall in intensity of these spectra follow Planck's Law for thermal emission, but the drop off at low wavenumbers is augmented by decreasing detector sensitivity. The sharp structure between 1300 and 1900 cm-1 are absorptions by water vapor in the air between the process stream and the spectrometer. Similarly, the features between 2300 and 2400 cm-1 and the one sharp feature at 667 cm-1 come from carbon dioxide in the air. The lower intensity of the With Jet spectrum compared to the Without Jet spectrum indicates the lower temperature of the process stream surface resulting from the cooling jet. The With Jet spectrum also has the features from the transmission spectrum of the cooled surface layer, but they are small and do not immediately catch the eye.

Fig. 2

Normalization makes the transmission features obvious by removing the Planck's Law intensity variations and the water and carbon dioxide absorptions. Division of the With Jet spectrum by the Without Jet spectrum results in the normalized 55% Waste spectrum in Fig. 2. The 35% Waste spectrum is a normalized spectrum at a different stream composition included for comparison (and offset to avoid overlap). All of the features in the normalized spectra can be assigned as absorptions by either the LLMW or the polyethylene (except for some noise above 3000 cm-1). Polyethylene produces the strong bands at 2850 and 2910 cm-1 and the moderate band at 720 cm-1, as well as other absorptions at 1300, 1360 and 1460 cm-1 that are covered up by LLMW bands (but are visible in the

spectra in the lower panel of Fig. 2). The remaining bands all arise from the LLMW. The intensities of all bands within a spectrum are dependent on noncompositional parameters such as stream temperature, but the relative sizes of the LLMW and polyethylene features indicate the stream composition. The polyethylene bands decrease relative to the LLMW bands in going from 35% to 55% waste.

The lower panel of Fig. 2 shows (offset to avoid overlap) normalized spectra of a stream of polyethylene-encapsulated flyash at the indicated compositions (% by weight flyash). All of the features in the 0% Flyash spectrum come from polyethylene, except for noise above 3000 cm-1. Flyash has a single, broad absorption between 600 and 1600 cm-1 that grows with increasing flyash concentration as the polyethylene bands shrink.

Fig. 3

Figure 3 shows the results from a single-elimination cross validation on six spectra of encapsulated nitrate-salt LLMW (including those in Fig. 2). The TIRS-predicted waste loadings are plotted against the actual waste loadings, and the linear-regression line for the points is shown. The ideal result would be a regression line that exactly bisects the plot (i.e., identical predicted and actual values). The real line comes very close to this. The SEP for the cross validation is 0.64% by weight, demonstrating an accurate TIRS analysis. Fig. 4 shows a single-elimination cross validation of 21 spectra of encapsulated flyash (including those in Fig. 2). As in Fig. 3, the TIRS predictions are plotted against the actual values (open circles) and a linear-regression line is included. The SEP is 0.84% by weight. In addition, the analyses from a set of "unknowns" not included in the cross validation are marked by triangles.

Fig. 4

CONCLUSION

The TIRS monitor provides accurate, real-time analysis of the composition of polyethylene-encapsulated waste streams. The TIRS monitor is a noncontact device that acquires the midinfrared spectrum of the moving process stream. It has been demonstrated on DOE process lines encapsulating nitrate-salt LLMW and its surrogate, molten salt oxidation LLMW and its surrogate, and flyash. On nitrate-salt LLMW and flyash, the specific demonstrations discussed in this paper, the monitor achieved a standard error of prediction of better than 1% by weight with an analysis time of less than 15 seconds.

ACKNOWLEDGMENT

Ames Laboratory is operated for the U. S. Department of Energy by Iowa State University under Contract No. W-7405-ENG-82. This paper was supported by the Office of Technology Development.

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THERMAL CHARACTERIZATION OF MIXED WASTES

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ABSTRACT

A method for analyzing small samples of mixed wastes (MW) by a new invention known as the laser driven thermal reactor (LDTR) is presented. The processes of pyrolysis, volatilization, combustion, and thermal decomposition can be studied. The LDTR allows one to measure the kinetic parameters of these processes. Also, compositional analysis of gas phase products by gas chromatography can be realized using the LDTR. As an application of this technique, the volatilization of radionuclides, as indicated by b and g radioactivity of gaseous effluents, and residual activity of solid residues will be determined. It is also shown how a LDTR-based thermal analysis of a solid residue of MW under different heating rates will provide quantitative information about its stability and the existence of trace organics. In conclusion, it is shown that the LDTR represents a valuable test device for characterization of waste destruction processes.

INTRODUCTION

The purpose of the Laser Driven Thermal Reactor (LDTR) (1-3,7) is to measure the thermal parameters of processes for the destruction of organic and mixed wastes to enable one to develop mathematical models of the processes with numerical parameters. The models, in turn, permit one to determine optimal parameters for the thermal treatment of the wastes in large-scale systems. Because the LDTR requires only small samples (~10-3g), it is particularly useful in analyzing the thermal treatment of low level radioactive organic compounds. Further, through the measurement of the radioactivity of gaseous products and of the solid residues of the LDTR, it is possible to optimize systems for the treatment of such wastes.

A general schematic of the LDTR is given in Fig.1. Two hemispheres fabricated from copper foil form a spherical reactor, within which the waste sample, supported by a substrate, is located. The reactor is irradiated by cw YAG laser beams from two sides and is heated up rapidly to a temperature of 1000 K. The rate of heating can reach up to several

hundred degree Kelvin per second. Temperatures on the surface of the reactor vary by no more than 100 K. The sample temperature rises rapidly with increases in reactor temperature. Exothermal or endothermal processes of the sample at high temperatures lead to an additional increase or decrease of the temperature of the sample and substrate. The time dependencies of the reactor and sample temperatures permit one to determine the parameters for the thermal treatment of the sample.
Along with a determination of the thermal parameters of the process, the gaseous products are analyzed. In addition, the sample weight is determined before and after the process. All of this permits one to describe the process in the form of a mathematical model with numerical parameters.
Fig. 1
AN ILLUSTRATION OF THE LDTR METHOD
To provide an example of the determination of thermal parameters by the LDTR method, let us consider a one-stage process. The specific heat release (the energy released per unit mass of the sample), $Q = Iq(T)dt$, is the principal parameter of interest, and shall be derived. We assume that a sample is complex substance and consist of combustible (e.t. organics) and non-combustible parts. For a determination of the parameters of interest, it is necessary to analyze temperature evolution dependence of the reactor and of the sample with substrate. This method assumes that the temperatures of both the sample and substrate are the same during the process, as well as uniform throughout these substances. Therefore, this method is valid for slow processes with typical completion times exceeding 0.01 s.
Then the equation in terms of the temperature T of the sample with substrate may be expressed by the heat balance:
Eq. 1
where Tr is the reactor temperature, cp is the specific heat of the sample with substrate, m is the total mass, P(Tr) is the rate at which heat is transferred from the reactor to the sample with substrate, and R(T) is the rate of heat loss from the sample with substrate. At equilibrium, a certain sample temperature T corresponds to each reactor temperature Tr , requiring the condition,
Eq. 2
During the course of heating, the sample temperature will differ from its equilibrium temperature because of transient response to heat transport. We will give below a method for determining the equilibrium sample temperature from two measurements.
Let the reactor temperature Tr be constant, and the sample temperature be T, which will differ from its ultimate equilibrium value To. Then the process of sample temperature evolution is given by the equation:
Eq. 3
where t is the temperature dependent relaxation time to the equilibrium, which may be defined in any infinitesimal time interval as:
Eq. 4
On a basis of two measurements, for the temperatures of reactor, and for the substrate with the sample, one can determine the equilibrium temperature To and relaxation time of the system. They are given by formulas:
Eq. 5
Eq. 6

where T1 and T2 are sample temperatures for two different heating rates 1 and 2 which result in these temperatures and (dT/dt)1, (dT/dt)2 are their derivatives. Determining the parameter t allows us to analyze exothermic chemical processes. This parameter can be determined not only by the measurement of delay in the sample temperature rise during heating of a reactor, but also by another method: the sample can be heated by laser radiation separately at the constant reactor temperature, and then t is measured directly as the relaxation time of the sample temperature. The instrumentation for measuring T is shown on Fig. 1.

Thus we have the following equation for the sample temperature, derived from the Eq. 1 heat balance, with expression of R(T):

Eq. 7

Here cp', m' are the specific heat capacity and mass of the combustible part of the sample, respectively, and q is the specific heat release rate of the process. This equation allows one to determine the specific parameters of combustion on the basis of measurement of the heat release of the process. In particular, if the mass of combustible part is small compared to the sample mass m' m and it's consumed in the course of the process, we have the following set of equations which describe the heat process, from Eq. 7:

Eq. 8

Eq. 9

where Q = q(t)dt is the specific heat release of the combustibles. From this, the algorithm of the treatment of measurements follows. We then have these equations:

Eq. 10

Eq. 11

where DT = Tf -Ti, and Ti, Tf are the sample temperatures at the beginning and the end of the process, respectively. Since Dm is the change of the combustibles mass, Eq.11 permits one to determine Q, the specific heat release of the combustibles. Then from Eq. 9 one can determine m'(t), and Eq. 9 gives the value of the specific heat release rate

Eq. 12

This methodology pertains to one-stage processes. Comparing values of q(t) at different rates of heating, one can determine the validity of a model of a one-stage process in a concrete case, and also determine the accuracy of numerical experimental data.

The method has two requirements for reliability and accuracy. First, the position of the substrate inside the reactor must be the same in different experiments. Secondly, although the temperature can be varied over the reactor surface, this distribution must be the same in different heat regimes. Measurements of temperature at one point on the surface of the reactor can be sufficient to verify the second requirement, and can be used for comparison with the temperature evolution with and without a sample. The degree to which these requirements are violated determines the accuracy of the method. Note that heating of the reactor can be made not only by laser, but also by electrical heating of the reactor in the analysis of metallurgical processes. However, the laser-based method of heating offers some advantages over electrical heating, including higher heating rate, better controllability, and direct determination of heat loss.

EXAMPLES OF ANALYZED PROCESSES

Below we consider some processes which are analogous to thermal treatment of wastes and which were analyzed by the above method. Organic wastes can consist of a mixture of various organic compounds, and variations in the content of components of this mixture can require changes in the optimal parameters of the phenomenological thermal processes. Therefore, for optimization of the process of treating a given mixture, it is necessary to determine numerical parameters of the process specifically for this waste. However, the qualitative character of the process is the same for different wastes. Therefore, first let us consider the general peculiarities of the thermal treatment of wastes. There are two forms of wastes for thermal treatment. In the first case, there is inhomogeneous mixture of individual particles and, in the other case, there is liquid waste.

In the first case the total waste volume takes part in combustion, and a suitable model for this process is the combustion of a porous or powdery organic substance. Let us take activated coal as a model for this process. Different components of a waste are burned at different temperatures and the thermal process can proceed for them in different ways. Indeed, combustion of organic volatile matter (VM) which takes place at low temperatures, is accompanied by vaporization of these components. The competition of these processes is such that at high temperatures, combustion of the VM proceeds quickly, while at low temperatures they can volatize and leave the reactor without combustion. Thus the probability of combustion of volatile matter in a reactor depends on the rate of heating and the size of the combustion region. Within the framework of this method, the value of this probability can be calculated for certain parameters of the thermal reactor.

Let us consider these processes in the case of thermal treatment of an activated coal in oxygen or air, which is a model for combustion of solid organic wastes consisting of small organic particles. One can separate the combustion of a VM and of the main solid residue. This process proceeds according to this scheme (1,2):

Eq. 13

Thus a part of a coal VM can be vaporized and does not partake in the combustion process inside a coal. It becomes a factor in the heat release inside a coal and influences the temperature. Clearly, the greater the gas pressure and temperature, the more likely it is that combustion takes place inside a coal.

Let us introduce the probability Pcom that combustion of a VM proceeds inside a coal and the probability Pout of a leakage of a VM outside a coal. We have

Eq. 14a

and the ratio of these values is approximated by the Arrhenius law:

Eq. 14b

Parameters of this formula and conditions of measurement are given in Table I and this function is represented in Fig. 2 for the activated coal for two different oxygen pressures. If the oxygen pressure or the sample temperature are high enough, the chemical process goes according to the first mechanism (13). The parameters of this process - the specific chemical energy Q and the specific power of the process q(T) - are given in Table II for the combustion of an activated coal.

Fig. 2

Table I

Table II

Let us analyze the process of the combustion of a coal VM. It is described by 5 parameters that can be used if this process proceeds in a certain system. For example, if coal particles are burned out in a special energy facility, these parameters permit one to choose optimal conditions for this facility. From the data in Tables I, II, one can assume that parameters of the chemical process depend on the size of coal particles. The data show a strong dependency on the type of a coal, which can be accounted for in the estimation of optimal conditions of an energy facility. For example, if we change a type of coal in an energy reactor, the optimal conditions inside also change. Thus, it is necessary to obtain parameters of the chemical process for a given type of coal, providing an opportunity to evaluate the optimal conditions for this reactor.

The formation of gaseous components is of importance for the thermal treatment of organic compounds, both in the intermediate stage of the process, and as final oxidation products. As an example of such measurements, we can consider combustion of an activated coal in ozone, which is absorbed by coal. Accepting products of this process to be CO and CO2, assume the concentration of these compounds to be dependent only on the absorption temperature T. Figure 3 shows this dependence, obtained on the basis of chromatographic measurements (6). These data are included in the model of the process.

Fig. 3

As a model of the thermal treatment of liquid organic materials , the process of heavy fuel oil (HFO) combustion was analyzed. The first stage of this process is transformation of a liquid fuel to a gas which proceeds according to scheme:

Eq. 15

This measurement gives the specific absorption energy 0.7 0.1 kJ/g, corresponding to n = 7-10. The second stage of the combustion process leads to combustion of 30 10 % of the fuel under the conditions considered. The rate constant of the process in the temperature range750 - 1300 K and oxygen pressure range 4 - 23 kPa is approximated by the formula

Eq. 16

where q is expressed in W/g, p is the oxygen pressure in kPa and x = 1.4 0.3.

The above data demonstrate abilities of the LDTR method. Note that along with parameters of the chemical process, this method permits one to obtain thermal parameters of a sample (the heat capacity, radiative parameters) which are of interest in the calculations of the processes considered. Some examples of this capability were demonstrated in (7).

CONCLUSION

Thus the LDTR method allows one to determine parameters for the thermal treatment of any organic materials. These parameters can be used for optimization of processes involving multiple reactions of organic wastes. Moreover, if the content of organic components in a waste changes from one stage to another of an industrial process, it is useful to make such measurements prior to large-scale implementation. Because such measurements may be automated and can be interfaced with computer programs for control and optimization of these processes, this approach can lead to cleaner and more effective technologies for the thermal treatment of wastes.

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WASTE MINIMIZATION APPLICATIONS FOR ENVIRONMENTAL RESTORATION AT THE NEVADA TEST SITE

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ABSTRACT

Traditionally, waste minimization has been applied to process line applications, but not to clean-up operations, such as environmental restoration (ER) and decontamination and decommissioning (D&D) operations. Because the purpose of ER and D&D activities is, essentially, to generate waste, waste minimization was thought not to apply. In 1994, the U.S. Department of Energy provided limited funding to a pilot project at the Nevada Test Site to determine whether traditional waste reduction techniques, such as pollution prevention opportunity assessments, could be applied to ER and D&D projects. The purpose of the pilot was to demonstrate that there are many repeatable elements in ER and D&D activities; therefore, traditional waste reduction techniques could be applied. In addition, tools such as up-front planning, regulatory negotiations, employee awareness campaigns, and involvement of pollution prevention personnel in all phases of a project, can be used to greatly reduce the amount of secondary waste generated by ER and D&D projects. Legacy waste can also be reduced through recycling or reusing building materials, planning for waste minimization in selection of treatment and characterization technologies, and negotiating clean-up levels. The success of the program has led to additional funding to study other methods of applying waste minimization and pollution prevention to ER and

D&D projects. To date, five pollution prevention opportunity assessments have been completed for Nevada ER projects, and over 30 tons of scrap metal have been recycled from clean-up operations at the Tonopah Test Range.

INTRODUCTION

In November 1989, the Secretary of Energy established the Office of Environmental Restoration and Waste Management (EM) to improve the management of remediation, waste management, and facility decommissioning by consolidating these missions into one office. In Nevada, Environmental Restoration (ER) activities are under the auspices of the Environmental Restoration Division (ERD) and involve activities at the Nevada Test Site (NTS), Tonopah Test Range (TTR), and sites in central Nevada, Colorado, Mississippi, Alaska, and New Mexico.

The NTS is a Federally-controlled facility located 105 kilometers (65 miles) northwest of Las Vegas. The NTS occupies 3,510 square kilometers (1,350 square miles). From 1951 until 1992, the NTS has been the location for numerous above- and below-ground nuclear tests. These tests and their associated operations have contaminated certain areas of the NTS with various radioactive and hazardous materials.

The ER Project field activities at the Nevada Test Site (NTS) include decontamination and decommissioning of various facilities, and characterization of a variety of industrial sites, including ordnance detonation areas; unexploded ordnance sites; unlined surface impoundments, injection wells, and underground storage tanks; and the deep aquifer beneath the NTS.

WASTE MINIMIZATION APPLIED TO ENVIRONMENTAL RESTORATION ACTIVITIES

The pollution prevention awareness campaign, which includes an employee awareness and incentive program, has been the source of many ideas that have been incorporated into ER activities. For example, at a project which involved detonation of small cluster bombs, an employee suggested that cardboard tubing be used instead of polyvinyl chloride (PVC) pipe. The cardboard leaves behind no potentially hazardous waste residue, eliminating the need to sample for PVC degradation productions. At the clean-up of an historic landfill at the TTR, staff worked closely with recycling companies and were successful in recycling over 30 tons of scrap metal (1). Employees have also suggested rinsing and reusing plastic bottles that contain bromide standards; previously, these bottles were thrown in the trash after one use. Another employee suggestion resulted in the reuse of laboratory standards for measuring pH in the field. Other employee suggestions included using only recycled paper, establishing an office recycling program for aluminum cans and paper, and turning off lights and computers at the end of the day. These suggestions result in awards such as polo shirts and car sun shades bearing the campaign logo. Employees also participate in a variety of community outreach activities involving waste reduction such as a beach clean-up at Lake Mead, restoration of an area formerly used for desert dumping, and working with Scouts on environmental merit badge projects. While these suggestions and community outreach activities do not directly impact field activities, they help keep employees aware of the importance of resource conservation.

Waste minimization has been an integral part of the up-front planning for all ER field activities. Part of the up-front planning process includes involving waste minimization experts during development of site characterization plans and field procedures. Through the involvement of

waste minimization staff, procedures were changed so that decontamination of sampling equipment is accomplished using no regulated decontamination solvents; thereby avoiding generation of hazardous or mixed waste. The up-front planning also includes working closely with regulators to determine the proposed action and clean-up levels and regulatory strategies for managing investigation-derived waste. Future land use negotiations may lead to a lesser need for clean-up and characterization. Regulatory strategies can also be negotiated. For example, the Underground Test Area (UGTA) Subproject involves characterization of the deep aquifer beneath the NTS and produces millions of gallons of fluid in the form of groundwater and drilling fluids. At wells where radioactive contamination is not present, the fluid is managed as industrial effluent under the Nevada Water Pollution Control Act (similar to Safe Drinking Water and Clean Water Acts), which allow for discharge of the fluid into unlined infiltration areas as long as fluid quality objectives are met. If fluids do not meet fluid quality objectives, they are allowed to evaporate in lined sumps at the wellsite. The remaining uncontaminated solids are managed as industrial waste and left on site; contaminated solids are transported to an appropriate waste management facility (2).
Waste minimization is also accomplished through application of pollution prevention opportunity assessments (PPOAs). During FY 1994, the U.S. Department of Energy provided limited funding to pilot PPOAs for the Nevada ER Project. The UGTA Subproject was selected because it is a long-term project (projected to span 20 years) and has many repeatable elements. The aquifer beneath the NTS is thousands of feet deep; during drilling, millions of gallons of fluid are generated. If contaminated with radioactive or hazardous constituents, the solids must be managed as radioactive or hazardous waste. Students from the local community college were employed to form the core PPOA team, with drilling program and waste minimization staff assisting as technical experts. The PPOA investigation found a method to separate and reuse drilling additives (3).

Because of the success of the pilot PPOA program, additional funding was obtained in FY 1995 and additional college students were employed to conduct PPOAs of other ER activities, such as field sampling and decommissioning. These students are employed as part of a cooperative internship with the local university's Environmental Studies Program. Students receive classroom credit for job experience.

During 1995, five PPOAs were completed and four more are underway during 1996. These PPOAs included: four for buildings associated with decommissioning activities; one for compaction of radioactively-contaminated trash; one for life cycle costs of concrete recycling; and three for industrial sites scheduled for characterization and remediation. The PPOAs for the decommissioning activities include examining costs versus benefits of 1) recycling various building materials and appurtenances, 2) decontamination versus disposal as radioactive waste, and 3) innovative decontamination technologies. The initial PPOA conducted for a building slated for demolition resulted in segregation of building materials and chipping radioactive "hot spots" off of the concrete walls and floors. This allowed the majority of the concrete to be treated as reusable material or industrial trash rather than radioactive waste. The PPOAs for the industrial sites include examination of various characterization and remedial technologies that may reduce waste generation. For example, one PPOA examined various drilling techniques for determining the extent and magnitude of a

hydrocarbon plume. Characterization techniques examined included the cone penetrometer, the SEAMIST System, and conventional drilling. Another PPOA examined various alternatives that could be used to remediate a hydrocarbon plume, such as vapor extraction and bioremediation.

Upcoming waste minimization projects include development of a training video focusing on ER applications for waste minimization. The training video will be nationally distributed in late 1996 and will explain how waste minimization tools can be applied to ER activities. The basis for this video will be the DOE Headquarters' Guidance Document for application of pollution prevention to ER projects and information gathered from various DOE sites as to how they apply pollution prevention tools to their projects. All ER characterization and remediation plans are now required to include a PPOA as part of the plan. Waste minimization staff are part of the project team for each characterization and remediation activity and work closely with project staff to determine the amount and direction of research required to examine pollution prevention opportunities.

SUMMARY

Up-front planning is the most important tool in applying pollution prevention and waste minimization to ER activities. It is imperative that waste minimization concepts be included in site characterization plans and procedures. By including waste minimization into the early stages of a project, opportunities to prevent pollution and minimize waste can be identified and benefit the entire project. Understanding issues such as whether decontamination is cost effective for certain building materials or which treatment technologies are the most cost-effective and environmentally beneficial can help guide characterization plans and activities by ensuring that adequate data are gathered in one sampling event. Negotiating clean-up and action levels with regulators can also lead to minimized characterization and remediation, thereby reducing cost and waste. PPOAs are a viable technique for conducting cost/benefit analyses of ER activities. The PPOA technique helps ensure that decisions are thoroughly documented and can help form the groundwork for restoration technology application decisions.

Previously, pollution prevention and waste minimization techniques were not routinely applied to ER activities; the focus of ER is usually to generate waste through site clean-up. However, through activities such as the PPOAs conducted at the Nevada Test Site, pollution prevention and waste minimization are now seen as the keys to the future in reducing costs and liability for the DOE.

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15-28

A RISK BASED COMPARISON OF REMEDIAL ACTION ALTERNATIVES INCLUDING WORKER RISK

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ABSTRACT

Current budget planning strategy within the Department of Energy emphasizes risk reduction potential as a basis for prioritizing expenditures within the complex. To implement such a strategy, a consistent methodology must be available for estimating health risk to workers as a result of project activity. This report provides a comprehensive methodology for considering worker risk in the evaluation of project risk reduction potential. An example comparison of alternative projects to remediate surface impoundments containing radionuclides and hazardous chemicals at the Oak Ridge National Laboratory is used to illustrate the methodology.

Results of this example evaluation clearly indicate that the choice of a preferred environmental management alternative can be affected by considering worker risk. In particular, the result of "net valuing" potentially achievable reductions in long term environmental risk against incurred worker risk tends to support a conclusion that less complex remediation alternatives have a higher net risk reduction value.

INTRODUCTION

Prioritization of Projects Using Risk Based Cost Benefit

DOE field operations are currently documenting an integrated risk based prioritization of proposed activities in support of a complex wide risk based budgeting process. The intent of the integrated risk based cost benefit process is to establish a comprehensive and consistent knowledge base of information for each activity related to environmental management in the complex. Examples of activities include a diversity of DOE expenditures such as remediation projects, waste stabilization efforts, and surveillance actions. A consistent risk based cost benefit methodology requires the capability to evaluate achievable reduction in environmental risk over time, projected cost over time of an activity, cost per unit of risk reduction, measures of the potential importance and impact of assumptions and uncertainties, and other measures which address important activity values not included in the primary cost benefit measure, such as progress toward overall cleanup goals, impact on stakeholder perceptions, and projected land use scenarios.

An infrastructure to supply these necessary elements of a cost benefit methodology addresses the intent of DOE's original Report to Congressa on risk based prioritization, yet it bespeaks a need for a technical process which goes beyond the current methodology for Risk Data Sheets.b It includes:

- an independently defined database of quantitative risk information on environmental risks and accrued worker risks,
- standardizing criteria which address inherent problems of estimating risk using documents of varying rigor and age,
- consistent assumptions and criteria for considering land use impacts,
- an independent and consistently applied set assumptions and uncertainty characterizations for risk estimates,

an independent methodology and database for estimating costs including cost of compliance, site overhead, and public perception management.

Role of Involved Worker Risk in Valuing Risk Reduction

A common source of error in establishing risk based budget priorities stems from the way in which risk reduction is estimated. For example, a common practice, in appraising the value of remediation activity, is to derive estimates of reducible environmental risk using fate and transport models with data from site characterization efforts. What is effectively assessed is an upper bound for achievable reductions in risk to the public as a result of actions which remove or immobilize radioactive and chemically hazardous materials. In other words, risk reduction potential is characterized almost entirely by the inventories at risk, the phenomenology of release, and the available pathways to man. When environmental risk projections are compared across multiple proposed projects or alternative remedies, the resulting "comparative" risk assessment, an indicator of project value and therefore priority, is based primarily on the physical characteristics of the sites and the long term hazards.

In effect, environmental risk reduction, a measure of the risk associated with "doing nothing" neglects the potential risk of "doing something", namely the risk to workers and the public as a result of an activity (or worker risk). Worker risk measures the inherent risk of implementing a project or remedy, the potential for excess cancers due to exposure due to radioactive materials and hazardous chemicals, and for fatalities due to industrial hazards.

As late as the publication of the Draft Report to Congress, estimates of worker risk were not commonly included in risk based evaluations of activities. This oversight may be a vestige of thinking which posits that there is an "implied consent" on the part of government contractor employees to accept risks over and above those involving the population as a whole, or that risks to workers are eliminated entirely through design considerations, worker protection, and operational precautions. Problems arise when these assumptions are invoked in estimating the value of remediation activity.

Involved workers (those who could be presumed to consent to excess risks) represent only a part of the population of persons potentially exposed to increased risks. Implied consent is less reasonable for collocated workers (on a site but not associated with ongoing project efforts), and is inapplicable for members of the public who might be exposed to accidents (especially where transportation is a key element of a proposed remedy).

Recent class action suits against DOE and its contractors by organizations representing employees and former employees indicate a much lower tolerance for job related risks and less of a willingness to support previous covenants regarding "accepted" on the job risks. Neighbors and the media, in general, rarely distinguish between workers at a facility and an "innocent" public.

It has been noted in the Report to Congress that worker risk can have a significant impact on the net achievable risk reduction and therefore the net value of a specific project or proposed remedy. It is well recognized that much of the contaminated land and facilities at DOE sites cannot be returned to public use without an investment in funds which is not affordable to this societyc. Worker risk is a potentially significant

impact on achievable risk reduction which must be considered in risk based budget allocations.

Attributes of a Worker Risk Evaluation Approach

Worker risk refers, specifically, to additional short term health risk posed to involved workers at a facility, collocated workers at a site, and the public by normal operations and possible operational accidents which may occur during facility construction, operation, or deconstruction. Key attributes of an effective worker risk evaluation include comprehensiveness (the capability of identifying all potential sources of risk), differentiability (the capability to prioritize hazards on the basis of potential risk significance), and quantifiability (the capability to bound the residual risk of health impacts as a function of operation scenario). Functional requirements for a consistent and comprehensive worker risk methodology include:

a comprehensive basis for identifying potentially significant accident hazards and accidental exposure scenarios for most major industrial and transportation activities involved in the management of wastes and in the remediation of contaminated sites, a capability to differentiate "residual"exposure hazards (which exist even after the implementation of administrative controls and personnel protection requirements from hazards to unaware and unprotected populations,

a quantitative basis for comparing risks of accidents and chronic exposures in DOE activities with those in other hazardous industries,

at least a qualitative basis for estimating the potential impact of additional investments in protection, training, and conduct of operations on estimated worker risks.

The effects of human exposure to ionizing radiation have been well documented in the DOE complexd. It has often been stated that our knowledge of radioactive material hazards exceeds that of most other sources of risk in the environment. Data on industrial accidents and hazardous chemicals is not as highly developed. Although industrial accident events are routinely reported, until recently it has not been feasible to estimate risks from industrial hazards and hazardous chemicals as a function of time of exposure with any degree of consistency. Recent efforts to improve the quality of available industrial risk information provide a basis for normalizing such information in worker risk evaluations.

TECHNICAL APPROACH AND ANALYSIS

Description of Site and Facilities

A set of four contaminated ponds at Oak Ridge National Laboratory contain sediments, sludges, and waters contaminated with radioactive and hazardous constituents. Although two of the impoundments are small and only lightly contaminated, two impoundments form a principal source of environmental risk at the site. One impoundment, in particular is not isolated from ground water and the continued integrity of its containment (a dike) is questionable.

Cleanup of the impoundments is a high priority commitment for the continuing cleanup program at ORNL. Since the impoundments represent the major source of contamination in that portion of the site, their cleanup would make render the local area stabilized and its facilities useful under some land use scenarios without further cleanup actions. Cleanup of the impoundments should be a part of a cost effective budget optimization strategy at ORNL.

Six alternative approaches have been developed for remediating the impoundments. Table I, taken from information in the RI/FS (1), summarizes some key attributes of each alternative. Table II summarizes, in a qualitative sense, some of the risk-based cost benefit characteristics for each alternative. Clearly, worker risk could be a key differentiator in the process of evaluating alternatives.

Table Ia

Table Ib

Table II

Scope of Worker Risk Review

The worker risk evaluation considers a full range of activities and hazards which could result in additional risk as a result of accidents and residual chronic exposures. Since persons other than workers can be impacted by risk producing events, the term worker risk is not strictly descriptive. However, it is used in here for consistency. Also, in some cases, design choices and technology to implement alternative remedial actions are not fully defined at this time. The worker risk evaluation is not intended to be a rigorous review of safety issues for each alternative. A rigorous safety assurance review consistent with DOE 5480.23 and attendant orders would be required for implementation of any of the alternatives.

The comprehensive set of activities which implement each alternative action roughly correspond to an operation which, to one extent or another, involves permanent modification of the local area. As a convenient nomenclature, therefore, the term facility describes each alternative. Total risk to workers and the public from facility operations is accrued over all phases of facility "life". For a typical DOE facility, this includes the periods of construction, operation, and deconstruction / decommissioning. For the facilities described in the six cleanup options, two separate phases of activity can be defined; 1) construction, and 2) implementation of long term monitoring and control measures.

For a worker risk evaluation it is useful to consider three sources of hazards, industrial accidents, exposure to radioactive and hazardous materials during normal operations, and exposure to radioactive or hazardous release accidents.

During the processes of facility construction and operation, personnel involved in the activities can be injured or killed in industrial accidents. Some of the activities would be typical of large civil construction efforts involving the movement of soils and rock, installing various kinds of structures and barriers, and replacement of materials. Other activities could involve pumping and treating of materials more akin to industrial or mining activities. Still other activities involve transportation of materials around the site, to adjacent government controlled areas, or to remote places.

Heavy construction activities pose risks mainly to involved workers. Pump and treat activities pose risks mainly to involved workers or collocated workers in nearby facilities. Transportation activities pose risks to involved workers (such as drivers), collocated workers within the ORNL, and the public along transportation routes.

During the processes of facility construction and operation, personnel involved in the activities will be exposed to radioactive and hazardous materials. DOE imposes a variety of radiological controls on its operations to reduce additional risks associated with exposure to

radioactive materials. These include designed in barriers, use of protective gear, and the imposition of exposure limits for personnel. DOE implements equivalent controls for hazardous materials. These controls effectively minimize the additional risk of radioactive exposure, but they do not eliminate it entirely. Expressed over a population of involved workers, exposure to radiation and hazardous materials will have the effect of increased likelihoods of cancer.

During the processes of facility construction and operation, personnel involved in the activities may be exposed to radioactive and hazardous materials as a result of radiological accidents. Examples of accidents include spills, fires, and natural catastrophes which result in failures of barriers and / or increased mobility for radioactive and hazardous materials.

Radiological health effects from accidents may result from direct exposure to gamma radiation or as a result of inhaling or ingesting various radioactive materials. Health impacts from exposure to hazardous materials can result from ingestion or inhalation. Due to the high radioactive inventories in some impoundments, it is conceivable that immediate of prompt fatal exposures could occur. However, due the lack of driving energies of release for the radioactive materials and the limited amounts of material being processed at any time provide no credible basis for early fatalities. Rather the types of exposures involved would produce an increased likelihood of cancers in an exposed population.

Accidents may be initiated by operational errors (such as operator errors), equipment failures, or natural events. A typical set of initiating events considered in SARs is shown in Table III. Due to the nature of the facilities used in the cleanup and the limited potential for operating events only a small portion of these initiating events are credible.

A subset of the six alternatives, defined in Tables I and II, are currently under active consideration for implementation and planning purposes. These alternatives 2,3, and 5 were chosen for a worker risk evaluation. Table IV provides a review of applicable sources of risk for each of the reviewed alternatives, based on descriptions of major activities associated with construction and long term monitoring of facilities defined in Ref. 1. Sources of credible risk in Table IV are reviewed with the methodology defined in the next section.

Table III

Table IV

Methodology and Analysis

This section provides methodology and analytical considerations for developing estimates of worker risk. Due to the preliminary nature of available design information on alternative cleanup facilities, the methods used are not rigorous in nature. However, methodology is sufficiently comprehensive to provide an understanding of the type of risks involved and their relative magnitude with respect to existing long term environmental risks.

Methodology and Analysis for Estimating Industrial Accident Risk

The potential for fatalities to result from industrial accidents can be evaluated using OSHA information on industrial activities similar to those defined for each of the cleanup alternatives. Activities involved in the alternative cleanup approaches include dirt and rock removal and replacement, construction of liners, caps and other structures, pumping and treating wastes, and transportation of materials. The Journal of Risk

Analysise provides an information base for evaluating industrial accidents during large scale remediation actions. The industrial accident data base includes fatality rates per hour of exposure for various types of involved worker descriptions (from OSHA sources) and estimates of the number of hours of exposure for a well defined remediation activity. Fatality rates in the industrial data review are used in this preliminary evaluation since they were reviewed for use in a similarly defined remediation effort. Exposure hours for various types of personnel were modified based on a comparison of the activities associated with each of the reviewed alternatives vs activities described for the alternatives in the industrial data review.

Estimating times of exposure based on comparisons with industrial accident data sources is given in Table V. Table V also summarizes an alternate approach to estimating exposure hours in which data used to construct initial cost estimates for each alternativef are used to define hours of exposure for various personnel types. Table V indicates that the exposure hours calculated with the different methods may differ significantly, those calculated using cost data being far more conservative.

Table V

Methodology and Analysis for Risk of Exposure to Radioactivity During Normal Operations

In a sense, DOE radiological controls and allowable doses are developed from the standpoint of restricting additional cancer risk for a population of workers to be less than one in ten thousand (1.0E-4). However, data available on the ORNL ER alternatives affords the opportunity to develop a better information source on radiological exposure, a source which can later be used for cost benefit evaluations of various enforced "burnout" levels.

For the evaluation of worker risk due normal exposure, it is assumed that workers are protected from the inhalation pathways through the use of respirators (possibly with the addition of protective clothing if particulate loadings or chemically hazardous materials are high enough to warrant it). It is also assumed that no protection against gamma radiation is afforded other than by controlling the time of exposure. A simple one dimensional exposure model assumes that:

worker doses from normal operations are a result of direct gamma radiation exposure,

most of the radioactive inventory is in currently in contaminated sediments at the bottom of the impoundments,

the major inventory of concern for direct gamma exposure is Cs137,

most of the exposure occurs during brief periods when a shallow layer of water is all that shields workers from contaminated sediments.

Table VI provides inventories of radioactive and chemically hazardous materials for each of the four impoundments. Exposure times for workers involved in less shielded activities are estimated using ORNL cost data. An estimate of the increased risk of cancer for the population of exposed personnel is estimated using commonly accepted dose conversion factors for involved workers such as ICRP (footnote e).

Table VI

Methodology and Analysis for Risk Due to Radioactive / Hazardous Chemical Release Accidents

A comprehensive systems assessment of potential radioactive release accidents was not feasible for this worker risk evaluation. Much of the

design information needed to conduct such an evaluation is not yet available, and the resulting accuracy is not needed for a preliminary comparison of alternatives. Most of the information for this worker risk evaluation has been extrapolated from reviews of safety analyses reports for similar facilities at other sites. This is not to imply that safety assurance evaluations for the impoundments need not be any more rigorous or thorough in its approach. One impoundment in particular qualifies as a Hazard Category 2 facility, and under the graded approach criteria in DOE STD 1027-93, all of the remediation alternatives would require a detailed safety assurance evaluation to meet DOE 5480.23 standards.

For facilities of the type being considered for remediation alternatives, several events can be identified which illustrate the potential for accidental exposures of involved workers, collocated workers, and the public to increased levels of radioactive or chemically hazardous materials:

a large facility fire due to an equipment failure, operational error, or other catastrophic occurrence in the system for pumping, dewatering, and treating waste slurries which results in release and dispersal of part or all of the available inventory;

a natural catastrophe such as a tornado or flood which directly affects the integrity of the SIOU and its facilities during a phase of construction when sediments can be disperse relatively easily;

an event in which long term institutional controls are ignored and a capped area or consolidation cell is penetrated.

The frequency of a major facility fire has been estimated using commercial nuclear industry and Department of Energy experience and an event sequence model for the frequency of a major release (Fig. 1). Table VII provides a summary of fire frequency data used in the evaluation of a major fire frequency. The frequency for a tornado in the local region is estimated from available weather data and an event sequence model for frequency of a major release (Fig. 2).

Fig. 1

Fig. 2

Table VII

A major flood , which overwhelms containments of the impoundments and causes a sudden release of water and sediments to local waterways is not considered part of the total risk associated with each of the projects, primarily because the vulnerability to flood induced release would exist with or without the remediation effort. However, since it is part of the spectrum of unanticipated events which can cause additional risk to the community, the frequency of a major flood induced release is estimated using an event sequence model (Fig. 3), assuming failure in a greater than 100 year flood.

The frequency of a failure of institutional controls is estimated using SAR data for a buried waste facility and an event sequence model for frequency of a major exposure to nearby personnel (Fig. 4).

Accident risk is estimated from accident frequency information using the following process:

It is assumed that the period of active construction is one year and that an accident which results in a release of nuclear material would occur during this time. The exception is an intrusion on buried waste. For intrusion on buried waste, the period in which the sediments pose a significant health hazard and the SIOU is a controlled access area, is used.

Dose to receptors is estimated using the amount of material available for release in an accident (Ia), and the fractional release of material at the point of the accident (Ra), to produce dose to an individual at the point of receptor (Di) in the algorithm

f[Ia * Ra] = Di

The functional relationship in the algorithm is estimated for the fire and tornado events using RSAC 5 to determine downwind dose at the receptor (in Rem). For the intrusion event, dose information is developed using occupational exposure information with elimination of shielding assumed.

Risk is estimated by estimating dose to an exposed population Pe from the relationship

Di * Pe = Dp

and a conversion between dose and likelihood of cancer induced fatality (5E-4 fatality / Rem for exposed members of the public, and 4E-4 for exposed workers). For the flood event, dose is inferred from No Action risk data summarized in Ref. 1, with equivalent doses estimated using dose conversion factors for public exposure.

Fig. 3

Fig. 4

Results of Worker Risk Evaluation

Results of the worker risk evaluation are developed using the information and assumptions of Section II. These results are summarized in the following sections. In order to generate a consistent comparison, the results of Ref. 1 are interpreted in a fashion which can be compared with worker risk information.

Industrial Accident Risk

Table VIII provides a summary of the risk of industrial accidents to workers, collocated workers, and the public for the three alternative remediation projects. Since vulnerability of the public is limited to conditions where offsite transportation occurs and none of the alternatives involve offsite transportation, all industrial accident risk given in Table VIII is to accrued by workers involved in the remediation efforts. Table VIII estimates are made using costing data for the ER alternatives (3) and fatality frequency data (2).

Table VIII

Risk of Occupational Radiation Exposure During Normal Operations

Risk to involved workers from occupational exposure to radiation during the construction process is estimated in Table IX. Table IX provides estimates of dose to involved workers from gamma radiation sources during periods where shielding may be minimal. These estimates utilize dose rates from one dimensional exposure models using a hypothetical worker performing tasks at the closest point to the sediment layer. These estimates also use exposure time data derived from remediation alternative cost information (3).

Table IX

Risk Due to Accidents Involving Release of Nuclear of Chemically Hazardous Materials

Risk from accidents involving a release of radioactive materials is estimated in Tables X, Xa and Xb. Table X provides a base case calculation using best information derived from all sources. Table Xa provides a sensitivity calculation appropriate for estimating the nuclear accident contribution from Alternative 2. Table Xb provides similar information for Alternatives 3 and 5. Notably, in Tables Xa and Xb, flood

caused release is not summarized as part of the accident contribution to worker risk.
Table X
Table Xa
Table Xb
Achievable and Net Risk Reduction
Table XIa summarizes in the potential reduction in risk achieved by applying the risk data in Ref. 1 to a population of persons along each of the nearby waterways. Notably, an assumption is made that a specific percent % of the population (1% in the base case) is exposed to radioactive contamination over a lifetime.
Table XIb integrates information from the worker risk evaluation with information on reducible risk in Table XIa. Results are presented as a net risk reduction which could be expected if each of the subset of Ref. 1 alternatives were selected for implementation.
Table XIa
Table XIb
Table XIc
Table XId
CONCLUSIONS AND RECOMMENDATIONS
Table XIb provides a comparison between risk estimates for worker risk and risk estimates for achievable reductions in risk, establishing a net expected risk reduction for each ER alternative. Results of the worker risk evaluation are as follows:

The sum of worker risks are small when compared with the achievable reduction in risk which can be made by implementing each of the proposed alternatives. In all cases the expected differential is an order of magnitude or more.

The net benefit associated with each of the alternatives is equal,. Thus the choice of which alternative to implement can only be made on the basis of expected cost or other factors such as uncertainty in cost, stakeholder preference, and land use issues.

Several recommendations, if implemented, would improve, validate, and extend the results of this preliminary worker risk evaluation.

Risk information from the remedial investigation should be systematically reviewed with less bounding assumptions regarding the level of risk posed by existing conditions. A systematic appraisal of populations potentially affected by leakage from existing impoundments should also be performed. The sensitivity data in Table XIc and XId indicated that a smaller estimate of population exposed to the risk of slowly leaking contamination would dramatically affect the overall value of the cleanup as well as the relative choice of alternatives.

Involved worker risk, particularly accident risk should be more systematically evaluated to assure comprehensiveness of the accident scenario identification process (This information would not be wasted since SARs and other documents using the same information base could be generated in a more cost effective manner at a later date). Also, realistic assumptions used to assess the potential for releases from energetic accidents, and the potential for populations of persons to be exposed to accidental releases. Finally, hazardous chemical releases were not estimated for this report since the makeup of impoundment inventories and postulated release scenarios did not indicate a significant potential for generating a health effect producing release. This assumption should be verified.

Net risk information compiled in this study should be integrated with cost information and other factors which could affect the overall cost benefit of each alternative using uncertainty distributions and an appropriate propagation technique.

15-30

INTEGRATED WASTEWATER MANAGEMENT PLANNING FOR DOE'S ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE

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ABSTRACT

Rocky Mountain Remediation Services, L.L.C. (RMRS), jointly formed by Morrison Knudsen Corporation and BNFL Inc., provides international experience in the nuclear, environmental, waste management, decontamination and decommissioning (D&D) , and project management industry. The company is currently the environmental restoration, waste management, and D&D subcontractor for Kaiser-Hill Company at the Rocky Flats Environmental Technology Site (RFETS). RMRS offers unique solutions and state-of-the-art technology to assist in resolving the issues that face industries today.

RMRS has been working on methods to improve cost savings recognized at RFETS, through application of unique technologies and process engineering. RMRS prepared and is implementing a strategy that focused on identifying an approach to improve cost savings in current wastewater treatment systems and to define a low-cost, safe and versatile wastewater treatment system for the future. Development of this strategy was targeted by Department of Energy (DOE) Headquarters, DOE Rocky Flats Field Office and Kaiser-Hill as a "Project Breakthrough" where old concepts were thrown out the door and the project goals and objectives were developed from the ground up. The objectives of the strategy developed in a project break through session with DOE included lower lifecycle costs, shutdown of one of two buildings at RFETS, Building 374 or Building 774, reduced government capital investment, and support of site closure program goals, identified as the site's Accelerated Site Action Plan (ASAP). The recommended option allows for removal of water treatment functions from Building 374, the existing process wastewater treatment facility. This option affords the lowest capital cost, lowest unit operating cost, lowest technical management risk, greatest support of ASAP phasing and provides the greatest flexibility for design with unforeseen future needs.

The recommended alternative provides for substantial near-term cost and technological advantages over the present operating baseline and planned capital improvement program. The total estimated capital expenditures for the recommended alternative is $6.8 million which is considerably less than the current capital funding level of approximately $65 million for

full upgrades to Building 374. In addition, the recommended alternative saves approximately $6.1 million per year in operating costs. Accelerated program implementation will produce the desired improvements as specified by the key objectives, and will release capital and operating funds for investment in the site's higher risk reduction activities, supporting ASAP programs.

RMRS and Kaiser-Hill recommended, and DOE concurred, that as a result of the project's low Net Present Value and financial and technical rate of return, the preferred recommendation be implemented through a single consolidated project. A single consolidated project will allow for direct focus across multiple functional programs (i.e., Operations, Permitting, Environmental, etc.) assuring schedule and cost compliance. RMRS also recommended that the Project Implementation Plan be prepared to support Fixed Price and Fixed Unit Price contracting terms as a means of assuring the following:

- Accelerated schedule implementation;
- Competitive project cost;
- Government/commercial risk sharing; and
- Reduced government capital investment.

BACKGROUND

RFETS is a government-owned, contractor-operated facility which is a part of the nationwide DOE nuclear weapons production complex. Prior to 1989, the primary mission of the site was the continual production of components for nuclear weapons. Production activities included metalworking, fabrication and component assembly, plutonium recovery and purification, and associated quality control functions ensuring the technical performance of the weapons' components. The plant was built in 1951 and began operations in 1952. In 1989, as a result of a changing international political climate, the decision was made by the United States government to discontinue production of components for nuclear weapons at Rocky Flats. Rocky Flats has undergone a transition from a weapons production facility to an environmental restoration and waste management site. The current mission of the site is to manage waste and material, clean up and convert RFETS to beneficial use in a manner that is safe, environmentally and socially responsible, physically secure and cost-effective.

The need for an integrated waste water management strategy was driven by the following:

- the need to reduce routine facility operating costs, to provide financing of risk reduction activities, and to provide support to site closure activities requiring accelerated waste water treatment to support deactivation and decommissioning activities over a 10-year period.
- major near-term facility improvements would be necessary to keep existing wastewater treatment facilities operational to support deactivation and decommissioning activities.
- negotiations on regulatory relief from overly restrictive stream standards and defining necessary and sufficient standards will impact the requirements for waste water treatment.
- changes in site mission from a weapons production mission to waste management, environmental clean-up and conversion to beneficial use have dramatically shifted the requirements for wastewater treatment.

PURPOSE AND OBJECTIVES

RFETS will continue to produce diverse wastewater streams as it completes its mission of environmental restoration, D&D, and waste management. This

study included the integration of previous wastewater management strategies into one overall strategy and the provision for cost-effective treatment of all wastewater to be produced at the site. Alternatives developed in the study were designed to support the ASAP site closure concept.

The primary objective of this document was to develop and document the basis of this strategy and to develop short- and long-term implementation plans. To achieve the primary objective, the following supporting objectives were identified.

The strategy must integrate multiple focused wastewater management strategies already in place or in preparation into one overall strategy.

The strategy must evaluate the routing and treatment of wastewater streams based on composition and regulatory requirements rather than the point of generation.

The strategy must ensure that adequate capacity is provided for all wastewater treatment over the foreseeable future.

The strategy will be integrated into the RFETS Water Management Plan.

The strategy must contain information on the identity and characteristics of all known wastewater sources and conveyance methods on the site. This will allow for identification of waste segregation and minimization opportunities. In addition, impacts of wastewater stream elimination on the balance of the wastewater to be treated can be evaluated.

The strategy must identify which wastewater treatment facilities currently in operation can be cost-effective components in an overall strategy.

Development of the strategy considered all current and anticipated sources of wastewater potentially requiring treatment. This included wastewater from domestic use, building process operations, facility deactivation, facility decontamination and decommissioning, and environmental restoration. Excluding domestic wastewater, the largest sources of wastewater in terms of average annual volume projected for the future include the Interceptor Trench System (ITS) (3.5 million gallons per year), the Building 566 laundry (1.3 million gallons per year), and environmental restoration activities, primarily groundwater (up to 6.5 million gallons per year). The actual volume of environmental restoration water could decrease dramatically depending upon final agreements on cleanup levels. Facility deactivation will also produce the most highly contaminated wastewater, although volumes will be low in comparison to the total of the other major sources (greater than 1 million gallons per year). Deactivation wastewater production will also peak fairly rapidly and then begin to decline, and will only be produced over the next one to five years. Characteristics and estimated volumes of future process wastewater sources are shown in Fig. 1.

Fig. 1

Development of the strategy also considered the capacities and capabilities of existing treatment facilities at Building 374, Building 774, Building 995 (the Sanitary Treatment Plant (STP)), and the Site Treatment Facility (treatment of water generated from environmental restoration activities).

ALTERNATIVES EVALUATED

Four alternatives were identified for wastewater treatment. These alternatives support achievement of the ASAP site closure goal, and include the following:

Alternative 1 - Minimum Building 374 upgrades;
Alternative 2 - Building 374 Liquid Waste Treatment Facility upgrades;
Alternative 3 - Building 374 Waste system Evaporator upgrades; and
Alternative 4 - Building 374 Elimination

These alternatives were subjected to a technical and cost effectiveness evaluation and a sensitivity analysis, and a final selection made. Alternative 4 (see schematics in Figs. 2 and 3) was selected as the recommended alternative for the following reasons:

It is the only alternative capable of supporting an early closure of Building 374 and can also support closure of building 771/774,

It has the lowest overall life cycle cost of dollars and capital cost dollars of the ten-year alternatives,

It has low risk of delay in implementation because of relatively low capital funding requirements;

It is ranked high technically due primarily to minimization of waste and overall flexibility in addressing changes in wastewater characteristics, and

It is the best alternative to support achievement of the ASAP closure goals.

Fig. 2

Fig. 3

The estimated costs of the recommended alternative is summarized below:

DOE Capital Investment $6,800,000

Average O&M Costs$8,100,000

These costs demonstrate the following savings over current operations:

The total estimated capital expenditures for the recommended alternatives is $6.8 million, which is considerably lower than the current capital funding level of approximately $65 million for full upgrades to Building 374.

The current operating budget of Building 374 is approximately $9.3 million. If operations of the Sitewide Treatment Facility and waste disposal are added, this cost increases to approximately $14.2 million. The recommended alternative saves approximately $6.1 million.

An overall life cycle cost curve for the recommended alternative is illustrated in Fig. 4.

Fig. 4

IMPLEMENTATION

Implementation of the strategy was approved by DOE in November, 1995 and alternative treatment technologies have been evaluated and a Conceptual Design Report subsequently prepared. The conceptual design includes the following features.

Building 374 will be closed in FY 97 and a new temporary treatment facility (TTF) will be designed and constructed to treat wastewater from building operations, deactivation, and decontamination and decommissioning. This facility will be located near Building 374 to take advantage of existing collection and support systems. Either leased or modular equipment will be utilized in the facility as the operational life of the facility will be less than ten years.

Regulatory relief has been sought on the existing plutonium standard for discharges from the TTF facility. This relief is based on raising the plutonium level from the current site specific standard to the Statewide plutonium standard for up to five years while tanks are drained and pipelines are flushed during initial D&D activities. Review of historic records shows that the proposed increase in stream loading would only

double the loading over levels discharged in the last five years but would not be a risk to human health or the environment. This temporary modification, currently being negotiated with the regulatory agencies would save approximately $73 million in capital and operating costs which could be redirected to higher priority risk reduction activities.

Liquids produced from deactivation activities in Building 371 will be treated for initial reduction of radionuclide and metal concentrations in the caustic waste treatment system to be installed in Building 371. Treated effluent from this process could then be treated in the Building 774 carrier precipitation process for additional radionuclide removal. The supernatant wastewater from both Buildings 371 and 774 will then be treated for further reduction of radionuclides and metals in the new temporary facility replacing Building 374.

The general approach to handling of deactivation wastewater is shown schematically on Fig. 3. Liquids produced from deactivation activities in Building 771 will be treated for initial reduction of radionuclide and metal concentrations in the oxalate precipitation process and hydroxide precipitation process located in Building 771. Additional treatment of the effluent from these processes plus other miscellaneous liquids produced in Building 771 and 774 will be treated in the carrier precipitation process in Building 774.

A temporary sludge immobilization system (TSIS) is being designed to treat sludges currently stored in Building 374 and Building 774, and for sludges produced by the Building 774 carrier precipitation process. TSIS is a mobile system that can be reassembled elsewhere on-site or offsite to treat other sludges or waste forms.

Regulatory relief on nitrate and uranium limits has been sought to allow for direct discharge of ITS wastewater. This relief is based on removing the water supply use classification from Walnut Creek but leaving the agricultural use classification, thereby allowing for compliance with nitrate standards (based on no risk to human health or the environment for nitrates) and use of statistical methods to prove that uranium levels were below background concentrations. These actions are estimated to save $20 million dollars over a ten year period. Recent meetings with regulatory agencies and Stakeholders on regulatory relief have been positive and it appears that all parties will now request the Colorado Water Quality Control Commission to revise the stream standards.

Characterization of Building 566 laundry wastewater has been conducted to verify that discharge to the STP can take place and this stream has subsequently been eliminated from Building 374.

RMRS has worked with DOE and Kaiser-Hill to obtain variances from DOE Orders and Plant Standards in the application of necessary and sufficient standards in engineering, installation and operation, thereby producing substantial capital and operating cost reduction while allowing for commercial equivalent practices.

CONTRACTING/FINANCING APPROACH

The Kaiser-Hill team has committed to performance-based contracting with an evolution toward commercialized fixed-price contracting. In addition, the ASAP necessitates a projectized approach to providing improved near-term treatment services in conjunction with lower building and routine operating costs, supporting funding of high priority risk reduction activities. As a result, RMRS looked at several options to expedite the contracting approach, including using commercialized contracting and both cost plus fixed fee and fixed unit price contracting strategies.

A typical cost plus fixed fee approach would require the government to pay for design and installation of the facility and capitalization of the equipment without an assurance that the plant would perform. In addition, the government would own the building, and with the approach proposed by ASAP, this is in contradiction to taking the site down.
The approach that RMRS proposes is a fixed price services contract, whereby RMRS provides equipment and the services associated with that equipment.

FUTURE PLANNING AND INTEGRATION

The schedule for implementation of this strategy for the recommended alternative is presented in Fig. 5.
Fig. 5

CONCLUSIONS

Kaiser-Hill and RMRS have committed to accelerated closure of RFETS buildings as a means of substantial cost savings to the government. Implementation of the Integrated Wastewater Management Strategy is in keeping with this philosophy as it allows for early closure of Building 374, thereby allowing funds to be reallocated for other site closure activities. Assuming that accelerated funding is available and a design-build approach to procurement approved, Building 374 can be closed in FY 1997. The driver for this date is the completion of the design and construction of the new facilities. Given the services contract approach, an acceleration of 12 to 24 months is anticipated./
Lessons learned from this DOE breakthrough project are applicable to other DOE and DOD facilities.
Revisiting existing baseline operations may show that substantial near-term cost savings can be realized and these funds can then be redirected for other activities. Alternate contracting strategies may also reduce required government capital investment and lead to sharing of risk between the government and contractors. There is also a need to reassess existing cleanup levels to see if there is a good risk basis for negotiating changes to standards to reduce operating costs.

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GAS SAMPLING AND ANALYSIS OF 55-GALLON DRUMS CONTAINING TRANSURANIC WASTE USING COMMERCIAL VENDORS AND EXISTING TECHNOLOGY

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ABSTRACT

The U.S. Department of Energy National Transuranic Program has provided funding to the Oak Ridge National Laboratory (ORNL) to use a private vendor to gas sample and analyze for hazardous constituents regulated under the Resource Conservation and Recovery Act (RCRA) and other gases in some of their existing contact-handled (CH) transuranic (TRU) waste inventory. The project is intended to demonstrate the feasibility of using private industry and existing mobile Fourier Transform Infrared

Spectroscopy technology for gas analysis. The S.M. Stoller Corporation and the Entropy Corporation have teamed together to propose the use of an existing mobile laboratory technique that is accurate, rapid and that provides the advantage of "at-line" analysis on site to the ORNL. The two companies propose to sample and analyze the headspace gas from 500 drums at ORNL that contain CH TRU waste. Currently, there is no contract awarded to the Stoller/Entropy team from ORNL to participate in this project.

INTRODUCTION

The U.S. Department of Energy (DOE) has constructed the Waste Isolation Pilot Plant (WIPP), a deep geological repository, in southeastern New Mexico as a research and development facility intended to demonstrate methods for the safe disposal of transuranic (TRU) wastes generated from defense-related programs. TRU waste is defined in DOE Order 5820.2A as ". . . waste that is contaminated with alpha-emitting transuranium radionuclides with half-lives greater than 20 years and concentrations greater than 100 nCi/gram of waste at the time of assay" (1). A transuranium radionuclide is further defined in DOE Order 5820.2A as any element having an atomic number greater than 92. TRUwastes may be classified as contact-handled (CH) or remote-handled (RH), depending on surface radiation levels. TRU waste packages having an external surface dose rate of less than 200 mrem/h are classified as CH TRU waste, while those greater than or equal to 200 mrem/h are classified as RH TRU waste. The U.S. Environmental Protection Agency (EPA) granted WIPP a conditional no-migration determination (NMD) that requires DOE to characterize nonradioactive hazardous constituents and other gases contained in TRU waste, prior to delivery to the WIPP site (2). These constituents, regulated under the Resource Conservation and Recovery Act (RCRA) codified under Part 40 of the Code of Federal Regulations Section 268 (40 CFR 268), can be characterized by obtaining a gas sample from the headspace found under the lids of 55-gal drums containing TRU waste. Adequate characterization data must be collected from all DOE sites on their TRU waste intended for disposal at WIPP.

As part of the DOE program to promote technology development, demonstration, and utilization of private industry, this project will implement existing technology used by private vendors and adapt the methodology to fit the needs of the TRU Waste Characterization Program at DOE. This project will demonstrate that a private vendor can perform gas sampling and analysis activities using Fourier Transform Infrared Spectroscopy (FTIR) as the analytical technology. Prior to demonstration of the technology, a template of documents required to perform this operation at DOE sites will be developed for the Carlbad Area Office National TRU Program (3). Objectives of the FTIR gas sampling and analysis project are as follows:

- Develop a gas sample analysis project at the Oak Ridge National Laboratory (ORNL) that would provide the National TRU Program and the TRU Waste Characterization Program with an "Eastern Site" capability to analyze gas samples in accordance with the WIPP TRU Waste Quality Assurance Program Plan (QAPP) (3);
- Obtain TRU waste characterization data for the Department of Energy Carlsbad Area Office to assist in preparation of the WIPP Compliance Package (2);
- Demonstrate rapid and cost-effective gas sampling and analysis of DOE TRU waste headspace gas using a mobile FTIR laboratory; and

Demonstrate private sector capability to support characterization of TRU waste.

FOURIER TRANSFORM INFRARED SPECTROSCOPY TECHNOLOGY

FTIR is an analytical technique used in research and analytical laboratories. In recent years, FTIR has been applied to process control, emission measurement, and gas analysis applications. Most gas molecules absorb infrared radiation in recognizable and quantifiable pattern. Therefore, it is possible to measure multiple compounds simultaneously in gas samples because of the resolution and spectral range of FTIR spectrometer systems (4).

Currently, analyses are performed by gas chromatography and gas chromatography/mass spectrometry, which require a gas sample collection device and an analytical laboratory and, consequently, delay determination of the gas concentrations. The ability to do "at-line" analysis would reduce the necessary sample handling and the cost of analysis of each gas sample.

Parallel efforts have been conducted by two entities of the U.S. government for different applications of FTIR technology. The EPA has funded research in the detection and monitoring of hazardous air pollutants (HAPs) regulated under the Title III of the 1990 Clear Air Act Amendments at stationary source emissions, and the DOE has funded research in the development and evaluation of methods for the analysis of RCRA constituents in the gaseous headspace of TRU waste drums.

Since 1989, Entropy's Research Division has been developing Extractive FTIR spectroscopy technology for Air Emissions Testing in research funded by EPA. The method involves transporting the gas sample of interest to a gas cell for analysis. Entropy has demonstrated FTIR accuracy and precision by performing EPA Method 301Validations for over 30 HAPs (4). Since 1992, Entropy has developed a library of FTIR reference spectra for over 100 HAPs, FTIR specific QA/QC procedures as detailed in the EPA-accepted FTIR Protocol Document, and a fully-equipped mobile FTIR laboratory. Field testing using a mobile FTIR laboratory to analyze various source categories of on-site emissions has been performed at numerous industrial sites since May 1992.

The NMD requires that a representative waste drum headspace gas sample be collected for analysis. The Idaho National Engineering Laboratory (INEL) has been involved in the development and evaluation of methods for the analysis of RCRA constituents in the gaseous head space of CH TRU waste drums. INEL initiated a two-phase study (5) to assess the feasibility of using FTIR for the analysis of volatile organic compounds (VOCs) in waste drum headspace in fiscal year 1993.

PROJECT ACTIVITIES

Much of the CH TRU waste stored at ORNL is suspected to contain hazardous VOCs regulated under RCRA that must be characterized through container headspace gas sampling and analysis. ORNL, however, has no capability to conduct gas sampling and analysis of TRU waste drums. The S.M. Stoller Corporation and Entropy Incorporated have teamed together to propose the use of an existing mobile laboratory technique that is accurate, rapid, and provides the advantage of real-time analysis on site to ORNL. The two companies propose to sample and analyze the headspace gas from 500 drums at ORNL that contain CH TRU waste. In the teaming agreement, Stoller will act as the primary subcontractor to ORNL, and Entropy will be a subcontractor to Stoller, when a contract is awarded by ORNL. The team, referred to as Stoller/Entropy, proposes to act as a subcontract to

Lockheed Martin Energy Systems, inc., (LMES) the prime management and operations (M&O) contractor for ORNL. LMES operates ORNL for the Department of Energy Oak Ridge Operations Office (DOE-ORO).
Entropy Incorporated was previously contracted by the EPA to provide research and development for the sampling and analysis of stack emissions. Entropy has developed quantitative gas phase FTIR techniques and has created a validated measurement method accepted by the EPA. The Stoller Corporation provides technical consultation services to DOE field offices and DOE generator/storage site M&O contractors in the area of radioactive waste management.
This project will be limited to gas sampling and analyzing for RCRA regulated constituents and other gases in relatively low-risk drums. Low-risk drums are defined as having very low surface dose rates, typically less than 50 millirem/hour, that will not require venting (because a vent filter is already installed).
Utilization of low-risk drums will eliminate the need for an explosion-proof vessel required to install a filter in unvented drums that may contain a flammable mixture of gases. The project will be conducted in three phases. Phase I will consist of the development of documentation and modification of existing equipment. Phase II will consist of transporting the FTIR laboratory to the site and conducting the gas sample and analysis activities. Phase III will consist of project closeout, writing a final report with lessons learned, and submitting a template of documents to DOE.
Strategy
DOE-ORO and LMES will oversee all activities associated with the FTIR Gas Sampling and Analysis Demonstration Project. LMES will also provide support by assisting in the site selection, participating in the Operational Readiness Process and document review activities, and retrieving TRU waste drums to provide adequate throughput during gas sampling and analysis operations. Stoller/Entropy will conduct required preparation activities, perform the gas sampling and analysis, and conclude the project in accordance with the project management plan.
Within the subcontractor teaming agreement, Stoller will prepare all required documentation necessary to gain access to ORNL. Major Stoller responsibilities will include developing all the documentation needed to satisfy environmental, safety, health, operations, quality assurance, and project management requirements and leading the performance of the Operational Readiness Process. Entropy will furnish their existing FTIR mobile gas sampling and analysis laboratory, work with Stoller to complete equipment modifications and additions, and develop maintenance plans and procedures to adapt the laboratory for TRU waste application. Entropy will also provide any technical and operations support needed to apply the FTIR technology properly.
Phase I Activities
The first task will be to complete all the documentation required to gain access to the ORNL site. Phase I will "pave the road" for other private vendors requesting to characterize or process TRU waste on DOE sites. The documents prepared will provide DOE with a "template" of typical issues and concerns at DOE operating sites. It will also cover document preparation and the steps to be followed by private vendors.
The Fiscal Year 1996 activities will include developing documentation, conducting a Operational Readiness Process, and completing the needed equipment modifications required to allow a commercial vendor access to

the ORNL site. These activities initially will involve the identification of a suitable location on the ORNL site where the sampling activities will be conducted. Site selection will be based on interviews with appropriate Waste Management and Operations personnel and the assessment of applicable information of the available sites. The site selection will then be incorporated into the gas sampling program management plan, which will provide information on the cost and schedule of the activities associated with this project.

Plans to be developed that implement specific requirements and provide guidance on the conduct of operations include a Gas Sampling and Analysis Quality Assurance Project Plan, a Health and Safety Plan, a Project Management Plan, a Waste Management Plan, and a Sampling and Analysis Plan. Standard Operating Procedures, Testing and Calibration Procedures, and Emergency Procedures will also be developed. The procedures currently used by Entropy for emissions testing will require revision to implement the requirements specified for the FTIR Headspace Method and the quality assurance/quality control specifications in the QAPP and approved by the EPA Office of Solid Waste.

Activities conducted in parallel to document development will include a Safety Evaluation that involves review of the selected facility safety documentation to determine whether an unreviewed safety question exists. In addition, the Operational Readiness Process will have to be conducted and involve Entropy, Stoller, ORNL Waste Management, and DOE Oak Ridge. Because Entropy's mobile FTIR laboratory was originally constructed to provide FTIR emissions testing at industrial facilities under an EPA protocol, some minor modifications and additions are required to allow gas sampling of TRU wastes packaged in drums. These changes are described below.

The FTIR Method Development will include the revision of existing procedures or the development of new procedures to address the requirements for calibration and performance testing. Analytical software will be obtained to ensure consistency with the INEL method used to develop an automated analysis system for headspace gases. Finally, reference spectra obtained by the INEL will be obtained and assessed for application to Entropy-owned equipment.

The modified mobile FTIR laboratory will contain all the necessary equipment for on-site measurements, including sampling and sample conditioning systems, on-board diesel electrical generators, a climate control system, an air compressor, and a purge air generator. The gas sampling equipment is mounted on an on-board table that is specially designed to dampen vibrations. The detection equipment will include an in-line FTIR spectrometer for measuring the concentration of various VOCs and a residual gas analyzer (RGA) for measuring hydrogen gas concentration. Other peripheral sampling equipment will include a vacuum pump for drawing samples, a downstream constant air monitor (CAM) for alpha radiation detection, a SUMMA canister for collecting test samples, interconnecting sample lines, and a personal computer system for data collection, analysis, and reporting.

FTIR Equipment Modification will require the purchase and installation of a low-volume FTIR gas cell. The smaller cell is required because of the smaller volumes of gases collected from a 55-gal drum as compared to emissions collected from stacks. Due to the limitations of the FTIR in identifying and quantifying homonuclear diatomic gases such as hydrogen, an RGA will have to be installed in the mobile laboratory. Data

acquisition software will also be developed to allow logging both FTIR and RGA data together.

Phase II Activities

The modified mobile FTIR laboratory will be transported to the sampling location, which has been tentatively identified as an asphalt pad just adjacent to Building 7879. To support the operation, a temporary shelter will be erected to provide a weather-protected area for temporarily staging the TRU waste drums that will be sampled. The drums will be retrieved from Building 7879 by LMES personnel and placed in the staging enclosure at the beginning of each work day. Stoller/Entropy plans to sample 12 drums per day and to maintain no additional waste in the staging area at any time. At the close of each work day, LMES will return all TRU waste drums from the staging area to their designated storage location in Building 7879.

Before beginning the gas sampling procedure, a TRU waste drum is interfaced to the glovebox to seal off the sample area and prevent the potential release of radioactive particulate to the environment. The glovebox (Fig. 1) is equipped with a sample-line penetration containing a permanently installed high-efficiency particulate air (HEPA) filter, two glove ports, a transfer port, and a separate HEPA filter vent. The sampling technician places the required tools (an adjustable wrench and new filter) inside the glovebox using the installed transfer port. The drum and glovebox assembly are positioned adjacent to the mobile FTIR laboratory and linked together with a detachable sample line. The sampling technician will gain access to the drum headspace via the carbon-composite vent filter installed in the drum lid. This is accomplished by the following activities.

A hole is drilled through the stainless steel cover of the filter using the permanently mounted drill press in the glovebox.

The sample probe (Fig. 2) is inserted through the underlying carbon-composite filter media.

A sample is subsequently drawn by vacuum into the truck-mounted FTIR equipment for real-time analysis.

Fig. 1

Fig. 2

Following sample analysis, the probe is retracted from the drum internal headspace. Then the filter is removed and replaced.

The attendant health physics technician (HPT) smears the old drum filter, drill bit, sample probe, drum lid, and wrench for possible contamination. Provided these are radiologically clean, the drum is removed from the glovebox and returned to the staging area.

In the unlikely event of contamination, sampling activities will be terminated and surveys will be conducted to determine the extent of the contamination spread. Trained personnel and procedures will be in place to take the necessary actions to mitigate this or other potential off-normal events.

A second technician will operate the FTIR sampling equipment as the sample is drawn and analyzed as described below. The FTIR and RGA sampling equipment is fully automated and operated by personal computer via a programmable controller.

The technician will purge the system using nitrogen, to ensure that no moisture or gas contaminants are present.

Prior to extracting the headspace gas sample, the technician will evacuate the sampling equipment and verify that the system is leak-tight.

The sample is then drawn into the FTIR and RGA modules and analyzed for 29 different VOCs and for hydrogen concentration.

Following the analysis, a data package is saved on computer diskette and a hard copy is printed.

Several radiological controls will be implemented during the entire TRU waste gas sampling process. These include the following:

Stoller/Entropy will verify that the inner waste contents of each drum sampled are adequately bagged and sealed by reviewing previously recorded real-time radiography videotapes.

Stoller/Entropy will use a sealed glovebox during the sampling operation and will survey all tools, filters, and other materials before they are removed from the glovebox environment.

Stoller/Entropy will install a HEPA filter at the glovebox sample inlet and will use a CAM at the outlet of the sample line to monitor for potential releases.

Stoller/Entropy will provide project funding through LMES to support full-time HPT coverage during drum retrieval and sampling activities and will ensure that all operating technicians have received LevelI Radiation Worker Training and respirator fit tests.

Stoller/Entropy will operate with ORNL-approved procedures only and will comply with all applicable site-specific radiological controls. Stoller/Entropy will also conduct an As-Low-As-Reasonably-Achievable (ALARA) Evaluation. All candidate drums will be surveyed before sampling to ensure that ALARA goals will not be exceeded and that personnel exposure to penetrating radiation will be minimized. Drums with higher surface radiation levels that might result in exposures in excess of ALARA goals will be sampled on a case-by-case basis only and special precautionary measures will be used.

Phase III Activities

The activities in this phase will include closing out the project and compiling a final report. The final report will include lessons learned and document templates. The templates will provide general guidance as to the information to be included for implementation at other DOE generator/storage sites.

CONCLUSION

In summary, funding has been provided by the DOE National TRU Program to demonstrate the feasibility of sampling and analyzing the headspace gas for RCRA-hazardous constituents and other gases from 55-gal drums containing CH TRU waste. The project utilizes an innovative approach combining the use of private vendors, mobile technology, and FTIR technology for analyzing gases regulated under RCRA. The mobile FTIR laboratory will require minimum modifications to the equipment thereby reducing costs that are associated with the purchase of a new system. The proposed project may grant the Stoller/Entropy team a subcontract with LMES to conduct the gas sample and analysis activities on 500 of the vented 55-gal drums containing CH TRU waste at ORNL by the end of fiscal year 1996.

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QUANTUM-CEPTM TECHNOLOGY APPLICATION TO DOE MIXED WASTES

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ABSTRACT

Quantum-Catalytic Extraction Processing (Quantum-CEP) uses a molten metal bath, as both a solvent and catalyst, to convert feeds to their elemental constituents, and through select reactant addition, form valuable products. Work with U.S. Department of Energy's low level radioactive and mixed waste streams has verified the ability of Quantum-CEP to dissolve the wastes and partition the radionuclide contaminants to target phases of the system and to destroy any organic contaminants in the waste, converting them to valuable synthesis gas.

M4 Environmenta, LP, a limited partnership between Lockheed Martin Corp. And Molten Metal Technology, Inc. Has built and commissioned a small-scale processing/demonstration facility in Oak Ridge, Tenn., which has begun processing DOE mixed wastes in a privatized mode. This paper, in addition to discussing the technical achievements of Quantum-CEP, reviews the design, technical capabilities, and initial processing data of this first-of-a-kind commercial facility.

INTRODUCTION

Quantum-Catalytic Extraction Processing (Quantum-CEP) uses a molten metal bath, as both a solvent and catalyst, to convert feeds to their elemental constituents, and through select reactant addition, form valuable products. Tests have been carried out to demonstrate the applicability of Quantum-CEP to the processing of the U.S. Department of Energy's low level radioactive and mixed waste streams. The three-year project has focused on contaminated scrap metal, and includes optimization for liquids, soils, sludges, combustible and inorganic debris. Specific program objectives included demonstrated recycling of ferrous and non-ferrous metals, immobilization of radionuclides, destruction of hazardous organics, recovery of volatile metals, and conceptual design of a Quantum-CEP facility for contaminated scrap metal in the DOE inventory. Experiments have verified the ability of Quantum-CEP to dissolve the wastes and partition the radionuclide contaminants to targeted phases of the system (the durable ceramic phase in the case of uranium) and to

destroy any organic contaminants in the waste, converting them to valuable synthesis gas. Using EPA Method TO-14, principal organic hazardous constituents (POHCs) were not detected to below ppb levels in the synthesis gas product corresponding to destruction removal efficiencies (DREs) 99.9999%. NOx and SOx were not detected at lower detection limits of 3 and 1 ppm respectively. Dioxins and furans were non-detectable at the targeted regulatory standard of 0.1 ng/Nm3 TEQ using a modified EPA Method 23. POHCs were non- detectable in any of the condensed phase products at several orders of magnitude below EPA land disposal restriction limits (EPA Methods 8260 and 8270). Simultaneous to this conversion efficiency, >90% of the feed was recovered as commercially-viable products as certified by the Massachusetts Department of Environmental Protection for a broad range of inorganic, organo-metallic, and chlorinated organic waste feeds. All condensed phase products were shown to pass EPA toxic characteristic leach procedure (TCLP) and synthesized glasses containing high concentrations of halogens (e.g., up to 10% Cl) surpassed high level nuclear wastes in PCT analyses. In parallel to this development effort, partitioning of multiple radionuclides has been performed with major radionuclides including 235U, 60Co, 134Cs, 136Cs, 137Cs, 65Zn, 54Mn, 51Cr, 55Fe, 59Ni, and 141Ce. For example, uranium and cerium were successfully partitioned from the metal phase to the ceramic phase to demonstrate the decontamination of scrap metal. No uranium or cerium were detected in the decontaminated metal product (lower detection limit 0.1 ppm), which had an initial contaminant concentration of >1,000 ppm. For low level wastes containing volatile radionuclides (e.g., Zn, Cs), bench scale gas phase decontamination factors (DFs) >105 were achieved (analytically limited at intermediate point in gas handling train) corresponding to commercial designs with DFs >107. Hence the efficacy of targeting a radionuclide to a desired product phase to achieve maximum volume reduction and/or recovery of valuable products has been confirmed.

M4 Environmental, LP, a limited partnership between Lockheed Martin Corp. and Molten Metal Technology, Inc. has built and commissioned a small-scale processing/demonstration facility in Oak Ridge, TN, which has begun to process DOE mixed wastes in a privatized mode. Initial target streams include sludges and gaseous diffusion plant wastes..

TECHNOLOGY DESCRIPTION

Catalytic Extraction Processing (CEP) is a proprietary technology that allows waste materials of a wide range of chemical and physical forms to be processed leading to decontamination and resource recovery. For processing of radioactive and mixed waste streams, CEP permits both complete destruction of hazardous contaminants and controlled partitioning of radionuclides in a single step. This leads to the potential formation of one or more decontaminated product phases (fuel gas and/or metal alloys), as well as volume reduction and concentration of radionuclides in a stable condensed phase for final disposal. A typical CEP unit consists of a feed preparation system, the Catalytic Processing Unit (CPU) or the reactor holding the molten metal, followed by a gas handling and recovery train. Upon injection into the metal bath, feed materials dissociate into their elemental constituents aided by the catalytic and solvating properties of the liquid metal. Select addition of co-reactants or manipulation of operating conditions, together with the thermodynamically controlled reaction pathways, enables synthesis of products and/or partitioning of compounds into the desired phases.

The CEP process can be conceptually divided into two stages:

Catalytic dissociation and dissolution: The catalytic effect of the molten metal bath causes complex compounds in the feed to be dissociated into their elements, which readily dissolve in the metal bath. These soluble elements form dissolved intermediates.

Product synthesis/partitioning: Through the addition of select co-reactants or by controlling operating conditions, the dissolved elemental intermediates can be reacted to form desired products or made to partition to the desired phases. These reactions are driven by thermodynamic forces and solution equilibria.

The solubility of carbon in iron is key to CEP's processing of organic and organo- metallic feeds. Specifically, carbon from the dissociation of organic compounds in the feeds, readily dissolves maintaining a homogeneous reducing environment in the molten metal solvent for a large number of compounds including carbon dioxide, sulfur dioxide, water and a wide range of metal oxides. The reducing strength of dissolved carbon provides CEP the flexibility to recycle organics to synthesis gas, recover metals and inorganics such as halides, sulfur and phosphorus, and separate and concentrate transuranic compounds.

As outlined above, the molten metal bath in the CEP system ensures dissociation of the feed into 'singular' elemental intermediates. The partitioning of feed components and the synthesis of products of value is dependent on manipulating the reaction pathways of the dissolved elemental intermediates. Potential reaction pathways are controlled by reaction thermodynamics.

Metal Recovery

A significant advantage of CEP technology is its ability to recover and recycle metals from organo-metallic feeds. Consider an organo-metallic feed dissociating into its elemental constituents (e.g., C, H, M, where M is the metal) upon injection into the CEP reactor. Both the carbon and the metal will readily dissolve in the molten metal solvent. If oxygen is added as a co-reactant, two reaction pathways are possible:

$$2C + M + O_2(g) \rightarrow 2CO(g) + M \quad (1)$$

$$C + x\,M + y\,O_2(g) \rightarrow M_xO_{2y} + C \quad (2)$$

Investigation of the free energies of formation indicates which systems will proceed with formation of carbon monoxide (reaction pathway (A)) and which systems will proceed with formation of the metal oxide (reaction pathway (B)). Specifically, carbon monoxide formation is favored in systems containing metals such as nickel, cobalt and copper whose free energy of oxidation is higher than that of carbon. These metals can be recovered as alloys in the iron solvent. Metals with a free energy of oxidation lower than that of carbon (e.g., aluminum, calcium) will form oxides and partition into the ceramic phase. Certain metals can either be recovered as metal alloys or as oxides in the ceramic phase depending on the operating conditions. For example, magnesium is reduced by carbon at elevated temperatures (T1800C, P=l atm). However, at elevated pressures, magnesium oxide reduction by carbon, will not be favored.

Halogen Recovery

Under the Resource Conservation and Recovery Act (RCRA), chlorinated organic compounds of concern include the primary organic hazardous constituents (POHCs) in the waste. Carbon tetrachloride, chloroform, para-dichloro benzene, perchloroethylene, tetrachloroethane, 1,1,1-trichloroethane, trichloroethylene, methyl chloride, polyvinyl chloride

(PVC), and polychlorinated biphenyls (PCBs) and all their intermediates and derivatives are among the common organic chlorinated compounds.
CEP technology can recover and recycle halogens from halogenated organic and inorganic feed streams without the formation of undesired by-products (e.g., chlorophenols, dioxins and furans). In order to demonstrate the recovery and recycling of halides in the CEP system, consider the case of a chlorinated organic feed stream. On injection into the molten metal bath, the feed will dissociate into dissolved elemental intermediates. Chlorine can be recovered and recycled as either a metal chloride or hydrogen chloride, or as chlorine gas in the absence of hydrogen. When a ceramic layer is introduced during CEP operation, thermodynamic control predicts predominant retention of chlorine in the ceramic phase under optimized conditions.
The two reaction pathways are:
Formation of Metal Chloride: Calcium, in the form of calcium oxide, added to the molten iron bath will form calcium chloride. The effectiveness of calcium as a chloride scrubbing agent is demonstrated by the Gibbs Free Energy of formation of common chlorides as a function of temperature diagram, shown in Fig. 1. Calcium chloride is volatile under CEP operating conditions and will distribute itself in the gas product phase where it can be cooled and separated from the other gaseous components as a condensed solid. However, by manipulating the ceramic phase composition, the metal halide can be prevented from distributing into the gaseous phase.
Formation of Hydrogen Chloride: In the absence of calcium, both hydrogen chloride and ferrous chloride will be formed as the free energy of formation for both compounds is of the same order of magnitude (Fig. 1). If CEP is operated at temperatures of 1500C, the ferrous chloride will volatilize and be recovered as a condensed solid in the gaseous stream. If the CEP is operated at lower temperatures (1000C) ferrous chloride can be recovered as a liquid in the vitreous phase.
Fig. 1
Partitioning of chlorides in a metal/gas/ceramic phase system, however, must take into consideration the effects of mass transfer issues within the ceramic phase. Even though the free energy of reaction can be treated as a true measure of the driving force for the reaction, mass transfer issues can limit the extent of the reaction such that thermodynamic equilibrium is not achieved. In the case of chloride processing with a metal/CaO-Al2O3-SiO2 ceramic phase, the formation of 'CaCl2' in the ceramic phase is not only dependent on the activities of the constituents, but also influenced by ceramic phase viscosity, reaction-rate, and residence time of the gas in the ceramic phase. CaO-Al2O3-SiO2 systems are of primary interest to MMT and have been extensively studied and optimized for potential capture of chlorides as CaCl2.
The CEP flexibility in recovering halogens is a major advantage given the technical difficulties and expense associated with processing halogen-containing waste streams in typical open flame combustion processes such as incineration. Halogen containing compounds tend to suppress combustion and are often characterized as non-combustible. Therefore, incineration of these chemicals can result in incomplete combustion leading to formation of highly toxic compounds. The free radical chemistry that occurs during combustion produces radical intermediates which can react to produce dioxins and furans.
Transuranic Recovery

An important advantage of the CEP technology is that it can separate and concentrate radioactive compounds from complex contaminated feeds while also recovering and recycling the non-radioactive components of the feed. This attribute of CEP has important implications in the decontamination of radioactive sites as well as the decommissioning of nuclear weapons and the byproducts of their manufacture. This implies that radioactively-contaminated components from DOE sites can be separated, highly concentrated and immobilized in a small volume that can be safely disposed of. Furthermore, feed components such as reducible metals (e.g., nickel, chromium, manganese) can be recovered as alloys while hazardous organic components can be effectively converted to a stable final form. Table I shows that the Gibbs Free Energies of reduction by carbon of transuranic oxides are positive indicating that the reactions are not thermodynamically favored. It is this oxide stability that provides a mechanism for the partitioning of these radioactive species into the ceramic phase. Data collected from the literature demonstrates the partitioning of transuranic components from the metal to the vitreous phase. Specifically, residual concentrations ranging from 0.05 ppm to 2 ppm of such radioactive components were achieved by others using diffusion of oxidizing, vitreous-forming agents to partition the radioactive components.

Table I

Experimental Feed Conversion Demonstrations

The processing and conversion of organic, organo-metallic and inorganic compounds in mixed waste streams into stable materials has been demonstrated theoretically in the discussion above. In addition to the theoretical analysis, MMT has collected extensive experimental data in bench-scale and pilot plant units and in commercial-scale metallurgical systems. The experimental data validates the theoretical predictions and confirms CEP's capabilities to generate high quality products.

The physical chemistry of CEP, specifically, the solution equilibria and the formation of a 'singular' dissolved intermediate in the reaction pathway, ensures that complete dissociation of feed streams occurs and that product synthesis and recycling can be controlled and manipulated. Therefore, the molten metal, acting as a homogeneous catalyst and solvent with high chemical inertia, renders CEP a highly flexible and robust recycling technology.

Targeted Partitioning to Durable, Stable Form

Previously published studies have demonstrated the ability to partition radioactive components, such as uranium and plutonium, from the metal phase into a vitreous phase via melt refining. Residual concentrations ranging from 0.05 ppm to 2 ppm were achieved using diffusion of oxidizing, vitreous-forming agents to partition the radioactive components. Successful partitioning to less than 10 nCi/g levels (approximately < 0.1 ppm) has been demonstrated for uranium- and plutonium-contaminated metals. CEP offers the potential for superior performance, as the techniques involved in melt refining are completely incorporated and enhanced in CEP technology. Specifically, CEP incorporates active radionuclide partitioning through select co-reactant additions (e.g., oxygen) and enhanced mass transfer (e.g., convection), while melt refining is based upon 'passive' diffusion-based partitioning. MMT designed a series of bench-scale experiments using radioactive surrogate materials to demonstrate the oxidation and partitioning of radionuclides between the metal and vitreous phases and to identify and

optimize the effect of operating conditions on partitioning. The bench-scale experiments were followed by experiments on pilot- and demonstration-scale systems. Hafnium was selected as a radioactive surrogate due to its similarities in thermodynamic behavior (Fig. 3) and physical properties (Table II) to uranium. Zirconium, although similar to uranium in the thermodynamic properties, was not chosen as the primary surrogate material because of its significantly lower density. Following experiments with surrogates, actual radionuclides were successfully processed in CEP to confirm earlier results.

The experimental design parameters evaluated were metal system, gas environment and ceramic phase composition. Radioactive surrogate partitioning was measured using neutron activation analysis (NAA) for non-destructive trace analysis. The hafnium LDL was 0.2ppm in iron and 2ppm in nickel. X-ray fluorescence (XRF) was used for analysis of the vitreous phase composition. Metal samples were taken at different positions in the metal bath to ensure uniform decontamination. The contract objective was to demonstrate uniform metal decontamination above 98%. In addition, backscattered electron imaging and X- ray analysis were used to investigate the nature of the capture and stabilization of the radionuclide surrogate in the vitreous phase.

Fig. 2

Table II

Average decontamination of the metal samples were >99% and exceeded the contract's 98% objective in all experimental trials. The calculated decontamination factors were analytically limited with no radioactive surrogate detected in the metals. Table III summarizes the results. V1 refers to aluminosilicate vitreous compositions while V2 refers to borosilicate vitreous compositions.

Table III

Samples taken at different positions in the metal bath demonstrated uniform distribution of any trace amounts of hafnium across the metal (Fig. 3 and Fig. 5). This indicates that radioisotope transport to and incorporation in the oxide phase is very efficient under CEP operating conditions.

Fig. 3

Fig. 4

Ceramic samples were analyzed using backscatter electron imaging to identify the nature of the radionuclide capture and stabilization in the ceramic phase. This work indicated that operating conditions and vitreous phase composition can affect radionuclide capture and stabilization. Figure 6 shows the backscatter image of a vitreous sample which has separated into hafnium rich and hafnium poor regions. This is in contrast to the vitreous sample shown in Fig. 7, where hafnium is uniformly distributed. CEP conditions have been identified to achieve this desired vitreous radionuclide capture and stabilization mechanism.

Fig. 5

Fig. 6

Processing of RCRA Contaminants

CEP has been demonstrated on a range of RCRA-listed wastes as well as characteristic and RCRA-like surrogate material (Table IV) including high-molecular weight aromatics, chlorinated organics, organically-bound nitrogen species (isocyanates), plastics, and organometallics. The technology's environmental performance has been demonstrated during actual waste processing to meet and surpass current and proposed

regulatory standards. Specifically, DREs >99.9999% were achieved for principle organic hazardous constituents (POHCs). NOx and SOx were not detected in the product gases to detection limits of 3 ppm. Condensed phase non-leachable products, both ceramic and metal phases, of marketable composition were generated. Dioxins were not detected to the targeted regulatory limit of 0.1 ng 2,3,7,8 TCDD TEQ/Nm3.

Currently the effectiveness of hazardous waste treatment technologies is regulated by the EPA, by measuring the concentration of hazardous materials in the after-process wastewater and non-wastewater. All processes have to ensure that organic constituent concentrations in individual wastewater and non- wastewater streams are lower than the regulated limits set by EPA. CEP demonstrations surpass the current limits even when compared with the Best Demonstrated Available Technology (BDAT). CEP was recently approved by EPA (on October 24, 1994) as a non-combustion technical equivalent (BDAT) for eight RCRA-listed isocyanate waste codes (K027) for which incineration had previously been mandated as the commercially available BDAT technology.

The environmental performance of CEP has also been demonstrated on chlorinated waste streams, containing some of the most difficult to destroy hazardous constituents. The results of demonstration-scale processing of RCRA-listed waste F024 (chlorinated aliphatics) indicated that hazardous organic constituents in the feed were not detected in ceramic, metal and scrubber water, thereby surpassing the BDAT standards for all effluent streams (Table V). Destruction Removal Efficiency (DRE) on multiple organic hazardous constituents exceeded 99.9999%, which surpassed the current regulations mandating DREs >99.99% (Table VI). Trace constituents were not detected to the targeted regulatory limit of 0.1 ng/Nm3 TEQ.

Table IV

Table V

Table VI

CEP reaction pathways, combined with the uniform and highly reducing environment maintained over a wide range of operating conditions, preclude the formation of dioxins, furans and other undesirable by-products. Research suggests that all of the following conditions must be present to promote the formation of dioxins: excess free O2 (10-3 atm partial pressure), the presence of aromatic precursors, sufficient residence time in the temperature range around 300C (572F), and the presence of heterogeneous catalytic surfaces (e.g., particulates).iii,iv,v,vi These essential conditions for formation of dioxins are either inherently absent or can be minimized through appropriate design and operation of the CEP system.

Extremely low oxygen concentrations in the CEP off-gas will inhibit the formation of dioxins. Oxygen partial pressure is continuously monitored in the prototype demonstration unit and is not detected. Thermodynamic evaluation of CEP reaction scenarios shows that partial pressures of O2 can approach 10-15 atm.

CEP solution chemistry can ensure that all species pass through a dissolved intermediate and preclude the formation of aromatic precursors. Table IV shows that Destruction and Removal Efficiencies (DREs) are greater than 99.9999% for chlorinated plastics (PVC and polystyrene) and chlorinated solvents.

Delisting

CEP has been demonstrated on a range of RCRA-listed wastes as well as characteristic and RCRA-like surrogate material (Table IV) including high-molecular weight aromatics, chlorinated organics, organically-bound nitrogen species, plastics, and organometallics. CEP's ability to provide complete destruction of the hazardous constituents while maintaining high regulatory integrity has been proven. Due to the unique set of regulations governing the processing of RCRA-listed materials, the disposal of a stable final form containing radionuclides but not exhibiting any RCRA characteristics would benefit from delisting. Delisting would provide increased flexibility in terms of disposal destination and cost.

A delisting petition has been prepared based on experimental data generated from pilot plant and demonstration-scale testing. The delisting petition is an upfront application for a generic delisting of the ceramic phase generated from CEP systems. Five representative mixed waste streams have been selected for delisting:

- Organic sludges: e.g.,ORR MWIR 3090 (F001-F002 solvents, F006 electroplating waste); INEL ID-EGG-158:3
- Inorganic sludges: e.g., ORR MWIR 3004 (F006 electroplating waste); INEL ID- EGG-102:7 (toxic organic and metals with mercury)
- Soils and sediments: e.g., ORR MWIR 3151 (Pb, Hg); INEL ID-EGG-141:990
- Combustible debris: e.g., ORR MWIR 2028 (F003); INEL ID-EGG-114:337
- Scrap metals: e.g., INEL ID-EGG-132:20

The experimental results presented in support of the delisting petition included feed materials characterization, ceramic elemental composition, ceramic hazardous characteristics evaluation, toxicity characteristics leachability procedures (TCLP) testing, and hazardous organic constituents concentration in the ceramic phase.

For a waste to be successfully delisted, the petition must demonstrate that the waste:

- does not meet the criteria for which it was listed
- does not exhibit any hazardous waste characteristics
- does not exhibit any additional factors including additional constituents which may cause the waste to be a hazardous waste

In preparing the delisting petition, MMT has followed the requirements for delisting petitions set forth in 40 CFR 260.20 and 260.22, as well as in relevant EPA guidance (such as the guidance manual for petitions to delist hazardous waste (EPA/530-R-93-007, March 1993)). These regulations and guidance specify the administrative information, waste characterization and waste management history, process information, waste sampling and analysis information, and ground water monitoring information that must be included in the delisting petition.

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INNOVATIVE VITRIFICATION FOR SOIL REMEDIATION

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ABSTRACT
Vortec has successfully completed the detail design of a high throughput glass melter for the DOE "Innovative Fossil Fuel Fired Vitrification Technology for Soil Remediation" for demonstration program.
The principal objective of the program is to demonstrate the ability of the Vortec Cyclone Melting System (CMS) to remediate DOE contaminated soils, mixed wastes and other waste forms containing RCRA hazardous materials, low levels of radionuclides, and tSCA (PCB) containing wastes. The demonstration program will verify the ability of this vitrification process to produce a chemically stable glass final waste form which passes both TCLP and PCT quality control requirements, while meeting all federal and state emission control regulations. The demonstration system is designed to process 36-72 ton/day of as-received drummed or bulk wastes.
Contruction will begin in mid 1996 with the testing to be accomplished in the first quarter of 1997. The demonstration testing will be conducted at the DOE Gaseous Diffusion Plant in Paducah, KY.
This paper presents a description of the program, equipment, and testing accomplished to date.

INTRODUCTION

The objective of this DOE demonstration program is to validate the performance and operation of the Vortec Cyclone Melting System (CMS) for processing of LLW contaminated soils found at DOE sites. This DOE vitrification demonstration project has successfully progressed through the first two phases. Phase 1 consisted of pilot scale testing with surrogate wastes and the conceptual design of a process plant operating at a generic DOE site. During Phase 2, a site specific design was developed for the processing of LLW soils contaminated with TSCA organics and RCRA metal contaminants. Phase 3 will consist of construction of a scale demonstration plant at the DOE-Paducah Gaseous Diffusion Plant located in Paducah, KY.

The overall system includes the capability to shred entire drums and drum packs containing soil, concrete, plastics and PCB's. This enhanced processing capability will substantially expand the total DOE waste remediation applications of the technology.

A total of seven soil vitrification trials of simulated waste soils containing radionuclide surrogates were conducted during Phase 2 at Vortec's pilot scale vitrification plant located at the University of Pittsburgh Advanced Research Center (UPARC) in Harmarville, PA. The sampling of the effluent and influent streams taken during the tests confirmed that virtually all of the refractory radionuclides were retained in the glass and would not leach to the environmentas confirmed by both Product Consistency Tests (PCT) and Toxicity Characteristic Leaching Procedure (TCLP) testing. The organic contaminants, anthracene, 1-2 dichlorobenzene, and naphelene, was destroyed during testing with a Destruction and Removal Efficiency (DRE) of at least 99.99%. Semi-volatile RCRA metal surrogates were captured by the Air Pollution Control (APC) system, and data on the amount and the chemical composition of the particulate were established for use in the APC system design.

PROGRAM OBJECTIVES

The principal objective of the METC/Vortec program is to demonstrate the ability of the Vortec 36-72 ton per day vitrification system to remediate DOE contaminated mixed waste and other waste forms, contaminated with

both hazardous materials and low levels of radionuclides and PCB, by producing glass which passes TCLP and PCT.

The following other objectives will be met during the program:

1. Determine the glass chemistry requirements to achieve effective vitrification of contaminated soils.
2. Determine expected feedstock particle size distribution and the glass flux requirements.
3. Determine the Destruction Removal Efficiency (DRE) for organic contaminants.
4. Establish the characterization of the off-gas so that valid designs and cost estimates can be made.
5. Establish the cost of a fully integrated waste vitrification system with a 36-72 TPD capacity .
6. Conduct start-up, shake-down, and feasibility demonstrations using the fully integrated plant.

BACKGROUND INFORMATION

The Department of Energy's goal to clean-up its nuclear complex requires the development of innovative technologies to convert soils contaminated by hazardous and/or radioactive wastes to forms which can be readily disposed in accordance with current waste disposal methods. These technologies must be able to: accomplish this task with minimum public and occupational health risks, with minimum environmental risks, and in a timely and economical manner; transform the hazardous and/or radioactive waste into a form which has long-term stability to prevent migration of RCRA metals and radionuclides; not present any major obstacles to its own safe decontamination and decommissioning; and must have long term stability.

After four years of design evolution under various DOE and EPA programs, the melter is operational at the UPARC test facility. Vortec's system has demonstrated the production of glass and the vitrification of a variety of feedstocks including EPA surrogate soils, spent pot liners (K-088) wastes, coal fired boiler ash, sewage sludge ash, auto shredder residue ash, municipal solid waste incinerator ash, metal plating sludges, fiberglass waste with organic contaminants, dusts containing heavy metals and organic materials, and electronic industry wastes.

The unique features of the CMS technology make it a particularly cost-effective process for vitrification. Benefits with respect to DOE's needs are:

1. Long-term immobilization of heavy metals, toxicinorganics, and radionuclides.
2. Effectively oxidize and destroy organic contaminants.
3. Flexibility in processing solid wastes with substantial variations of feedstock composition.
4. Ability to oxidize and vitrify waste materials introduced as slurries. Vortec has demonstrated the ability to vitrify Hanford low level tank waste simulant with a water content of approximately 70% liquid and 30% solids.
5. A sealed process which can be operated at negative pressure to prevent leakage of contaminated gases.
6. A small physical size which is transportable and modular to reduce the decommissioning and disposal costs of the process equipment.
7. Life cycle cost lower than other existing vitrification processes. In commercial applications, a 72 TPD process unit typically has total processing costs in the range of $50 - $100 per ton of material

processed. Radionuclide and PCB contamination increases the per-ton cost somewhat. Vortec estimates that the processing costs of low level waste with mixtures of TSCA or RCRA wastes at Paducah will be in the range of $50 to $200 per barrel for the Paducah drummed wastes. In comparison, of the types of wastes to be processed, other competing remediation technologies have estimated processing costs in the range of $500 to $1000 per barrel for drummed wastes.

Process Description

The primary components of the vitrification system are a counter-rotating vortex (CRV) heater and a cyclone melter. An artist's rendering of the concept is shown in Fig. 1. A unique feature of the process is the rapid suspension heating and oxidation of feedstock materials in the CRV prior to the physical and chemical melting processes which occur within the cyclone melter. The use of the Vortec CRV in conjunction with a cyclone melter distinguishes the Vortec melting technology from other types of cyclone systems. In the CMS process, granular glass-forming ingredients and other feedstocks are introduced into the top region of the CRV along with fuel and air. As a result of the intense counter-rotating vortex mixing, it is possible to achieve stable combustion in the presence of large quantities of inert particulate matter (solids-to-gas mass ratios on the order of 1:1). Both convection and radiation heat transfer mechanisms contribute to the rapid heating of the feedstock materials within the CRV. Any organic contaminants in the feedstocks are also effectively oxidized.

Fig. 1

The melted product formed in the cyclone melter, and the off-gas, exit through the cyclone melter and enter a separator-reservoir (not shown in the figure) where the liquid exits the reservoir through a bottom or side tap, and the off-gas exhausts through an air preheating and pollution control system.

The off-gas exiting the separator-reservoir is treated in an air pollution control system prior to being exhausted out the stack. As a result of the high thermal efficiency of the Vortec melter, the off-gas flow rates are relatively modest. Because the temperature and composition of the vitrified product can be closely controlled, the amount of process fuming (volatile carryover) can also be minimized.

The average gas-solids suspension temperature leaving the CRV is typically on the order of 2000F to 2700F. The process temperatures in the cyclone melter are typically in the range of 2000F to 3000F. The NOx emissions have been found to be substantially lower than those which occur in conventional cyclone systems. Excess air levels are typically in the range of 5 to 20% depending on the composition of the feedstock being processed. The pilot system has demonstrated NOx emissions of less than 4 pounds per ton of vitrified product, meeting the California emission standard for glass melting furnacescurrently the most stringent in the United States.

Heat rates demonstrated by the Vortec pilot scale facility typically ranged between 3.5 and 6 million Btu/ton at a glass production rate of 15 TPD. This heat rate is 50% to 80% lower than heat rates for conventional gas-fired glass melting at similar capacity. The savings are due to more efficient heating of the glass ingredients and lower structural heat losses. The melter can also accommodate the use of a variety of fuels, such as oil and coal-derived fuels, and even organic waste materials.

The system has demonstrated uncontrolled emissions levels of less than 0.5% of feed materials which did not contain low temperature volatiles, such as utility flyash. For materials containing heavy metals and other volatiles, such as MSWI flyash and fiberglass waste, the uncontrolled emissions levels have typically been in the range from 1% to 4%.

RESULTS

Vortec has successfully completed the verification testing and final baseline plant design required in Phase 2 of this program, and will continue the development, construction, and operation of the Demonstration Plant during Phase 3. Vortec is developing the technology to commercial readiness, and will meet all public, occupational, and environmental health and safety requirements for remediation technology. Commercial offerings of the technology, in plant sizes up to 200 TPD, have been made during the last year.

Test Program-Phase 2 Results Summary

A total of seven soil vitrification trials were conducted at Vortec's pilot plant during Phase 2 of the program. The objectives of the pilot testing were to demonstrate the effective vitrification of low level waste streams (soil) with the characteristics and compositions found at DOE-Hanford and DOE-Paducah; evaluate the melting performance with this material, that is, the feedstock composition and viscosity relationship; define the expected range of flue gas emissions; optimize the system operating parameters for the waste by determining the effect of temperature on the melting performance and on the capture rate of the surrogate contaminants in the vitrified product; and to determine the PCB destruction efficiency of the CMS.

The first set of trials used a surrogate soil composition representing the contaminated soil found at DOE's Hanford site. These tests evaluated the melting performance and expected range of flue gas emissions from the system when processing the Hanford surrogate soil feedstocks which consisted of a synthetic soil spiked with surrogate heavy metal and radionuclide contaminants. Success was measured by the ability of the melter to produce a fully-reacted vitrified product which passed both the Toxicity Characteristic Leaching Procedure (TCLP) and the Product Consistency Testing (PCT). Samples of all effluent streams were analyzed to establish the partitioning of the heavy metal and radionuclide surrogates. These tests determined the effect of temperature on the melting performance and confirmed the capture rates of the surrogate heavy metal and radionuclide contaminants in the vitrified product. Analyses of the off-gas were conducted to establish the design specification for the air pollution control system.

Five additional vitrification trials were also conducted using surrogate contaminated soil representative of the soil found at the DOE-Paducah site. The surrogate soil feedstock used consisted of a synthetic soil modeled on the data received from DOE-Paducah from their low level waste inventory. Samples of all effluent streams were analyzed to establish the partitioning of the heavy metal and radionuclide surrogates. Flue gas samples were also analyzed for PCB's.

The glass analysis was conducted on the collected samples by Corning Engineering Laboratory Services (CELS), and the TCLP testing was conducted by Blue Marsh Laboratories (BML). The TCLP analysis indicated that the TCLP extract contained very little measurable quantities of metals and in all cases were significantly below EPA TCLP limits. PCT test results indicated a Na-normalized leach rate of 0.0032 to 0.015

grams of glass/square-meter/day. The PCT specification for nuclear glasses is a Na-normalized leach rate of no greater than 1.0 grams of glass/square-meter/day. These results indicate that the glass produced is approximately 2 orders of magnitude better in retaining radionuclides than the conventional environmental assessment glasses.

The best data available from DOE-Paducah indicated that the low level waste stream (soil) contained small amounts of organic materials and small amounts of heavy metals, uranium, and plutonium. Since the worst organic material to remediate is the PCB, 1, 2 dichlorobenzene was used as an organic surrogate at a concentration approximately of 1000 PPM, a concentration well beyond what is expected in the actual low-level waste stream. Each test focused on establishing the DRE for its chemical compound. As with the tests with the Hanford soils, the DRE results were at least 99.99%.

In addition, Cerium was included at 500 PPM as a surrogate for uranium or plutonium, and the semi-volatile RCRA metals lead and cadmium were also included. Vortec has demonstrated that approximately 95% to 100% of the non-volatile RCRA metals report to the glass.

Design Program-Integrated Demonstration Plant

The major system requirements for the Demonstration Plant are as follows:

1. Targeted waste stream is 55 gallon drums of contaminated soils containing debris such as concrete, tramp metal, wood and plastics.
2. The process will be capable of processing waste containing radionuclides, TSCA, and RCRA contaminants.
3. Waste streams with up to 30% moisture at a nominal processing capacity of drummed waste of 160 drums per day.
4. The plant will be transportable and modular allowing use at multiple DOE sites and/or multiple locations at a single site.
5. The Demonstration Plant will be capable of processing a wide variety of physical and chemical waste forms throughout the DOE complex. The wastes include soils, sediments, and/or sludges contaminated with hazardous wastes and low-level radioactive wastes. Both volatile (Technetium) and nonvolatile (Uranium, Neptunium, Thorium, and Plutonium) radionuclides may be present in the soil waste stream. The eight heavy metals regulated by 40 CFR 261.24 are also present in the soil. Organic materials that can result in Hazardous Air Pollutants regulated by State of Kentucky 401 KAR 63.022 are also present in selected waste streams. Optional waste streams may include but are not limited to: personnel protective equipment (PPE), HEPA filters, treated scrubber / ESP water particulate and spent ion exchange materials.
6. The system will produce a glass frit, a chemically stable and reduced volume final waste form, that will pass the Toxicity Characteristic Leaching Procedure. The Air Pollution Control (APC) system will be required to meet DOE/EPA and the State of Kentucky standards for the removal of hazardous material and radionuclides.
7. A wastewater treatment process will remove radionuclides and other solids from process water and these solids will be recycled through the melting system.
8. The vitrified product generated will be disposed on-site or at an approved DOE site.

A system flow diagram is shown in Fig. 2, and an isometric drawing of the plant arrangement in shown in Fig. 3. The demonstration plant has been

designed as a transportable and modular system; that is, the individual, skid
Fig. 2
Fig. 3
mounted components of the process have the capability to be transported by truck, without special permits, to the site, erected, and when operation is complete, dismantled, decontaminated, loaded back onto trucks, and hauled off-site.

To demonstrate the effectiveness of the technology, 400 hours of start-up and functional testing are planned, followed by a 30-day period of nearly continuous testing. During operation, contaminated soil is transported by DOE in 55 gallon drums from the DOE-PGDP storage area to the vitrification facility. There is always at least a three day supply of the material in the storage area. Soil samples collected prior to the 30 day demonstration test will be used to determine the batch composition.

The process of vitrifying the soil begins in the Feed Preparation Subsystem. It consists of 1) transportation of drums to the drum shredder for introduction to the feed preparation system; and 2) a drying and screening operation to assure that the material has the proper moisture content and size. Upon receipt, the drums are emptied into a drum shredder using a conventional fork lift truck with standard drum holding fixture. To preclude the escape of dust particles when dumping or transporting the soil, all the conveying systems will be designed with an enclosure and operate under negative pressure. In addition, all hoppers and transfer points (dumping points) will also be enclosed and will be under negative pressure. The dust laden air from these devices will pass through a dust collector for particle removal. Solids collected in the dust collector will be transported back into the system. Discharge from the dust collector will pass through a parallel pass HEPA filter system.

The sized and dried soil is transported to a storage silo. Glass making additives are mixed with the soil. Additives (limestone and soda ash) are used to aid in glass forming, obtain the proper glass properties, or modify the temperature-viscosity curve. The blending system consists of storage silos and pneumatic feed system for the delivery of the soil and additives to a blend tank. Batch mixing precedes feeding into the Cyclone Melting System.

The CMS equipment components consist of a counter-rotating vortex (CRV) oxidizer/heater, a cyclone melter (CM), a separator/reservoir, and a recuperator heat recovery unit. The prepared feedstock is introduced into the CRV oxidizer/heater through injectors located at the top of the CRV and the glass product and the off-gases exit the CM through a tangential exit channel and enter a separator/reservoir.

The separator/reservoir separates the off-gas from the melted material and provides an interface with a vitrified product handling system. The off-gas exits through an exhaust port which is the interface for the recuperator. The recuperator utilizes the waste heat to preheat air going to the melter. Molten glass flows out the separator/reservoir to the Vitrified Product Handling System. The molten glass from the melter will be water quenched to produce a cullet approximately 1/8" in average size. The cullet will be transported by conveyor to B-25 boxes, that when full, will be moved to an area for pick-up and disposal by the DOE-Paducah.

The Air Pollution Control System (APCS) will consist of a wet electrostatic precipitator (WESP) system for particulate collection preceded by a venturi scrubber. The scrubber will remove large

particulate as well as serve the function of reducing the off-gas temperature. After removal of particles in the WESP, the temperature of the off-gas is raised in an off-gas heater prior to entering the HEPA filter for removal of fine particles. Redundant HEPA filters are used to facilitate maintenance. The off-gas exits the HEPA filters and flows to the exhaust stack. The APC system also includes a waste water treatment system to remove radionuclides from the process water used in the venturi scrubber and WESP. This system consists of a clarifier, a filter press, sand filter, ion exchange unit, and various pumps and tanks. Process water from the WESP flows through a wastewater tank, a chemical precipitation tank for chrome removal, and on to a clarifier. The solids from the clarifier, which contain some contaminants not captured in the glass, are dewatered in a filter press and are returned to the Feed Preparation System. Radionuclides are removed by first filtering the supernate water in a sand bed. The solids are removed periodically from the sand bed by back flushing with the treated water, and the backwash is reintroduced into the clarifier. Radionuclides are removed through ion exchange treatment. The treated effluent is stored in a holding tank for reuse as quench water within the quencher/venturi scrubber.

The Instrumentation and Control System consists of the sensors, electronics, instrumentation, computers, and programmable logic controllers (PLC) to control the process in real time, gather data for analysis on system and equipment performance, and monitor process off-gas. The system is capable of being shut down in emergency situations in a controlled manner using the auxiliary power unit and structured logic.

SCHEDULE

Vortec has completed Phases 1 and 2 of a three phase program to design, construct, and demonstrate the effectiveness of the CMS technology at remediating contaminated soils. The ability of the CMS to vitrify soils similar to the soil found at DOE-Paducah was demonstrated. The vitrified product passed TCLP as well as PCT for leachability of and durability. In addition, the final design of a 36-72 TPD demonstration plant to process contaminated soil is completed. Phase 3 will carry out the construction of the plant and conduct the 30 days of demonstration testing. Figure 4 represents the tentative schedule for Phase 3.

Fig. 4

ACKNOWLEDGMENT

Project sponsored by U.S. Department of Energy's Morgantown Energy Technology Center, under Contract DE-AC21-92MC29120 with Vortec Corporation. The authors wish to acknowledge the contributions of DOE-METC COR Mr. Cliff Carpenter. The period of performance of the contract is March 1993 to September 1997.

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THE FERNALD MOBILE MIXED WASTE STABILIZATION PROJECT

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ABSTRACT

DOE Fernald is nearing completion of a project objective to treat legacy mixed waste for shipment and disposal to the Nevada Test Site (NTS). In pursuing this challenging objective, DOE-Fernald and the site cleanup contractor, Fernald Environmental Restoration Management Corporation (FERMCO), have defined the requirements and purchased the fixed-price services of a private waste treatment subcontractor. The subcontractor has mobilized and initiated stabilization of the mixed waste. The subcontractor is a team consisting of Perma-Fix Environmental Services, Inc. (Perma-Fix), and Performance Development Corporation (PDC).

The mixed waste at Fernald was generated during uranium processing operations and remedial actions at the DOE facility. This waste is classified as both low-level radioactive and hazardous, which requires treatment prior to disposal at NTS. The Mobile Mixed Waste Stabilization Project is using the Perma-Fix Process to chemically stabilize and solidify approximately 1,850 drums of waste. Chemical hazards of the targeted project waste are soluble heavy metals including lead, barium, and chromium (EPA waste codes D004 - D011).

Following processing by the Perma-Fix team, successful treatment of the waste is verified by an NTS approved laboratory using the toxicity characteristic leaching procedure (TCLP). The treated waste no longer exhibits the hazardous characteristic and will be shipped in metal boxes as solid grout monoliths to DOE's NTS Facility for disposal as low-level radioactive waste.

Under the Federal Facility Compliance Act (FFCAct), FERMCO has developed a Site Treatment Plan (STP) that specifies a preferred option for each of several mixed waste streams at Fernald. This project meets the requirements for mobile stabilization as Fernald's preferred option in the STP. By implementing the STP at the Fernald Site, FERMCO has taken the lead role in complying with the requirements of the FFCAct.

A number of lessons have been generated by this bold new approach to cleaning up DOE's waste inventories. This paper will capture many of the lessons learned on this project in promoting what we believe is a promising strategy for site clean-up that is faster, safer, and cheaper than many competing options. The key features of the project include mobile on-site treatment, fixed-price contracting, and integrated teamwork among Perma-Fix, FERMCO, and DOE FERNALD participants.

INTRODUCTION

The objective of the Mobile Mixed Waste Stabilization Project is to safely treat approximately 1850 drums of characteristic mixed low-level radioactive waste (LLRW) currently being stored at Fernald. These wastes have been characterized as containing waste constituents regulated under EPA waste codes D004 through D011. The wastes will be treated at Fernald and sent to the NTS for disposal using Best Demonstrated Available Technology (BDAT) to meet RCRA Land Disposal Restrictions (LDR) and NTS waste acceptance criteria (WAC) established under the Nevada Test Site - Defense Waste Acceptance Criteria, Certification, and Transfer Requirements (NVO-325, Rev. 1). BDAT will be accomplished using a cement based stabilization/solidification process that renders toxic heavy

metals and radionuclides insoluble by precipitation, and locks them in a grout matrix.

On-site waste treatment services are being accomplished by personnel from Perma-Fix and PDC working in conjunction with DOE Fernald and FERMCO personnel.

WASTE STREAMS

The 1850 drums of legacy mixed waste have been in storage in RCRA-permitted warehouses at Fernald. The drums of mixed waste have been grouped and categorized using Material Evaluation Form procedures developed by FERMCO. For many of the drums, Real Time Radiography (RTR) was used for identifying drum contents. This procedure uses x-ray technology to view the contents of a container, and to determine the physical properties of the waste such as presence of free liquid, particle size, and internal container capacity. The 1850 drums have been divided into six general waste categories:

1. Grit Blast residues - approx 905
2. Solidified furnace salts - approx. 145 drums
3. Sump cakes - approx. 465 drums
4. Construction rubble - approx. 40 drums
5. Miscellaneous trash - approx. 25 drums
6. Miscellaneous (1-5) additional waste - approx. 300 drums

STABILIZATION PROCESS

The waste treatment process employed in stabilizing Fernald waste is a production process based on optimal staging of incoming waste containers, screening and sorting the larger fraction, removing incompatible material, size reducing the larger fraction, and redistributing the reduced material with original container material. Treatment consists of mixing chemical reagents to precipitate soluble heavy metals, pozzolans, and fresh or contaminated water to create a slurry that is poured in metal boxes supplied by FERMCO. Figure 1 is a block flow diagram of the Mobile Mixed Waste Stabilization process. Figure 2 is a schematic illustrating how materials handling is accomplished.

Fig. 1

Fig. 2a

Fig. 2b

STABILIZATION UNIT

The mobile treatment unit includes the following engineered facility and equipment and are depicted in Fig. 3 - Equipment Layout:

1. Incoming waste staging area
2. Deheading and inspection station
3. Screen hopper
4. Sorting table
5. Shredder
6. Mixers
7. Contaminated water holding tank
8. Liquid reagent tank
9. HEPA filtration units
10. Cement silo
11. Empty drum staging area
12. Decanting and curing area
13. Tented soaking area

Fig. 3

RESULTS

As of January 31, 1996, 905 drums of grit blast have been successfully treated and are being prepared for shipment and disposal at NTS. Another 145 drums of solidified furnace salts have been treated and are awaiting the results of confirmation sampling by TCLP. The project had planned to process an average of 35 drums per work day. To date, processing results have varied significantly, some days have included no processing due to equipment maintenance or staging efforts, and many days have seen processing rates of 50 to 65 drums. Our best efforts have sustained better than 10 drums per operating hour. The project is planned to complete processing of the contract goal inventory by March 20, 1996. Based on our results to date, this is attainable.

LESSONS LEARNED FROM THE PROJECT

The lessons learned as the result of this very successful project include the following:

On-site treatment by a private subcontractor is achievable

There are doubts within DOE and prime contractor ranks that it is possible to bring private subcontractors on site, have them live by DOE rules, work with the site labor force, manage any kind of schedule, or control costs. This project has demonstrated each of these concepts to be false. For the Mobile Mixed Waste Stabilization Project, we entered into a fixed- price contract and prepared project-specific plans acceptable to the Ohio EPA, US EPA, and DOE in less than 60 days. This documentation included a CERCLA Work Plan, Health & Safety Plan, Quality Assurance Plan, Process Control Plan, project procedures, and Processing Area drawings. After Ohio EPA and US EPA approval, Perma-Fix mobilized the processing equipment, performed pre-operational testing, and successfully responded to an independent Operational Readiness Assessment demonstrating compliance with DOE requirements. This was also done in less than 60 days. When elements of the contract were found inconsistent with a pre-existing agreement, Perma-Fix worked out arrangements to allow appropriate participation of site workers. This action enhanced site response and created a teamwork atmosphere among project workers. The project is nearing completion. Treatment and disposal costs are low compared to off-site alternatives. There have been no price increases to the subcontract. Major portions of the site's mixed waste have been treated to greatly reduce public risk prior to shipment on the open highways.

DOE site labor force can work with private contractors if concerns are considered and resolved before operations begin, and cooperatively managed during the project

After the project began, FERMCO discovered that elements of Perma-Fix's work scope conflicted with the pre-existing labor agreement. Perma-Fix renegotiated project activities with FERMCO such that Fernald Atomic Trades and Labor Council (FAT&LC) union members had an expanding role in project activities. There had been considerable concern at the beginning of the project that it would be hard to manage the schedule in this way, but all schedules have been met to date.

Technology is only as important as a good plan, competent personnel, constant communication, and cooperation

There is considerable opinion that stabilization is a low-tech technology, not robust enough to attack DOE's mixed waste problems. Our experience has demonstrated that stabilization does work on a fairly broad range of waste types, can be mobilized and employed quickly, has a great deal of operational experience, has equipment and project costs

that are much better understood, is very amenable to small batches for handling the wide variety of waste types, can be performed inside currently existing facilities, and is much less costly than many of the more exotic technologies crowding the national headlines.

One of the lessons learned on this project has been the relative importance of technology compared to other project challenges. With adequate experience and due care, making a solid that passes TCLP is not that difficult. The challenge is in planning and executing project activities with sufficient detail to satisfy project regulators. These include overviewers from: quality assurance and health & safety, FERMCO project, DOE, Ohio EPA and US EPA, and NTS. Any project that is inadequately prepared to deal with this bigger challenge runs a much higher risk of failure.

Use waste to treat waste

Stabilization requires the addition of a number of non-waste materials during treatment of the waste. These include water, pozzolans, chemicals used to precipitate contaminants, and mixing. Some waste products can be used in this process. For the Mobile Mixed Waste Stabilization Project, the water balance was managed to allow for the reuse of waste water. Additionally, we identified and were able to use surplus chemicals that were waiting for disposition on the Fernald site. These chemicals were used in waste processing for precipitating contaminants. The project also processed much of the solid waste generated during operations with the legacy waste stream. These actions support the waste minimization/ pollution preventative initiatives driven by both EPA and DOE.

Limit government liability with low cost commercially available equipment

One of the major risks of treating mixed waste, on site or off site, is the potential for the processing equipment to become contaminated, and failure to decontaminate causing the government to become the owner of some used contaminated equipment. This situation can and has happened many times. The lowest-risk way of limiting government liability is for contractors to propose and use as much simple, inexpensive, and commercially available equipment as possible, considering the minimum volume of equipment subject to waste disposal. Also design of equipment must address decontamination ability.

Small batch capability provides control features not available in continuous feed processes and is better suited to the many "cats and dogs" that exist in legacy wastes

Most of the money and the press in waste treatment has been spent on concepts using continuous processing equipment designed for the enormous volume of mixed waste to be treated. The problem with most large processes is that they generally work best on homogeneous and/or well characterized material. Much of the waste that has been identified at different DOE sites is far from homogeneous. Waste stream characteristics are rarely known well enough for continuous flow processes to respond to variations in the waste. Expensive automated systems can seldom manage highly variable waste streams, as well as trained and experienced operators can manage them. Stabilization is one of the few existing technologies offering the needed body of experience. When continuous processes are forced to deal with such varied material, it is often necessary to resort to batching. The equipment is usually not designed for batch operation. Batching in processes designed for continuous flow is achievable, but quite expensive compared to simpler options.

Small business can do the job

Another myth often believed by people in this business is that only major corporations are viable to attempt this type of project. The false impressions include having a larger talent pool to draw upon, deeper pockets if there is a problem, a more relevant experience base, an ability to impress the regulators and a skeptical public, and a better capability to deal with problems. The Mobile Mixed Waste Stabilization Project has employed the services of Perma-Fix and PDC, both small businesses. They have performed well. There have been many challenges for both companies on a number of occasions. Their senior managements have participated fully. Basically there is a lot more at stake with these two small businesses than there would have been with large businesses and the net effect has been an extremely responsive project organization. Our aggressive schedule would not likely have been met except for this level of attention.

On-site CERCLA treatment can be faster, safer, and cheaper than off-site RCRA Treatment, Storage, and Disposal (TSD) Facilities

Many DOE sites have concluded that the only viable alternative is to send mixed waste off site to RCRA TSD facilities that also have NRC licenses. The problem has been that this includes a very small group of companies, with technologies having narrow application, difficult prerequisites, and considerable expense. Off-site treatment generally requires repackaging into DOT approved containers. Repackaging can be very expensive. Handling and monitoring of waste prior to disposal is also a major operating cost. The Fernald Mobile Mixed Waste Stabilization Project has been performed as a CERCLA removal action, allowing on-site treatment with mobile equipment without either a RCRA part B permit or a Radioactive Material license. While the project was authorized by the Ohio EPA and the US EPA under CERCLA, the project Work Plan addressed all substantive requirements of a RCRA part B permit and was actually reviewed by the RCRA group within Ohio EPA and US EPA. This efficiency has allowed schedules to be moved forward and permitting expense to be avoided. The greatest enhancement is the ability to stabilize waste at the source, greatly reducing health & safety risk prior to shipment on the open highways. This type of project also allows much greater control of the treatment process which is appropriate considering generator liability. On-site treatment also allows better response to dealing with any abnormal situations related to characterization, fitness for treatment, container degradation and identity, and sampling. By estimate, the life-cycle cost of this project, compared to known off-site possibilities, is much less.

Demonstrate subcontractor ability to comply with DOE requirements with an Operational Readiness Assessment

One of the challenges of the Mobile Mixed Waste Stabilization Project has been addressing the impression that private treatment companies will not be able to work under DOE rules. For this project, we demonstrated compliance with DOE requirements by subjecting the project to an independent Operational Readiness Assessment (ORA). These ORAs are by nature quite a challenge for participants on all sides of the assessment, but acceptable findings and a recommendation to proceed do much to bolster DOE confidence that contractors can perform safely as advertised.

Develop planning documents to level 5 detail to avoid regulatory delays

The Mobile Mixed Waste Stabilization Project is pursuing an aggressive schedule and has relied on prompt attention by regulating bodies on

several occasions. One of our approaches to minimizing review cycles and potential delays has been to provide extraordinary detail about our plans for the project, right up front, to Ohio EPA and US EPA to minimize their asking for more detail later. Initially, we developed a work breakdown structure expanded down to level 5 activities. We then defined our plan of action and schedule down to this level. The challenge in doing this was keeping our aggressive schedule while developing this level of detail.

The contract was issued to Perma-Fix on May 30, 1995. The contract schedule called for delivery of planning deliverables to FERMCO by July 5, 1995. In the following 3 weeks, as a result of teamwork between Perma-Fix, FERMCO, and DOE, the project submitted a Work Plan to Ohio EPA and US EPA. This plan was conditionally approved without revision on September 28, 1995.

DOE and FERMCO have spent a great deal of energy in keeping Ohio EPA and US EPA abreast of upcoming activities, and they have been characteristically prompt in approving clean-up and removal actions.

Prepare good waste characterization including ample sampling, traceable documentation, and real time radiography to get good fixed-price quotations and regulatory buy-in to proposed plans

DOE has been working on an initiative to increase fixed-price contracting in an effort to improve cost performance on the nations tax base. The problem is getting fixed-price quotations for waste treatment services on the basis of process knowledge accrued under the security of the cold war. Proceeding on the basis of this approach is fairly unrealistic. Equally important is securing regulatory buy-in to planned activities rooted in a firm knowledge about the contaminants of concern. Extensive characterization is required providing significant tangible evidence of the nature and extent of contaminants in the waste stream. The approach we took on the Mobile Mixed Waste Stabilization Project was to perform ample confirmatory sampling compared to process knowledge, prepare RTR films on a large population of the waste drums considered, and prepare a strategy for waste grouping that is traceable to container ID, sampling data, RTR, and process knowledge data.

Rumors of waste swell related to solidification have been greatly exaggerated

There is information that suggests that waste swell due to the addition of pozzolans to hazardous, mixed, and radioactive waste streams can be as much as 100% to 200%. These figures, as assumptions, have been used to estimate the disposed life-cycle cost of stabilization to be greater than other more complex technologies. Our experience on the Mobile Mixed Waste Stabilization Project has demonstrated waste swell less than 15% as a result of existing void space and overpacks in the incoming waste drums. There are situations that would result in waste swell in multiples of original volume, but our project results challenge some life-cycle cost claims that are being made. A key consideration is the compressive strength desired. Producing a low strength grout minimizes volumetric swell due to added reagents.

Carefully select Source Evaluation Board (SEB) to level playing field for proposals

Evaluation criteria are usually established by the SEB to provide equal evaluation for all proposers. However, there is still a normal bias for the known, the big, and the powerful. This is perfectly logical in the sense that if they are big and powerful they must be doing things right,

have a talented work force, and have faced and solved many problems. However, in an attempt to support small business participation, DOE has set goals for subcontracting with small business. This suggests SEB's need to be equipped with more knowledge about proposers than is available in the Wall Street Journal and special interest trade publications. Major corporations were included in an unrestricted competition for this important project. However, a small business was selected and has performed quite well.

Evaluate subcontractor ability to decontaminate and remove equipment at contract completion

In general, the extent of the government's liability for a project such as the Mobile Mixed Waste Stabilization Project is the price of services, plus the fair market value of contaminated equipment. This cost is perhaps complicated by the disposal cost of equipment as radioactive or mixed waste. With this in mind there are a few things that contractors can do that can have the enhancing effect of limiting the government's risk. These are keeping the treatment scheme no more complicated than necessary, using as much simple and commercially available equipment as possible, keeping any disposal volume as small as possible, bolstering your proposal with a plan for decontamination, and generally describing corporate plans for released equipment after project completion.

Have DOE's contractor provide HEPA ventilation meeting subcontractors needs because of DOE safety rules.

There are a number of practices and validation tests used on DOE sites that are not generally available to the private sector. For the Mobile Mixed Waste Stabilization Project, we intended for the selected subcontractor to size and provide HEPA ventilation adequate for the project. After the units were purchased, delivered, and ductwork installation was underway, we discovered that site practice required di-octyl-phthalate (DOP) testing by DOE in Oak Ridge for each filter element used. Given the schedule constraints we faced, this was not possible. The site had a number of surplus HEPA filtration units with in-stock filters which met the requirements. In the end, FERMCO provided the needed HEPA units meeting subcontractor requirements. Given site rules for monitoring, filter change, sampling, record keeping, and filter testing, it makes more sense for DOE's contractor to support this requirement rather than passing it on to subcontractors.

Consider weight, handling, and disposal criteria of the disposal facility in selection of waste containers

For the Mobile Mixed Waste Stabilization Project, FERMCO selected and purchased white metal boxes. These boxes are roughly half as high as conventional B-25 boxes, are made with an integral skid for fork truck handling, and are designed for maximum load-bearing requirements based on an empty container. These containers weigh approximately 6000 pounds when filled with the solid waste form. These features have been chosen to satisfy the waste acceptance criteria of the NTS.

Integrating participation of subcontractors in weekly project meetings with site interface groups

Early after award of the project subcontract, Perma-fix began attending the weekly project meetings. The result was that much planning and other project information was made available to site groups earlier, and by the time equipment started showing up, Perma-fix was well integrated into the site team and could address and resolve issues with the site groups

involved. Meeting the project's schedule would not have been possible otherwise.

Select a subcontractor who demonstrates an ability to live and operate within DOE's "culture"

When the SEB evaluated submitted bids for the Mobile Mixed Waste Stabilization Project, most of the bidders could demonstrate that they had stabilized hazardous or radioactive waste. What became impressive to the SEB were proposals that demonstrated they understood and could operate within DOE's culture of security, site access, directives, standards, guides, and procedures. Perma-Fix demonstrated this capability by partnering the project with PDC, a company having extensive experience in DOE culture. PDC has a very strong and successful background working with contractors in support of DOE initiatives. Perma-Fix and PDC composed a project team that identified needed project positions and personnel from both companies. The proposed project team had demonstrated qualifications and experience to perform project functions on a DOE site.

Integrate the disposal site early in the process identifying their needs and implementing systems to meet them (i.e., a Process Control Plan, Nuclear Quality Assurance Program, ample documentation, and traceability)

Mixed waste treatment has a limited purpose unless there is a subsequent strategy to dispose of the treated waste. For DOE waste, there is not a less expensive option than acceptable disposal at the Nevada Test Site. NTS was developed and operated to handle radioactive waste, not mixed waste. A fairly new version of the NTS Waste Acceptance Criteria (NVO-325 Rev. 1) allows disposal of mixed waste where the hazardous characteristic has been removed. Until now, this feature has not been extensively used and operating practices at NTS are still in the developmental phase.

The key to paving the way for NTS acceptance of Fernald mixed waste has been to develop a site program for characterizing, identifying, containerizing, confirmation sampling, documenting and transporting waste products acceptable to NTS. For FERMCO's subcontractor, Perma-Fix, this included preparation of a Process Control Plan, demonstration of a Nuclear Quality Assurance Program, and a project-specific Sampling Plan responsive to FERMCO's agreement with NTS.

In conjunction with the Mobile Mixed Waste Stabilization Project, FERMCO has focused on keeping NTS abreast of project plans, conducted Fernald site visits for NTS participants, worked through closure of NTS audit findings, and supported NTS inquiries for mixed waste treatment. These actions have paved the way for disposal of the projects treated mixed waste at the NTS facility.

Keep in close contact with regulators (i.e., DOE, Ohio EPA, US EPA, and NTS)

None of the responsive behavior that has been demonstrated by DOE, Ohio EPA, US EPA, and NTS would have been possible without an informational network that allowed these participants to keep up with project plans and schedules. Having information available and flowing has contributed to creating team work among DOE, FERMCO, Perma-Fix, Ohio EPA, US EPA, and NTS participants.

Plan for a realistic schedule considering the limitations of the site infrastructure

Contractors might make big claims about how much waste per unit time they have processed on superfund sites, but none of that really matters for a DOE on-site treatment. Many site groups interface with project

activities, and sooner or later site infrastructure is going to limit processing capability. For this project, the Request for Proposal defined what the desired daily processing rate would be. Even so, several proposals were submitted on the basis of much greater processing ability.

Temper the zeal to mix waste streams to minimize volume with the need to have traceability of the waste and accountability of Special Nuclear Material (SNM)

DOE Orders require FERMCO to retain accountability of SNM on site. For the Mobile Mixed Waste Stabilization Project accountability, this has meant that we devise a scheme to track isotopic balances of SNM in project waste streams prior to, during, and after processing by Perma-Fix. This need has driven the procedural systems developed for the project, and information taken and managed at various stages during treatment operations.

The complexity of this task caused us to carefully define and understand the waste contents programed for different treatment episodes. We also decided not to mix some waste streams that could have further minimized the final disposed volume.

Start with the easier material first

The waste streams in the Fernald Mobile Mixed Waste Stabilization Project had considerable variation. Some were expected to be more difficult to successfully stabilize than others. There was a strong temptation to require the processing of the most-difficult-to-treat waste stream first to allow the maximum of time for the toughest challenge. We wisely chose however, to start with the waste stream expected to provide the minimum challenge for treatment. This allowed the best possible set of conditions for the project team to get: 1) all equipment functioning properly, 2) all organizational and process interfaces defined, 3) all project procedures improved, 4) operating data, 5) on-line air monitoring established, and 6) personal protective equipment evaluated.

Closely work the interface with the Federal Facility Compliance Act

In the early project planning stage, there was a concerted effort to identify and recognize the participants and stakeholders in the project under the Fernald Federal Facilities Agreement. The master plan for the project was established recognizing the interfaces and coordination necessary to satisfy site needs and commitments. The pre-planning and close coordination of project services with site commitments proved very effective in minimizing stops and starts and promoting project integration and efficiency.

CONCLUSION

We believe that the Fernald Mixed Waste Stabilization Project demonstrates that technically challenging mixed waste treatment projects can be successfully performed on DOE sites, within the DOE culture, by small business commercial waste treatment contractors. To be successful, the small business team must combine extensive waste treatment knowledge and experience with DOE site know-how. The successful small business team must be willing to adapt to the formalized operational environment of a DOE site, while the site prime contractor and DOE site office must recognize and capitalize on the innovative approaches from the commercial sector.

This project has shown that stabilization/solidification processes have many performance advantages over competing technologies when dealing with waste streams having highly variable chemical and physical properties, such as DOE site legacy mixed wastes. This project has provided evidence

that waste volume increases commonly attributed to cement-based waste stabilization processes are greatly exaggerated. Waste management planning decisions based on assumptions of 100 to 200 percent swell should be critically reviewed. DOE may be able to greatly reduce final treatment costs and speed up legacy waste treatment and disposal schedules by taking advantage of this inexpensive and proven technology. For highly variable wastes, the judgment of properly trained, supervised, and experienced operators can be a better recipe for success than multi-million dollar automated machines.

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MOBILE SYSTEMS - A CHEAPER AND FASTER METHOD OF WASTE CHARACTERIZATION

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ABSTRACT

A mobile waste characterization systems analysis was performed by the Carlsbad Area Office National Transuranic Program, Department of Energy (DOE). The analysis identified and evaluated mobile system parameters required for input to a National TRU Waste System Model-a computer simulation specifically designed to evaluate options related to the characterization and transportation the of transuranic (TRU) waste from generator site locations throughout the DOE complex to final disposal at the Waste Isolation Pilot Plant (WIPP). Numerical estimates of these parameters were obtained from a sampling of mobile waste characterization systems currently available for lease or purchase within the DOE complex and the private sector. The analysis addressed the utility and feasibility of mobile systems to characterize contact-handled (CH) and remote-handled (RH) TRU waste stored in a variety of containers including drums and boxes. Two baseline mobile system configurations were identified as necessary and sufficient to address the transportation and disposal of CH-TRU waste. Preliminary results obtained from a spreadsheet analysis indicated cost savings could be realized when these baseline mobile systems are used in conjunction with fixed waste characterization facilities.

INTRODUCTION

Evolving regulatory requirements, a shrinking budget, and a desire to accelerate schedules have resulted in the DOE's examination of alternative resources to supplement its current waste characterization capabilities. The utilization of mobile and/or portable waste characterization systems is one of these alternatives. Mobile systems are self contained and self sufficient units capable of independent field

operation. Portable systems require the host site to provide some support functions consisting of containment facilities and/or utilities. Mobile/portable systems offer the option of providing supplemental assistance to major sites having limited fixed system capability and/or major support as in the case of small quantity sites (SQSs) having little or no waste characterization capability. When compared to fixed systems, mobile/portable systems offer several advantages including their mobility (ideally suited for field operations at major sites and/or transportation to numerous small quantity sites), modular design (capable of being easily reconfigured to address changing conditions and requirements), reduced capital costs (suitable for facilitating small business privatization of services), and reduced time between placement of a purchase order and system delivery.

WASTE CHARACTERIZATION METHODOLOGIES

The methodologies approved by the TRU Waste Characterization Quality Assurance Program Plan (QAPP) to address the requirements for disposal at the WIPP include:

- Nondestructive assay (NDA)
- Nondestructive examination (NDE)
- Headspace gas sampling and analysis
- Visual Examination
- Homogeneous waste sampling and analysis

A subset of these methodologies is considered necessary and sufficient to address the waste characterization requirements for transportation between generator/storage sites. This subset consists of:

- NDA
- NDE
- Headspace gas sampling and analysis.

WASTE CHARACTERIZATION SYSTEMS

For purposes of configuring the smallest core assembly of mobile waste characterization systems, the three methodologies identified above as necessary and sufficient for meeting the requirements for transportation are the preferred candidates. This core assembly of three mobile systems is referred to in the systems analysis as the baseline transportation configuration (BTC) or baseline transportation fleet. The larger core assembly of mobile waste characterization systems, which includes the necessary and sufficient characterization methodologies for purposes of disposal, includes all five of the methodologies listed above. This core assembly of five mobile systems is referred to in the systems analysis as the baseline disposal configuration (BDC) or baseline disposal fleet.

In recognition of the variability of the waste matrices and the variety of packaging types (drums, boxes) used to contain TRU waste, several additional systems have been identified to support the baseline transportation and disposal waste characterization core assemblies. These supporting or secondary systems provide the following functions:

- Drum venting and filter insertion
- TRUPACT-II loading/unloading
- Data integration and compression
- Box repackaging
- Size reduction (compaction, shredding)

To illustrate the flexibility in tailoring an assembly of mobile systems to address a specific problem, consider the example where a SQS desires to dispose of its CH-TRU waste contained in non-vented 55-gallon drums. Assuming all elements of its certification program have been approved,

the preferred mobile waste characterization system configuration in this case would consist of the BDC in addition to a drum venting and filter insertion system plus the TRUPACT-II mobile loading unit-a minimum of seven trailers to facilitate transport and disposal of the waste at the WIPP. (Note: The TRUPACT-II is a TRU waste shipping package.) If instead, it was deemed necessary to ship the waste to a major site for purposes of treating the waste to meet the WIPP waste acceptance criteria (WAC), the preferred waste characterization configuration would consist of the BTC in addition to the two ancillary systems used in the preceding scenario - a minimum of five trailers to facilitate transport to a site other than the WIPP.

Fixed System Capabilities

Information pertaining to the generator/storage sites and their waste characterization resources was obtained from several draft documents including the CH-TRU Waste Management Data Packages (August 1995), the CH-TRU Waste Characterization System Analysis (August 1995), and the RH-TRU Waste Facility Characterization Capability Assessment (August 1995). This data provides a compendium of information for each CH-TRU and RH-TRU waste management activity at the generator/storage sites.

A review of the above listed resources shows that only three sites currently possess the entire suite of five methodologies to characterize CH-TRU waste contained in 55-gallon drums. These sites include the Idaho National Engineering Laboratory (INEL), the Los Alamos National Laboratory (LANL), and the Rocky Flats Environmental Technology Site (RFETS). From a DOE complex wide standpoint, there is presently a limited capacity to perform waste characterization to ensure a sufficient quantity of certified CH-TRU waste will be staged and ready for disposal upon the opening of the WIPP in 1998.

Analysis of the collected data shows that no site within the DOE complex is presently capable of characterizing (using all five methodologies) CH-TRU waste contained in boxes. This result prompts the question of whether it is more efficient and effective to repackage all the boxed waste into drums or build the prerequisite fixed and/or mobile box waste characterization systems. Data analysis also discloses skid-mounted waste characterization systems, designed especially for operation within hot cells, offers an option for using portable systems to complement the existing and planned RH-TRU waste characterization capabilities site wide. Lastly, preliminary data received from a survey of SQSs indicate little or no capability to characterize waste for either transportation or disposal purposes.

Prompted in part by the gaps in the generator sites' capabilities to characterize waste for purposes of transportation and disposal, a need to evaluate alternatives for ensuring a sufficient flow of certifiable waste to the WIPP for disposal in the near term became apparent. As will be shown, the use of mobile waste characterization systems capable of addressing CH-TRU waste contained in 55-gallon drums offers the most cost-effective and time-efficient methods for characterizing the waste at the generator/storage sites in the near term.

Mobile System Capabilities (CH-TRU Waste)

The mobile system parameters required for input to the National TRU Waste System Model include the following quantities:

- Fielding time
- Throughput
- Capital cost

Operating cost
Maintenance cost
Set-up/breakdown time
Life cycle

Numerical estimates of these parameters were obtained from both the commercial and government sectors via a survey. The accompanying Table summarizes the results of that survey. Except for mobile box repackaging systems, all 13 of the mobile systems identified in the Table are available for purchase and/or lease and are capable of addressing the characterization of CH-TRU waste contained in 55-gallon drums.

In some instances the numbers obtained from the survey participants represent best estimates. Those mobile system parameters having the greatest variability relative to fixed system parameters include fielding times and capital costs. The following points of clarification are provided regarding the tabulated data:

All operating costs pertain exclusively to the mobile system and neglect all site-related functions, including oversight authority, certification authority, training, waste transportation (from storage to staging area), health physics, radiation safety, safeguards, etc. The role of site personnel in the operation of the system is determined by several factors. These factors include whether union involvement is mandated, whether cognizant engineers are required for data interpretation, and whether the waste characterization methodology is intrusive or nonintrusive.

Capital costs do not encompass ancillary documentation, including a health and safety plan, sampling and analysis plan, standard operating procedures, National Environmental Policy Act (NEPA) documentation, Safety Analysis Report (SAR), Quality Assurance Project Plans (QAPjPs), etc.

The majority of the mobile systems addressed in the table are not available off the shelf; rather, these systems are built on an as-needed basis. Consequently, manufacturing costs are best estimates due to limited production experience. The costs associated with field deployment of the systems are a function of several variables, including the number of operating personnel, anticipated maintenance and repair costs, transportation costs, rate of amortization on investment, training, calibration, etc.

Not included in the set-up and breakdown times are other factors that add significantly to the time required for accessing or departing a generator site. These factors include system inspection, decontamination (if applicable), administrative processing, approvals, etc.

The identified drum venting system uses an enclosure that completely surrounds the drum.

Data integration and compression are a consequence of the large volume of data associated with digital techniques typically used in NDA and NDE applications. Data fusion techniques using artificial intelligence serve to emphasize the need for a dedicated data integration and compression system.

Mobile box repackaging systems, although feasible in principle, are not practical due to the large trailer size required for housing the glove box. The use of tandem (side-by-side) trailers has potential liability due to the risk of contamination when coupling and uncoupling the two systems.

A decontamination unit is required if the mobile systems are to be self contained and capable of responding to a contamination occurrence.

Other data in the Table requiring edification are the quoted processing rates for several of the mobile systems. In particular the processing rate for the radiometric examination system needs to be qualified. For clarity, examples of radiometric examination include real time radiography (RTR), digital radiography (DR), and computed tomography (CT). Although RTR is the most prevalent method employed throughout the DOE complex, other methodologies are emerging and finding application at several of the generator sites.

Table I

Digital radiography has been incorporated in the Waste Receiving and Process facility at Hanford, and CT is being explored for use at the INEL. An examination of these varied methodologies and their associated operating procedures shows that a wide range of throughputs are feasible depending on several assumptions-the most important being the waste form under interrogation. In brief, there is no single value for throughput that is representative of the varied methodologies and waste forms found across the DOE complex. Resolution of this problem is best achieved by bounding the throughput, at least in the case of NDE and NDA. The other waste characterization methodologies and their associated throughputs reported in the Table are considered reasonable based on previous field experience.

For NDE systems, the lower bound, upper bound, and mean throughputs are estimated to be as follows:

Lower bound 4 minutes per drum (using DR)

Upper bound 2 hours per drum (using CT @ 2 mm per slice and assuming a full drum)

Mean = 20 minutes per drum (using RTR and assuming a full drum)

For NDA (gamma-ray and neutron) systems, the throughput is a function of the waste matrix and counting statistics (signal-to-noise ratio). Consequently, only the mean throughput rates are listed in the Table.

Using the data contained in this Table as input to the National TRU Waste System Model, output resulting from the computer simulations will provide information on a variety of system configurations as measured against related waste work-off schedules, costs, and throughput rates. Using various serial and parallel combinations of fixed and mobile systems, it is possible to arrive at the optimum TRU waste system configuration.

In comparison to stationary systems, which may take twelve or more years between concept and final construction of new facilities, the time to field mobile systems is relatively short. Capital costs are also modest in comparison with stationary systems. Processing rates for mobile and fixed systems are about the same, depending on the assumptions made.

Mobile System Capabilities (RH-TRU Waste)

Limited experience exists in the arena of RH-TRU waste characterization. This situation is due in part to the fact that quality assurance objectives (QAOs) are still in the process of being defined for RH-TRU waste data measurements. Without knowing a priori these performance requirements, it is impossible to design and develop new instrumentation or processes to characterize this waste category. Consequently, when polling potential suppliers to identify their capabilities for developing systems able to characterize RH-TRU waste, their responses usually return as a question (e.g., "What do you want?").

The minimum generic requirements for RH-TRU waste characterization include the following:

Dedicated/controlled areas and laboratories (either hot cells, special facilities, or remote field locations)

Robotics (designed to interface with each of the waste characterization methodologies). Waste handling involving 55-gallon drums and sample handling involving core specimens necessitate the need for two or more unique robot systems

Shielding (to be provided by either hot cells, portable constructions, or natural [earth] constructions)

Dedicated systems (the requirements for the use of dedicated/controlled areas, laboratories, and the use of shielding necessitate the use of two independent sets of waste characterization systems to address CH-TRU and RH-TRU waste)

Present commercially available technology using DR and/or CT facilitate the NDE of RH-TRU waste. Features such as background subtraction allow for the enhancement of the signal-to-noise ratio when working within intense background radiation fields. Image processing and enhancement can be used to provide automated pattern recognition of objects using a stored library of attributes associated with a set of preselected objects. Penetration of shielded containers is not a problem when a linear accelerator is used in place of the standard X-ray generators. In principle, NDE can be used for the characterization of RH-TRU waste, whether shielded or unshielded, up to the 1000 rem/hr limit allowed by the WIPP WAC. Although practical experience already exists in the NDE of fuel rod elements and tank waste core samples having extremely high dose rates, the application of NDE technology to RH-TRU waste has not been demonstrated. A technology demonstration to show that RH canisters and casks can be examined nondestructively constitutes the first step in validating the application of NDE to RH-TRU waste.

Though presently not one of the approved waste characterization methods listed in the QAPP, the use of X-ray CT also provides an alternative for performing visual examination of RH-TRU waste. This approach avoids the costly and time-consuming process of using a hot cell for unpackaging, inspecting, and repackaging of the waste. Image reconstruction using CT data has been demonstrated capable of identifying objects as small as a dime when buried within a simulated waste matrix consisting of portland cement.

In the case of the radioassay of RH-TRU waste, many of the problems identified as germane to CH-TRU waste will also apply to RH-TRU waste (i.e., nonideal matrix and source distributions in addition to self-absorption, self-shielding, and matrix shielding). The increased background radiation fields, however, will only serve to further complicate the measurement of RH-TRU waste. Discussions with NDA instrument developers and the review of published papers indicate the method of choice in the presence of high gamma fields (usually from fission and activation products) is one that uses neutron detection. Even this approach, however, has limitations as determined by the helium-3 tubes used in the process. Above 1 rem/hr, the performance of these tubes is no longer reliable. This value is a factor of 1000 below the upper allowed dose rate for RH-TRU waste. Ongoing research and development is expected to extend the operating limits of the tubes to 100 rem/hr within the next few years. In summary, the greatest risk in the process of characterizing RH-TRU waste appears to be the ability of NDA to comply

with applicable QAOs-assuming they are similar to the QAOs used for CH-TRU waste.

CONCLUSIONS

This analysis has shown the feasibility of using mobile systems to characterize CH-TRU waste contained in 55-gallon drums for purposes of transportation and disposal. The application of mobile systems to characterize boxed CH-TRU waste, although feasible in principle, is not practical based on the multitude of box sizes employed and the associated limitations of scale in performing NDA. Portable systems have application to RH-TRU waste due to the preference that all waste characterization measurements be performed within hot cells. Other limiting factors to RH-TRU waste characterization includes NDA.

With the opening of the WIPP rapidly approaching, this analysis has demonstrated the need for the establishment of a National Mobile Waste Characterization Program. The initial function of this program would be to identify the preferred method(s) for acquiring these systems (purchase, lease, or service contract), prepare solicitations, evaluate solicitations, make awards, and oversee the manufacturing of the systems. Before any mobile or portable waste characterization systems can be used in conducting a campaign across the DOE complex, sufficient time must be allotted for the systems to go through a rigorous environmental, safety, and health evaluation to identify all regulatory requirements. One such evaluation has already been performed by the LANL, and the results of that investigation constitute the first step in constructing a generic template for this process.

The data and results obtained by this analysis will be fed into the National TRU Waste System Model. Output from this model is expected to result in the identification of the optimum fixed/mobile system configuration(s) for the development of a comprehensive waste characterization strategy and reveal alternative opportunities for reducing cost and compressing the schedule for the preparation and flow of TRU waste.

ACKNOWLEDGMENTS

Technical input to the analysis was obtained from numerous sources, including the commercial sector, government laboratories, waste generator/storage sites, and support contractors.

Processing and final preparation of this report was performed by the Waste Isolation Pilot Plant Management and Operating Contractor for the U.S. Department of Energy Under Contract No. DE-AC04-86AL31950.

Session 16 -- INTERNATIONAL EXPERIENCE IN LLW DISPOSAL: ASSESSMENT/PERFORMANCE

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16-1

A SYSTEMATIC EVALUATION OF RADIONUCLIDES CONTRIBUTING TO THE PERFORMANCE ASSESSMENT OF A LLRW DISPOSAL FACILITY

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ABSTRACT
The development of a low-level radioactive waste (LLRW) disposal facility requires that a performance assessment be done to ensure that the effect of the LLRW on the surrounding environment is within regulatory limits. One of these assessments is the evaluation of the long-term performance assessment which typically includes the time frame from approximately when the facility stops receiving waste to a time in the distant future. This long-term performance assessment is usually dominated by a groundwater transport pathway. Because a typical LLRW disposal facility will contain in excess of 100 radionuclides that need to evaluated, a detailed groundwater transport evaluation of all the radionuclides in the LLRW disposal facility can be quite cumbersome. A systematic screening methodology is developed that considers the radionuclide half-life, daughter products, and partition coefficient between soil and water. The result for the site evaluated is that only 15 radionuclides (including daughter products) remain that require detailed groundwater transport evaluation. In addition, an inventory limit can be proposed on only 7 radionuclides and have confidence that the dose from the long term performance assessment will remain within regulatory limits.
INTRODUCTION
The licensing of a low-level radioactive waste (LLRW) disposal facility requires that a performance assessment be done to ensure that the effect of the LLRW on the surrounding environment is within regulatory limits. One of these assessments is the evaluation of the long-term performance assessment which typically includes the time frame from approximately when the facility stops receiving waste to a time in the distant future. The time in the distant future has been taken to be a period long enough that other large scale environmental effects could occur, such as weather variations and associated glaciations, and the environmental effect of the LLRW would be insignificant in comparison.
The long term performance assessment is typically initiated by evaluating the waste stream that is expected to be received at the facility. Evaluation of this waste stream typically results in excess of 100 radionuclides being present. A detailed evaluation of this large number of radionuclides can be quite cumbersome. In addition, the evaluation usually finds that only a small number of the radionuclides make a significant contribution to the performance assessment. During performance assessments, a number of the radionuclides are often evaluated individually and eliminated from further consideration. This paper presents a systematic criteria and methodology for determining the radionuclides that make a significant contribution to the long-term performance assessment of a LLRW disposal facility. Table I at the end of the paper provides an example matrix for a limited number of nuclides that can be used for this systematic screening.
Table I
SOURCE TERM
In order to evaluate the environmental effect of the LLRW disposal facility, an estimate of the waste to be received must be evaluated. A reasonable estimate of the future waste stream may be determined by evaluating the historical waste stream in the region the disposal facility is to serve. This information is readily available from the U. S. D.O.E. National Low-Level Waste Management Program, Manifest Information System (MIMS) that retains all commercial LLRW shipping manifest information. The information includes the radionuclides,

activity, and volume of LLRW shipped by waste generator, by state the generator is in, and by year of shipment. (The identity of the waste generator is not readily available because generators are labeled only by identification number.) Reports are readily available from MIMS for the states and time periods of interest.

The Central Interstate Compact (CIC) LLRW disposal project obtained radionuclide, activity, and volume information from MIMS for the 7 year period from 1986 through and including 1992 for waste shipped from the five states within the CIC region. This information was averaged over the seven years to obtain an average annual average total volume and activity by radionuclide shipped for disposal. This annual activity by radionuclide is input to a computer code (SOURCE2) that solves the bateman equations and determines the 30 year facility inventory including daughter products.

The average waste volume during this 7 year period was approximately 87,000 ft3/y, which would produce approximately 2.6 million ft3 in 30 years if waste generation remained at this average value. The design basis for the facility is 5 million cubic feet, therefore, the 30 year radionuclide activity from MIMS and SOURCE2 was multiplied by the factor of 5/2.6 = 1.9 to provide a 5 million ft3 equivalent inventory in 30 years of operation.

EVALUATION OF GROUNDWATER TRANSPORT

The long-term performance assessment of a LLRW disposal site usually requires an evaluation of groundwater transport because when a site is properly closed this groundwater pathway is one of the few credible ones that remains. In order to evaluate the effect from groundwater transport of radionuclides, the disposal site must be well characterized and include a model of groundwater flow in the region, a local model of groundwater flow beneath the disposal site, and regional meteorology. These models are matched against each other to arrive at appropriate boundary conditions. The model should also include a description of the geological strata at the site that is usually obtained from borings. The application of these models will produce a groundwater transport time to the site boundary. This time duration is important because the ingestion of water from a well that is drilled at the site boundary often dominates the dose assessment for the long-term performance.

The evaluation groundwater transport at the CIC site yields a mean transport time to the site boundary of 1420 years, and a bounding case transport time (conservatively assuming rainfall persists at a near record value and a corresponding high water table) of 240 years. The bounding case transport time of 240 years is used in the evaluation to provide confidence that a conservative analysis has been performed.

ESTABLISHING CUTOFF CRITERIA BASED ON HALF-LIFE

Many radionuclides in a LLRW disposal facility have a half life that is sufficiently short that they will decay to an insignificant quantity prior to reaching the site boundary. An initial estimate of a 10 year life was chosen as being the minimum half life that would make a contribution to the dose. For a radionuclide with a 10 year half life, and a site boundary travel time of 1420 years or 142 half lives, the activity in the original inventory would decrease by a factor of 2142, or approximately 2E-43 of the original inventory remains. For the bounding case travel time of 240 years, the 10-year half-life represents 24 half-lives, or a decrease in the original inventory by a factor of 224 , or approximately 6E-8 of the original inventory remains. The remaining

inventory fraction from both of these travel times is sufficiently small that one can confidently predict that there will be an insignificant contribution from radionuclides with a half-life of 10 years or less, and they can be eliminated from further evaluation.

EVALUATION OF RETARDATION OF RADIONUCLIDES BASED ON Kd

An evaluation of the groundwater travel time to the site boundary was provided above. This is the time period for the groundwater and chemicals and radionuclides that remain dissolved in the water to reach the site boundary. However, many chemicals interact with the soil that the groundwater travels through and the result is a retardation in the transport of the radionuclides. The time period in which it takes a particular chemical or radionuclide to move through soil is determined by multiplying the groundwater travel time to the site boundary times a retardation factor. The chemical parameter that is used to evaluate the retardation factor is called the distribution coefficient, Kd, and is related to the retardation by the following expression.

$$R_i = 1 + K_{di} * \text{bulk density}_j / \text{porosity}_j$$

where R is the retardation factor, Kd is the distribution coefficient, the subscript I indicates the chemical of interest and j indicates the material that the chemical is traveling through.

Values of the bulk density and the porosity of the soil can be obtained from site specific measurements. Kd values can also be obtained from site specific measurements, or may be obtained from previous tabulations such as are provided in Ref. 1.

An example of the application of this methodology is provided below using cobalt as a LLRW nuclide and soil properties from the CIC site. From Ref. 1, the Kd of cobalt in clay is 550cm3/gm. A conservative evaluation of the CIC soil properties provides for a bulk density of 1.533 gm/cm3 and a porosity of 0.4, or a ratio of bulk density to porosity of 3.83 gm/cm3. This ratio times the Kd of 550 cm3/gm yields 2107, and the addition of the constant 1 is clearly insignificant. The result is that cobalt released into the groundwater and traveling through clay will arrive at the site boundary 2107 times the groundwater travel time to the site boundary. The conservative bounding travel time to the site boundary is 240 years, therefore the cobalt may be expected to reach the site boundary after 506,000 years under these conditions.

ESTABLISHING A CUTOFF CRITERIA BASED ON Kd

The methodology described above can also be used to back calculate a Kd that correlates to a particular travel time to the site boundary. The travel time is taken to be a period long enough that other large scale environmental effects could occur, such as weather variations and associated glaciations, and the environmental effect of the LLRW would be insignificant in comparison. A time duration of 10,000 years fits this criteria.

Given the soil conditions at the site and a limiting travel time of 10,000 years, it is then possible to derive a limit on the Kd above which the radionuclides will not reach the site boundary in 10,00 years. The retardation factor is 10,000/240 = 41.7. Rearranging the expression for Kd above, and taking the constant 1 as an insignificant contributor, the result is

$$\begin{aligned} Kd &= R / (\text{bulk density}/\text{porosity}) \\ &= 41.7/(3.83) \\ &= 11 \end{aligned}$$

Therefore, one can establish a limiting Kd value of 11, and radionuclides with a Kd of greater than this value are not be expected to reach the site boundary within 10,000 years even under the conservative bounding conditions that would produce a 240 year groundwater travel time to the site boundary.

CONSIDERATION OF DAUGHTER PRODUCTS

An additional consideration that must be made is the contribution daughter products may make to the radionuclide concentration at the site boundary well. Daughter products are a different chemical species than the parent and therefore the Kd consequent transport properties would be different from the parent. These daughter products are treated in two different fashions. If the parent - daughter relation is not part of a continuing decay chain, and the half life of both radionuclides is short (10 years or less), neither the parent nor daughter will make a significant contribution at the site boundary. An example of this is zirconium-95. This radionuclide has a 64-day half-life and has a daughter product that is niobium-95, which has a half-life of 35 days.

If the daughters are part of a longer decay chain, the chain is reviewed more carefully. If the parent and all daughters in the chain have a large Kd (>11), the radionuclides do not make a significant contribution for the same reason that other radionuclides are eliminated on the basis of Kd. If one of the daughters has a small Kd (<11), then this daughter and all subsequent decay daughters are included in the evaluation. The subsequent daughters are included because they could arrive at the site boundary by being transported by a small Kd parent.

SUMMARY OF REMAINING RADIONUCLIDES

For the CIC project, the only radionuclides that remain after performing screening elimination described above are hydrogen-3, carbon-14, strontium-90, yttrium-90, technetium-99, iodine-129, americium-241 (and daughters neptunium-237, palladium-233, uranium-233, thorium-229, radium-225, actinium-225, francium-221, astatine-217, bismuth-213, polonium-213, thallium-209, and lead-209), and americium-243 (and daughter neptunium-239).

It is interesting to note that of these radionuclides, hydrogen-3, carbon-14, technetium-99, and iodine-129 are required to be listed on the shipment manifest in accordance with 10CFR20 Appendix F. Although common literature values (1) of Kd for strontium and americium would eliminate these radionuclides, a site specific measurement indicated that Kd for these radionuclides may be within the criteria and therefore they were retained. The strontium-90 daughter, yttrium-90, is retained because of the possibility of being transported by strontium-90. The americium daughters were retained because of the possibility of being transported by the parent americium.

A review of the inventory and dose effects indicated that the contribution from actinium-225 is negligible, and therefore contribution from the nuclides francium-221, astatine-217, bismuth-213, polonium-213, thallium-209, and lead-209 is also negligible. Therefore, these radionuclides were not evaluated further.

A long term performance assessment can then be performed using the inventory for these 15 radionuclides.

This methodology provides a systematic approach to determine a set of radionuclides that can be used in performing a long-term performance assessment. The approach greatly simplifies the actual evaluation of the dose assessment by justifiably reducing the number of radionuclides for

which a dose assessment needs to be performed. By restricting the dose assessment to the contributing radionuclides, resources are appropriately focused on evaluating and understanding the dose contribution from these significant contributors.
An additional use that can be made of the methodology and subsequent dose assessment is to determine a maximum radionuclide inventory that can be disposed and remain within regulatory limits. Of the 15 remaining radionuclides, one is a daughter of strontium-90, six are daughters of americium-241, and one is a daughter of americium-243. Because the inventory of the daughters is determined by the inventory of the parents, the inventory limit need only be established for the parents. Therefore, an inventory limit could be established for only seven radionuclides (hydrogen-3, carbon-14, strontium-90, technitium-99, iodine-129, americium-241, and americium-243) and have confidence about the resulting dose assessment. One could arbitrarily allocate a 1 mrem/yr dose to each of the seven radionuclides (including dose contribution from daughters) and then determine the radionuclide inventory in the LLRW disposal facility that would create this 1 mrem for each of these seven controlling radionuclides. This calculated inventory for the seven controlling radionuclides will result in a dose that is well within the regulatory limit of 25 mrem/yr.
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16-2
EFFECT OF NATURAL BARRIER ON RESTRAINING THE MIGRATION OF RADIONUCLIDES FROM DISPOSAL FACILITY OF LOW LEVEL RADIOACTIVE WASTE
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ABSTRACT
Since disposal of low level radioactive waste from nuclear power plants in Japan began in 1992, the operation is proceeding favorably and approximately 50,000 of 200 -drums of waste have been disposed of in the Rokkasho Disposal Center up to May 1995.
The burial facility of the Rokkasho Disposal Center is equipped with a double cover soil layer. One is a bentonite/soil mixture that is less permeable than the surrounding soil and is installed at the side and the top of the burial pits against water permeation. The other is ordinary soil that is placed over the layer of the bentonite/soil mixture.
The Radioactive Waste Management Center has carried out a large scale tracer test for both layers under contract with the Science and Technology Agency from 1991 to 1995.

A large scale experimental facility equipped with the double cover soil layer, which is some meters wide and some meters long, was applied to this tracer test for the purpose of evaluating the performance of these natural soil barriers on restraining the migration of radionuclides. The following results were obtained by this test and the safety of the burial disposal was confirmed.

1. For ordinary soil cover, the following RFs (Retardation Factor) of imitative (nonradioactive) nuclides were calculated from the experimental data in the case of the 40 cm/day of groundwater velocity, and those convinced us that ordinary soil cover also has a certain barrier function.

RD of SR=67, RF of Co=121, Rf of Cs=107

2. Bentonite/soil mixture, which consisted of 10wt% bentonite and 90wt% sandy soil, reduced the velocity of groundwater to 1/1300 compared with that in the ordinary soil cover layer.

3. The behavior of radionuclides for long term was evaluated by using experimental data, and the following analytical results showed clearly the effectiveness of the bentonite/soil mixture.

a. In the case of the installation of the bentonite/soil mixture, migration of short-lived radionuclides such as Co-60 could be restrained remarkably and they would disappear by radiation decrement before release from the burial facility.

b. The maximum release rate of long-lived radionuclides such as Nb-94 or I-129 would be reduced by the factor of 1/1000 to 1/10000 and the time at the maximum release rate would also be delayed to 40,000 years for Nb-94 and 400 years for I-129 by installation of the bentonite/soil mixture.

BACKGROUND

Since disposal of low level radioactive waste from nuclear power plants in Japan began in 1992, the operation is proceeding favorably and approximately 50,000 of 200 -drums of waste have been disposed of in the Rokkasho Disposal Center up to May 1995.

One of the technical criteria in Japanese Regulations respecting disposal of low level radioactive waste established that a disposal facility should be covered with sand and/or soil, which is less permeable than the surrounding soil. In accordance with this criterion, the burial facility of the Rokkasho Disposal Center is equipped with a double cover soil layer. One is a bentonite/soil mixture layer (hereinafter "B/S mixture layer") that is less permeable than the surrounding soil and is installed at the side and the top of the burial pits against water permeation. The other is an ordinary soil layer that is placed over the layer of bentonite/soil mixture. Many disposal facilities of LLW in the world also have developed similar methods to inhibit water permeation to the facility and to restrain the migration of radioactive nuclides.

The Radioactive Waste Management Center in Japan has carried out a large scale tracer test for both layers (bentonite/soil mixture layer and ordinary soil layer) in the outdoor concrete vaults under contract with the Science and Technology Agency from 1991 to 1995.

The purpose of this tracer test is summarized in the three points that follow:

- to confirm that the bentonite/soil mixture layer has better ability to reduce the velocity of groundwater than the ordinary soil layer,
- to testify that both layers could restrain the migration of radionuclides,

to demonstrate the performance of a natural soil barrier and the safety of the disposal facility for the local government and dwellers in the vicinity of the disposal site.

EXPERIMENTAL METHOD

Experimental Facility

A large scale experimental facility equipped with a double cover soil layer was constructed for tracer tests. This facility has two concrete vaults. One for the test of the bentonite/soil mixture layer has a 6 m width, 4 m length and 7 m depth (Fig. 1a). The other for the test of the ordinary soil layer has a 6 m width, 6 m length and 6 m depth (Fig. 1b). Each vault was equipped with an intake and discharge water tank for giving a groundwater flow. Artificial precipitation equipment was installed at the top of each vault.

Fig. 1

Tracers

Table I shows the kinds of tracers and additive concentration. The injection of all tracers in a lump sum including bromine (Br-: tracer of water) was applied for the test of the bentonite/soil mixture layer. In the test of the ordinary soil layer, a constant and continuous injection was applied.

Experimental Condition

The specification of each soil layer and the experimental conditions are summarized in Table II. In the test of the ordinary soil layer, prior to the tracer test of imitative nuclides, bromine was injected singly to measure the velocity of groundwater.

Table I

Table II

EXPERIMENTAL RESULTS

The Velocity of Groundwater

The velocity of groundwater of each soil layer measured by the tracer test of bromine and by flow meter is summarized in Table III. By these experimental data, it was confirmed that the B/S mixture layer reduced the velocity of groundwater to 1/960~1/1300 compared with that in the ordinary soil layer.

Table III

Migration of Imitative Nuclides

Concentration in Groundwater

One tracer test using imitative nuclides in the bentonite/soil mixture layer was carried out for 908 days. In this test no imitative nuclides were detected at the two monitoring points which are 0.5 m and 1 m distance from the injection pipe.

The other tracer test using imitative nuclides in the ordinary soil layer was carried out for 797 days. In this test obvious migration of all imitative nuclides was observed, especially iodine and strontium, which were detected at the farthest monitoring well (No. 5 well 4.0 m from the injection point)(Fig. 1b). This means some of these tracers may go out to the discharge water tank through the ordinary soil layer. Though cobalt and cesium were also detected at the monitoring well, which are 0.5 m and 1 m from the injection point, the migration distance of these tracers was remarkably shorter than that of strontium.

Surface Concentration on Soil Layer

Four vertical cross sections describing the surface concentration contours of imitative nuclides in the B/S mixture layer are shown in Fig. 2. All imitative nuclides injected remained within the region of a few

centimeters from the injection pipe. This small migration distance was probably caused by the molecular diffusion of the tracer not by groundwater flow.
Three horizontal and three vertical cross sections describing the surface concentration contours of imitative nuclides in ordinary soil layer are shown in Fig. 3 and Fig. 4, respectively. And characteristics data summarizing the migration of the three imitative nuclides (Co, Cs, Sr) are summarized in Table IV. From Table IV obvious migration of these imitative nuclides was recognized. On the other hand, we confirmed that Co, Cs, Sr were absorbed on the surface of the soil in high concentration and their migration in the ordinary soil layer was certainly restrained. This result convinced us that the ordinary soil layer also had a certain barrier function.
Fig. 2
Fig. 3
Fig. 4
Table IV
Discussion about Mass Balance
Table V shows the mass balance of the three imitative nuclides in both soil layers. In these tracer tests, favoritable mass balance was obtained. It means that each tracer test had been carried out successfully and the concentration of the imitative nuclides in the water and on the soil was measured correctly.
Table V
Long-Term Evaluation of the Effect of the Bentonite/Sand Mixture Layer
We predicted the migration of radionuclides near the disposal facility, and assessed the effect of the B/S mixture layer as natural barriers.
Condition
As to shape and dimension of the disposal facility, the only parameters necessary to calculate are assumed, and we evaluated the effect of the B/S mixture by comparing the time dependence of the normalized fluxes for the following two cases, whose models are shown in Fig. 5
  case-1: with the B/S mixture layer
  case-2: without the B/S mixture layer
Fig. 5
Normalized flux (hereinafter 'NF') is defined by Eq. 1, so its dimension is meter/second (= (Bq/s/m2)/(Bq/m3)).
Eq. 1
We compared each flux at a downstream distance of two meters from the end of the disposal facility because that point is the border of the B/S mixture layer and the ordinary cover soil for case-1. Calculation is implemented for five radionuclides (Co-60, Cs-137, Sr-90, Nb-94 and I-129) and by using one-dimensional mass migration code that takes account of radioactive decay. Those radionuclides are assumed to be reduced by disintegration and release from the disposal facility. In the B/S mixture layer, migration by diffusion is predominant.
Parameters
Parameters for migration prediction, which are mainly gained by complemental laboratory tests, are shown in Table VI. In addition, groundwater velocity in the normal cover soil gained by the field validation test is so fast that we assume two values in the calculation prediction: one is an experimental value and the other is two orders lower.
Table VI

Results

Calculation results, shown in Fig. 6 except for Co-60 are as follows.

Co-60: Since water flow is very low in the B/S mixture layer because of diffusion predominant in the area, those nuclides whose half lives are lower than several years such as Co-60 decay to a negligible order during migration in the B/S mixture layer in case-1.

Cs-137: Those nuclides whose half lives are tens of years but whose Kd's are large enough such as Cs-137 decay to a negligible order by the sorption function of the B/S mixture layer at the evaluation point in case-1.

Fig. 6

Sr-90 and Nb-94: For such nuclides that have the following characteristics, the effect of the B/S mixture layer establishment is expected if the groundwater velocity in the ordinary cover soil is typical for repository site condition.

a. whose half lives are several tens of years and whose Kd's are not so large such as Sr-90

b. whose half lives are very long but whose Kd's are large enough such as Nb-94

I-129: Nuclides of very long half life and small Kd such as I-129 are all released from the facility within a finite period in the case of "without the B/S mixture layer (case-2)" where the flow rate is the larger, so NF of case-1 (with the B/S mixture layer) is greater than that of case-2 after that time as a calculation result. We interpret that the effect of the B/S mixture layer establishment is expected because the absolute value is small enough if the water velocity in the normal cover soil layer is typical for a repository site condition.

In the case of short-half-life nuclides, the release rate out of the B/S mixture layer is reduced drastically because of the reduction of the flow rate and the sorption property by the B/S mixture layer. In the case of long-half-life nuclides, the establishment of the B/S mixture layer is suggested to be effective from the point of Normalized Flux, if the groundwater velocity in the cover soil is typical for a site condition of shallow land disposal.

CONCLUSION

1. For ordinary soil cover, the following RFs (Retardation Factor) of imitative (non-radioactive) nuclides were calculated from the experimental data in the case of the 40 cm/day of groundwater velocity, and those convinced us that the ordinary soil cover also has a certain barrier function.

Rf of Sr=67, Rf of Co=121, Rf of Cs=107

2. The B/S (Bentonite/soil mixture layer), which consisted of 10wt% bentonite and 90wt% sandy soil, reduced the velocity of the groundwater to 1/1300 compared with that in the ordinary soil cover layer.

3. The behavior of radionuclides for long term was evaluated by using experimental data, and the following analytical results showed clearly the effectiveness of the B/S mixture layer.

a. In case of the installation of the bentonite/soil mixture, migration of short-lived radio-nuclides such as Co-60 could be restrained remarkably and they would disappear by radiation decrement before release from the burial facility.

b. The maximum release rate of long-lived radionuclides such as Nb-94 or I-129 would be reduced by a factor of 1/1000 to 1/10000 by the installation of a bentonite/soil mixture, and the time at the maximum

release rate would be also delayed, compared with the case of no bentonite/soil mixture, to 40,000 years for Nb-94 and 400 years for I-129.

16-6
DISPOSAL OF 14C ... ASSESSMENT RESULTS
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ABSTRACT
Carbon-14 is an important radionuclide to retain in waste repositories and away from the environment. It has all the characteristics of a problematic radionuclide: long half-life; high mobility in most environments; high biological incorporation; and for some wastes, relatively high inventory. In Canadian CANDU power reactor systems, much of the 14C that is produced is captured and retained in various wasteforms. In previous studies, disposal in a deep plutonic-rock vault was assessed to be suitable for 14C in fuel wastes. Different disposal concepts were assessed for low- and intermediate-level wastes (L&ILW), and these included cementitious matrices and a carbonate rock formation. These concepts for L&ILW proved useful in a generic sense, although in stochastic simulations there were situations where average releases were higher than our target. Data were sparse for several features of the assessment, and these data needs are described.
INTRODUCTION
Of the many radionuclides in radioactive waste, 14C is unique because of a combination of important features. It has a long half-life. In many geochemical settings it behaves as a gas or as an anion, both with high mobility. It becomes chemically indistinguishable from the 12C in all life forms, and so is easily absorbed into biota including humans. At the same time, there is good potential to isolate 14C in durable wasteforms and repositories to keep it away from the surface, biosphere environment. In the Canadian radioactive waste disposal programs, 14C is dealt with in a number of ways that are estimated to lead to low or negligible doses. This report summarizes the concepts evaluated to date, with emphasis on low- and intermediate-level waste (L&ILW). The results presented here for L&ILW are preliminary, and should be interpreted as a scoping analysis only. Various assessment methodologies will be used before an assessment is considered complete, and this paper deals only with one, key, modelling approach.
CANADIAN DISPOSAL CONCEPTS
Canada has over 20 operating CANDU reactors, and all the waste is in interim storage awaiting final disposal. Several concepts for disposal facilities have been advanced. For nuclear fuel waste, the concept itself is now being formally evaluated by Federal agencies before there is any action taken toward siting. The nuclear fuel waste (NFW) disposal concept is for a disposal of aged but intact fuel bundles in corrosion-resistant metal (titanium or copper) containers. The containers are emplaced in clay-based buffer and backfill in disposal rooms mined into plutonic rock 500 to 1000m underground. The concept has been assessed (1) and is now under review, with hearings pending. If successful, a facility could be built in time for start-up in 2025. Documentation of the NFW disposal concept is complete and widely available (1, 2, 3, 4, 5), and so the

brief description of assessment results described here is only to show the relative effectiveness of various 14C retardation processes.
Disposal plans for L&ILW are at an even earlier stage: we are seeking and comparing concepts. Two concepts for L&ILW have been assessed to date, and several more assessments are underway. The documentation of the assessment results (6, 7) emphasizes the preliminary nature of these assessments and concepts.
The two concepts for disposal of L&ILW discussed here are disposal in glacial till, and disposal in carbonate rock. The conceptual till facility has a near-surface vault, and was deemed suitable only for low-level waste (LLW). This does include some 14C. The vault in till is constructed of reinforced, low-permeability concrete and is backfilled with porous concrete. The conceptual carbonate rock vault is at 150-m depth and was assessed for both LLW and intermediate-level waste (ILW), and contains substantial amounts of 14C. The vault consists of mined-rock rooms backfilled with porous concrete. Once a successful concept has been identified and approved, a disposal facility or facilities could be built in time for start-up in 2015.

WASTE STREAMS AND INVENTORIES

The NFW is expected to contain 5.0 x 107 Bq 14C per kg uranium, with a total inventory in the vault of 8.1 x 1015 Bq 14C (2.2x105Ci). This 14C will be present in the UO2 grains, with less than 0.1% in the grain boundaries and other gaps in the fuel bundle (1). The LLW assessed for the till vault is expected to contain 1.1x 107 Bq 14C (3.0x10-4Ci), all in readily leachable forms. The L&ILW assessed for the carbonate rock vault is expected to contain 1.1 x 1016 Bq 14C (3.0x105Ci) immobilized as calcite, 2.8x1015Bq (7.6x104Ci) 14C in metal components where the 14C will be released on corrosion of the metal, and 4.2x1012Bq (110Ci) 14C in high density concrete.

ASSESSMENT MODELS

The assessment models, used as part of the overall concept assessments, are quite detailed and are structured in three parts: vault, geosphere and biosphere. The combined models start with information about waste inventory and system properties and output dose estimates for humans and other biota with time after closure of the vault. These models are controlled through the executive SYstems Variability Analysis Code (SYVAC). Most parameter inputs are specified by probability density functions (PDFs) that described the most likely values, the variation about these values, and the upper and lower truncation limits. Parameters may also be statistically correlated, using a simple correlation coefficient. The SYVAC executive can operate the models in a probabilistic mode, where values are chosen randomly based on the PDFs, or in a deterministic mode. The deterministic mode used most often is one where the median values for each parameter is used. These are called median-values runs in this report.
When the models are operated in a probabilistic, Monte Carlo mode, they are run many times with many independent suites of parameter values. Typically >1000 runs are made to obtain meaningful probabilistic results. Although the dose estimates conform to a strongly skewed frequency distribution, the summation of interest to the regulator (8) is the arithmetic mean of these dose estimates.

RETARDATION AND DOSE-ATTENUATION PROCESSES IMPORTANT FOR 14C

In the NFW concept, a corrosion resistant container, a diffusion barrier adjacent to the waste containers and a long path length through the

plutonic geosphere to the biosphere are the major retardation features important to isolation of 14C. Geochemical retardation by processes such as sorption or isotopic exchange were assumed to be nil (9).

In the biosphere for both the NFW and L&ILW assessments, several processes built into the models contribute to dose attenuation (10, 11). A key one is an upper dose cutoff based on assumptions about specific activity in the geosphere. The 14C emerging into the biosphere from the geosphere will undergo isotopic exchange with 12C along the path. This is expected to be a massive dilution, especially in the biosphere, but it is difficult to model because the rate of mixing cannot be specified in a realistic yet conservative way. However, one defensible series of assumptions is that isotopic dilution is essentially irreversible, so that dilution that occurs at the discharge from the geosphere will be manifest throughout the biosphere. In fact, it will be augmented in the biosphere. The isotopic dilution in the geosphere is computed using the estimated 14C concentration in groundwater entering a bedrock well and the observed inorganic 12C concentration in groundwater at depths typical of bedrock wells. The values of the 12C concentration in groundwater are assigned using a PDF. The resulting isotopic ratio is then used to calculate a dose to humans and to other biota, assuming the same isotopic ratio throughout their bodies. The dose conversion factor is 2.5x10-7 Sv a-1 per Bq kg-1 soft tissue. The resulting dose estimate is assumed to be the highest dose that could happen, and implies that people and other biota derived all of their tissue C from the geosphere. Obviously, much really comes from the atmosphere.

The geosphere upper dose cutoff is used as a check of dose estimates derived through classical transfer parameter models (12). The lower dose of the two calculated in each run is used, and contributes to the arithmetic mean dose from all of the probabilistic runs. Sheppard et al. (11) describe the 14C biosphere model and the underlying parameter values in detail. Specific activity modelling is used in other parts of the biosphere model as well (5, 6), most notably for the soil-to-plant transfer of 14C.

For LLW disposal in till, wasteform degradation does not slow the release of 14C: it was assumed to be fully available for leaching once the vault barrier is breached. The 14C was moderately sorbed in the vault backfill concrete. The concrete will react with any aqueous 14C to form calcite reaction products, and this will further retard 14C migration (13). Isotopic exchange was not included in the present assessment. Once 14C is released from the vault, the configuration of the till disposal vault was such that groundwater flow was vertically downward through the vault and underlying till and into a carbonate rock formation underneath. Flow was then lateral towards a lake. Geochemical retardation was modelled in the carbonate geosphere. However, even here the retardation was set to nil in 10% of the runs to account for chemical species of 14C that may not be retarded by the carbonate rock.

For ILW, the wasteforms were an additional barrier, and the most important for controlling 14C migration. The 14C in metal matrices was released by corrosion of the metals. Most of the inventory of 14C is in cementitious wasteforms because here it will reside as calcite, which has been shown to be quite effective in attenuation of 14C (13). Laboratory experiments have demonstrated that the release of 14C from the calcite matrix is solubility-limited congruent release with no net dissolution of the calcite matrix. The geochemical aging of the concrete was also

modelled using the CHEQMATE code (14), and the results indicated that although pH decreased with aging, the original calcite remained stable and only a limited amount of 14C was released by isotopic exchange with inorganic C in groundwater. Once 14C was released from the vault, a long path length through a carbonate geosphere was modelled, with geochemical retardation in the carbonate rock in 90% of the runs.

RESULTS OF ASSESSMENT

The full assessment of the NFW concept embodied several approaches, only one of which was the running of the SYVAC simulations. Only SYVAC results are presented here.

The dose from the median-values runs and the mean dose from the probabilistic runs for the NFW concept were dominated by 129I, with 14C and 36Cl as the next most important radionuclides. The peak dose from 14C was 1.4x10-8Sva-1 at 4.0x104a. All doses were well below the regulatory criterion. The dominant pathway for exposure of humans to 14C (and 129I and 36Cl) was through irrigation of contaminated well water onto agricultural/gardensoils (2). The ratio of peak annual dose to humans per unit of 14C in the inventory was 1.7x10-24Sva-1 per Bq in inventory.

The LLW vault in till was assigned a relatively small inventory of 14C, and 14C was the eleventh highest dose-contributing radionuclide, with a peak dose in the median-values run of 2.5x10-10 Sv a-1 at 2.2 x 104 a. No radionuclides in this assessment gave unacceptable doses (Fig. 1), and the top dose contributor was 99Tc. For 14C, the ratio of peak annual dose to humans per unit of 14C in the inventory was 2.3x10-17Sva-1 per Bq in inventory.

The L&ILW vault in the carbonate rock is considered the major disposal facility for 14C, and the inventory here is the highest of the three conceptual facilities. In the median-values simulations, the top dose contributor was 36Cl, giving 1.3x10-5Sv a-1 at 4.4x 102 a (Fig. 2). The 14C in the calcite wasteform was the fourteenth highest contributor, giving 1.3x10-10Sv a-1 at 7.2x 104 a. This dose from 14C is both low and delayed because of the attenuation in the vault. For 14C in the median-values runs, the ratio of peak annual dose to humans per unit of 14C in the inventory was 1.2x10-26Sva-1 per Bq in inventory. This ratio is two orders of magnitude lower than that for the NFW, where the physical barriers were substantially greater. This indicates the importance of the geochemical barriers in the cementitious wasteforms and geological settings of the L&ILW concept.

The effect of geochemical retardation is well illustrated by the probabilistic runs. The median dose with time from these runs was dominated by 14C (Fig. 3), even though 14C was not an important contributor in the median-values runs. The peak mean dose was 7.0x10-5Sva-1 and occurred very early after closure of the vault. The reason 14C was important to mean dose and not to the median-values dose is that in the probabilistic simulations, the full range of possible parameter values was chosen. As an example, this included ~10% of the runs where the geochemical retardation mechanisms for 14C in the geosphere were nullified (Fig. 4). This is considered a very unlikely situation. However, the results of the probabilistic runs show how important it is to quantify these processes. An unacceptable outcome would result if the geochemical retardation mechanisms were less effective than expected.

Fig. 1

Fig. 2

Fig. 3

Fig. 4

FUTURE DEVELOPMENTS FOR L&ILW

The retardation of 14C in geological materials must be well known. Sheppard et al. (15) reported solid/liquid partition coefficients, Kd or Rd, of 8 to 85 L kg-1 in carbonate minerals and soils, but zero values in clays. Clearly, host rock and, especially, cementitious backfill are indeed important to the isolation of 14C from the environment (16). More work is needed to ensure that the chemical species of 14C that may leave a vault are inorganic or are retarded to the same extent as inorganic species.

Degradation of the cementitious vault and the subsequent release of 14C are critical processes. Much of the information here must be based on geochemical models, but there is potential for important analogue studies such as the Maqarin site in Jordan (17).

Several other concepts for disposal of L&ILW will be assessed, one including a large concrete silo in a deep geologic vault, and another including co-disposal or co-siting with NFW disposal.

CONCLUSIONS

Carbon-14 is actively accumulated in the operation of nuclear power reactors in Canada. The inventories are substantial, and may be divided among three waste disposal facilities. The NFW facility will contain about a third of the 14C, and although it is not in a high-pH environment, it achieves effective isolation through container durability, substantial diffusion barriers and depth of vault. The LLW vault will have only a modest inventory of 14C, and the conceptual facility analyzed was effective in isolating the 14C. The L&ILW facility will have about two-thirds of the total 14C disposed waste, and the concept analyzed was successful when the cementitious vault and carbonate geosphere provided the expected attenuation of 14C. However, when these barriers were not as effective as expected, the 14C did result in estimated doses above our target. More work is required to gain assurance that the retardation of 14C in the cementitious vault and carbonate geosphere for the L&ILW repository are effective.

ACKNOWLEDGEMENTS

This project was supported by the CANDU Owners Group, funded jointly by Atomic Energy of Canada Limited and Ontario Hydro. The author has summarized the contributions of a large assessment team, and especially acknowledges the expertise and guidance provided by R. Dayal, J.D. Garroni, W.C. Hajas, C.I. Kitson, D.M. LeNeveu, J. McMurry, T.W. Melnyk, J.R.Ramsay, S.B.Russell, J. Torok, K.H. Witzke and D.M. Wuschke. This manuscript was improved by the thoughtful comments of R. Dayal, D.M.LeNeveu, J. Torok, M. Stephens, D.M. Wuschke and R. Zach.

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16-7

CHARACTERIZATION OF MEDIUM AND LOW ACTIVE WASTES

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ABSTRACT
The characterization of a waste package, consists in determining its specific properties.
For FRANCE, this involves ANDRA specifications, the Basic Safety Rules (les Rgles Fondamentales de Sret - RFS) of the Nuclear Safety Direction (Ministry of Industry).

INTRODUCTION

The characterization of a waste package consists of determining its specific properties.
In the characterization of nuclear waste, we restrict ourselves to the acquisition of characteristics which have to be known in order to obtain agreement on near surface disposal and to evaluate safety in temporary and permanent storage conditions.

OBJECTIVES OF WASTE CHARACTERIZATION AND CHARACTERIZATION TYPES

Five main objectives are looked for when performing a characterization test. The tests are intended to:

- guarantee the mechanical properties of the waste form and the matrix,
- test the durability of the physical and mechanical properties of the waste for the disposal duration,
- evaluate the behavior of radionuclides in the waste and the long-term waste-matrix interaction,
- calculate the loss rate by isotope leaching,
- compare the results with specifications, and rules specified by the Authorities and Safety notices

For FRANCE, this involves ANDRA specifications, the basic Safety Rules (les Rgles Fondamentales de Sret-RFS) of the Nuclear Safety Direction (Ministry of Industry).
Several types of characterization are necessary to cover all knowledge useful for storage and disposal of radioactive waste.
There are three main types of characterization :

- characterization of packages and materials,
- behavioral studies,
- long-term behavioral studies.

Characterization of Packages and Materials

This type of characterization can be defined as the acquisition of the characteristics of a coating, a package or a material in its initial state. This type of characterization is carried out in three steps :

- characterization of the waste itself,
- pre-characterization of the material and the matrix,
- characterization of the finished product (the package intended for storage or disposal).

Characterization of the Waste

This is defined as acquisition of data necessary for :

- producing the waste package,

classification for its activity and its destination :
- (class FA-MA-HA),
- (class or),
- (destination : surface disposal, temporary storage before repository in a geological disposal),

analysis of basic safety and its entry into the approval file.

Pre-Characterization (process and coating)

This is the essential stage for the evaluation of the quality of an embedding process resulting from all the pre-production steps (formulation, reference tests, terminal tests, sensitivity test, etc). Tests for this stage are carried out using reference materials.

Two types of objective are involved :

a comparison of the various performances of the conditioning process (embedding rate, formulation, etc),

a demonstration that the embedding, the packaging, and the finished products satisfy standards and specifications in force.

This step normally results in the production of a packaging specification and the production of nominal specifications for the waste form.

Characterization of the Finished Product

This stage involves three types of investigation :

examination of the intrinsic characteristics of packagings and embeddings,

checking the durability of these characteristics (degradability, confinement, etc),

define the influence of variations of the various process parameters (sensitivity tests) in order to guarantee the quality of the waste formcoated material.

Characteristics and acceptance criteria for the finished product are examined with in chapter 2 of this document.

Long-term Behavior

This type of characterization is defined as the acquisition of data for mechanisms dealing with slowly evolving phenomena. Two approaches can be used :

the behavior of natural counterparts,

the thermodynamic definition of slow phenomena.

The final result that is expected to be obtained from this last characterization stage consists of changing from "high bound" type modelling to a modelling system based on extrapolationon the Long Term of results required by way of short term experiments.

PACKAGE APPROVAL: SPECIFICATIONS, PROCEDURES AND ACCEPTABILITY CRITERIA

Inspection of the current situation in countries developing nuclear energy shows that most countries base approval of waste packages on three main data items:

specifications produced by the authorities,

procedures produced by approved laboratories and organizations,

acceptability criteria or quality criteria which should lead to the respect of specifications.

Although the organization of waste-related responsibilities and therefore the specifications issued by the various authorities vary between countries, all have established some characterization procedures and acceptability criteria for waste packages.

Procedures - Technical Sheets

In France, approval of the waste package is announced after inspection of experimental results and tests in accordance with the minimum characterization program defined by ANDRA.
This minimum program affects the three selected activity levels and the two classes of waste packages (homogeneous-heterogeneous). Tests and checks are classified into four categories :

- physical characteristics,
- mechanical characteristics,
- confinement properties,
- stability-degradability (maintenance of confinement properties).

In order obtain an harmonization of tests required for the approval file, the Confinement Checking and Evaluation Office (Bureau d'Evaluation et de Contrle des Confinements - BECC) has published a set of technical forms codifying a general method of executing these tests.
A block diagram has been produced by the Waste and Effluents Management for the approval procedure . Figure 1 summarizes the global routing of an approval circuit and specifies the role of each participant for wastes which can be delivered to ANDRA on a surface site.
Fig. 1
ANDRA approval of waste packages requires the production of the following documents:

- evaluation of activity contained in the waste,
- description of the production process for the package containing the waste,
- quality assurance program for the implementation of the package production process as described in the process description,
- package characterization test report (technical characterization files).

The characterization test report assumes that the producer has previously submitted a draft characterization test program to ANDRA. When this draft has been accepted by ANDRA, it then becomes the characterization procedure, containing a description of the checks and tests to be carried out.
Therefore, the creation of a test requires the following definitions (Table I):

- the number of samples,
- the size and characteristics,
- the test procedure,
- the results to be provided,
- the test report.

All reports for all tests described in the protocol must then be checked and used to produce a tests summary report, or a characterization technical file which is forwarded, as already stated, to ANDRA by the producer.
The procedure for measuring the initial activity of raw waste and packages which meets ANDRA's requirement for the monitoring of the activity stored on the site in order to respect the radiological capacity of the site for each radionuclides is very important.
Consequently, waste activity measurement (raw or waste form) is one of the most important criteria. Characterization therefore requires the implementation of an adapted measurement scheme for each waste or coating type.

Each type of waste has its own measurement planning for taking measurements and for monitoring the procedure so as to satisfy safety authority requirements.
The main procedure for coatings is shown in Table III. This table clearly shows the two recommended types of measurements :

non-destructive measurements :

gamma scanning spectrometry for the evaluation of the uniformity of the distribution wastes and radionuclides within encapsulation. This technics is applied on samples cores taken from a package and on drums of homogeneous waste:

tomography for non-destructive examination methods are particularly useful in the characterization of radioactive wastes. Radiographic examinations combined with tomodensitometric examinations using gamma ray photon source make it possible to carry out exhaustive testing of the physical homogeneity of a low or medium activity waste package in preliminary characterization, quality control of the produced packages and expertise

destructive measurements with the various related determinations.

Finally, to complete the execution of these measurements, Table IV defines the essential elements which must appear in the test report intended for the producer in order to prepare the summary report for all the tests, as required by the final management authority (ANDRA for FRANCE).

Table I

Table II

Table III

Table IV

Package Acceptability Criteria

For FRANCE, characterization deals only with conditioning and package quality criteria. These are listed in the minimum characterization program defined by ANDRA as recalled in section Procedures - Technical sheets

In view of the available characterization results for various matrices enclosing various types of waste, the BECC established a summary of packaging qualities, as a function of the main parameters.
Three types of parameters were used for this summary (Table II) :

quality of the conditioning-quality of package (RFS + ANDRA specifications),

implementation and feasibility,

control.

An evaluation was made for each type of parameter, matrix and waste type (homogenous - heterogeneous) based on the acquired characterization results.

CONCLUSION

For several years now, research on raw or conditioned waste characterization has been carried out in France, in particular those aspects concerning :

legislation,

regulations,

standardization of methods and techniques,

laboratory and test hall equipment.

The BECC provides the main communication link between the waste producer, ANDRA and the CEA which can perform all services related to tests and corresponding measurements, either as part of its own research and

development program. CEA resources, methods, laboratory and test halls, now form a characterization system adapted to each waste or packaging type to which a quality assurance program is applied.
BECC manufacturing experience enables it to give correct advice to the waste producer and any nuclear material safety unit, both on the selection of packaging and test procedures, and on equipment and the selection of characterization materials or the organization to be set up. In particular, we can provide the following services :

qualitative or quantitative analysis of radionuclides present in already packaged waste (including badly packaged evaluation of the main physico-mechanical and confinement characteristics),

technical assistance for the characterization of packages including :

- developing specific technical test forms according to the customer,
- establishing characterization coordination units and training corresponding personnel,
- transferring basic elements for the implementation of a quality assurance program for package characterization,
- establishing basic characterization method and packaging expertise planning,

technical assistance with the design and construction of characterization laboratories.

To conclude, we would like to emphasize that the objective of the various units in charge of characterization is to supply technical assistance in view of the harmonious development of waste approval files and the development of test techniques to improve knowledge of the intrinsic characteristics of packages waste. Above all, the framework in which these characterization activities are carried out must respect the special features of this characterization, which is at the very interface of the various radioactive waste management partners : safety authorities, waste management (ANDRA), waste producers, and the managers of research and development programs.

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Co-chair: Bob Lawrence, WVNS

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DEPARTMENT OF ENERGY MANAGEMENT OF THE WEST VALLEY DEMONSTRATION PROJECT

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ABSTRACT

The US Department of Energy (DOE), in cooperation with the New York State Energy Research and Development Authority (NYSERDA), has managed the West Valley Demonstration Project (WVDP) since 1982, as prescribed by Congressional Act. One provision of the Act stated that nuclear high-level waste remaining from reprocessing operations at the West Valley, NY site was to be solidified into a safe, durable, and storable product. Vitrification was selected as the method to best accomplish this charter. As the site readies for production of radioactive glass canisters beginning in 1996, the DOE has been successful in its management of the clean-up mission. The DOE has taken this Project from its developmental stage to vitrification readiness by adopting several practices to ensure the scheduled completion of radioactive operations. From the onset of the Project, the DOE has been a strong proponent of oversight by the US

Nuclear Regulatory Commission (NRC) and other external agencies. This practice has led to the up-front identification and resolution of concerns relative to worker and facility safety, long before they become an issue during actual operations. Delegation of authority has been transferred from DOE Headquarters to increase responsibilities at the lowest level. From the very beginning, an open door policy was instituted between the site and the general public, media, and stakeholders; making our programs and progress accessible for everyone. Regularly scheduled meetings are used to inform the public of pending site issues. Cooperative relationships have been established between DOE approval units for such items as Safety Analysis Reports, waste specification and qualification issues, and Operational Readiness Reviews (ORR). Technology that has been developed and refined during the Project's five-year nonradioactive test program has been transferred in the form of lessons learned to other government-owned operations in similar pursuits.

The WVDP's continued success through all stages of operation adds benefit to the DOE. Technologies developed at the WVDP have been applied to other DOE sites, and other DOE sites' technologies have been adaptable to the WVDP, thus playing an important role in reducing costs in government-managed operations. Research and development costs for proven engineering designs are minimized when the final designs are shared by the DOE's Savannah River, Hanford, Oak Ridge, and Idaho Falls sites. The exemplary safety record exhibited during construction, testing, and start-up activities at the Project has also been a direct contribution to schedule accomplishments.

This paper describes highlights of the vitrification process, the cooperative efforts involved in making this Project a success, the lessons learned during the evolution from research and development through construction and start-up, successful technology transfer, and the benefits to be gained by the rest of the DOE complex.

PROJECT HISTORY

The WVDP located in West Valley, NY is the site of a former nuclear fuel reprocessing plant. From 1966 to 1972, reprocessing operations at the West Valley facility generated high-level radioactive acid PUREX waste, THOREX waste, and contaminated condensate. These wastes were stored in separate underground tanks at the West Valley site. Neutralization of the PUREX waste resulted in precipitation of some materials that formed a sludge at the bottom of one of the high-level waste tanks. When processing of nuclear fuels was terminated and the facility was closed, these radioactive liquids/sludge remained in the tanks for the New York State government to manage.

In October 1980, the US Congress passed the West Valley Demonstration Project Act which directed the DOE to safely dispose of the high-level radioactive wastes left at the site by transforming these wastes into forms suitable for storage and transfer to a federal repository. The Act further directed the DOE to clean and close the facilities used and dispose of the low-level and transuranic wastes generated during Project operations.

In 1981, the DOE selected West Valley Nuclear Services, Inc. (WVNS) as their prime contractor. DOE and WVNS assumed operational control of the site in 1982. Project funding comes from both the DOE (90 percent) and NYSERDA (10 percent). A 1981 photo of the site appears in Fig. 1.

Fig. 1

HIGH-LEVEL AND LOW-LEVEL WASTE PROCESSING

By 1988, high-level radioactive waste processing had begun. In the high-level waste tank, the waste had separated into two layers: a clear liquid above a heavy layer of precipitated particles or sludge. Solidification of these separate layers required processing in two stages. The liquid portion or supernatant was sent through a synthetic clay material that removed 99.9 percent of the radioactive materials. The resulting low-level waste, containing salts and sulfates, was concentrated and blended with cement and stored in a shielded, above-ground cell. Supernatant processing was completed in 1990 with a total production of about 10,000 cement drums.

Mixing pumps were then installed in the high-level waste tank to mobilize the sludge for washing. Washing included the addition of process water to the tank to dissolve the salts and sulfates. The wash water was processed through the same liquid treatment system as the low-level waste and produced another 9,800 cement drums. Three washes of the sludge were performed, readying the mixture for vitrification operations.

From 1984 to 1989, the WVDP operated a full-scale glass production test facility. This nonradioactive Functional and Checkout Testing of Systems (FACTS Testing) determined the correct component configurations and confirmed the glass recipe to be used during radioactive operations. Conversion of this test facility for radioactive processing was completed in late 1994. Functional checkout, precommissioning testing, and commissioning testing of vitrification systems were completed in mid-1995. Several integrated operations runs were also executed to confirm proper performance of utility systems, operator actions, alarm responses, and remote equipment operation.

The vitrification process itself entails first combining all of the radioactive constituents into one tank prior to transferring the mixture to the Vitrification Facility. The washed sludge or slurry, the radioactive synthetic clay material from supernatant processing, and the THOREX waste was combined into the high-level waste tank. In the Vitrification Facility, the waste stream will be combined with glass-forming chemicals, concentrated, and fed to 52-ton ceramic melter. The melter will heat the mixture of glass formers and radioactive waste to 1150oC, and the resultant molten glass will be poured into stainless steel canisters. Approximately 300 canisters will be produced during this radioactive campaign and stored on site. All solidification operations and transfers will be performed remotely. Refer to Fig. 2 for a drawing of the vitrification facilities.

Radioactive tie-ins between the Waste Tank Farm piping and the high-level waste transfer trenches leading to and from the Vitrification Facility began in November 1995. Actual radioactive glass production is scheduled to begin June 1996.

Fig. 2

SAFETY

The WVDP operates under one singular tenet: there is nothing more important than personal safety. From the onset of the Project, safety in the workplace has been stressed through the continuing education of ourselves and our subcontractors. Safety programs, such as Conduct of Operations, Voluntary Protection Program, Radiation Safety Training, Behavior-based Safety Training, and numerous other site-wide programs, have led to the development of a comprehensive safety culture encompassing all site workers alike. The work force has been given the individual authority to stop any activity that might jeopardize a

person's well-being. To develop a stronger safety awareness among the many construction subcontractors for the vitrification facilities, safety field engineers were hired to walk down areas prior to and during construction to determine potential safety hazards and correct them before injuries could occur.

Additionally, a WVDP Conduct of Operations manual was developed specifically to augment DOE Order 5480.19, "Conduct of Operations Requirements for DOE Facilities." The WVDP manual outlines adherence to best management practices for every site activity including communications, logkeeping, operations turnover, equipment labeling, lock and tag, postings, and housekeeping.

According to the Bureau of Labor Statistics, the Total Recordable Case Rate (TRC) for all types of construction across industry is 12.2 (1). This figure is derived by taking the number of accidents, multiplying that by 200,000 hours, and dividing the result by the number of hours worked. In actuality, the case rate is representative of the number of OSHA-recordable injuries per 200,000 manhours, or the number of accidents per 100 full-time employees.

The TRC for all types of construction across DOE sites, including their contractors, is 6.8. For lump sum construction (the type of construction contracted at the WVDP), the TRC rises to 7.5. For comparison, the WVDP has maintained a TRC significantly lower than both private industry and DOE site averages. For construction of all the facilities necessary for radioactive glass production between January 1984 and December 1995, the TRC was 3.56. This figure is indicative of only 58 OSHA-recordable accidents in approximately 2,358,469 construction manhours worked over a period of almost 11 years. The Lost Work Day Case Rate (LWDC) figures for the site are also well below commercial and DOE averages, with the WVDP case rate for 1994 being 1.48 in comparison to the private industry rate of 5.5 and the DOE sites' rate of 3.6 for the same year. For all construction activities performed during 1995, the LWDC rate fell even lower to 0.59.

OPEN DOOR POLICY

At the time the DOE assumed management of the WVDP, the site was primarily closed to the media and public. Expectedly, media coverage was negative and activities occurring at the site were viewed somewhat with suspicion by stakeholders and surrounding neighbors. To remedy this condition, the DOE quickly adopted an Open Door Policy, inviting the public and media on site for briefings and tours to outline site characteristics and proposed clean-up activities. Additionally, a yearly Open House and regularly scheduled meetings were initiated so the public, local media, and government officials could tour the site and partake in numerous educational exhibits designed to explain waste processing operations. Attendance at the Project's Open House has averaged about 1,500 people per year (see Fig. 3). During 1995, the WVDP's Community Relations Department responded to over 200 inquiries, conducted approximately 60 site tours, and made about 50 off-site presentations to educational groups, technical societies, and institutions. The DOE and NYSERDA have also made general Project information available to the public and placed environmental documents in reading rooms located throughout the Western New York area.

This new, open communication policy between the DOE and its stakeholders has been greeted by both positive media coverage and increased public awareness of the site's radiological activities, thereby creating a

better understanding of future waste disposal and site closure activities. In February 1996, the site's draft Environmental Impact Statement (EIS), required by the National Environmental Policy Act, was issued for a six-month public review and comment period. This EIS document, developed jointly by DOE and NYSERDA, details five years of site characterization information and evaluates a full range of alternatives for Project completion and site management. A Record of Decision by DOE and NYSERDA with regard to which alternative is the best site closure and management method will be completed after public review and comment.

Fig. 3

PROJECT MANAGEMENT

DOE management of the WVDP has shifted from a long-term Cost-Plus-Award-Fee contract to a yearly performance-based contract. This type of contract, negotiated for the WVDP in 1995, outlines mutually developed milestones that represent a fairly accurate estimation of achievable goals by specific completion dates. These types of milestones carry added or reduced monetary incentives dependent upon early, on-time, or late completion of the milestone.

Performance indicators are similarly used to gauge the monthly progress of support activities for such items as, As Low As Reasonably Achievable (ALARA), LWDC, Occupational Safety Violations, Clothing and Skin Contaminations, Procurements, Waste Minimization, etc. Indicators also carry incentives for maintaining budgets under established limits. Performance-based contracts provide a clearer plan to follow for each year's activities. Milestones and performance indicators have focused the Project in its entirety on critical path events for radioactive vitrification operations, out-year environmental aspects, and long-term waste stabilization efforts. The prior subjectivity of a full cost-plus-award-fee contract has been eliminated and replaced with increased contractor accountability and measurable contract performance. For the WVDP, all 1995 milestones were completed on time with the majority of performance indicators accomplished within set boundaries.

EXTERNAL AGENCY OVERSIGHT

In the Project's early stages, the DOE recognized the benefits of inviting external agencies to provide oversight of upcoming radiological operations. Audit groups, technical review groups, and nuclear energy associations were given access to engineering calculations, drawings, and reports in order to independently determine the adequacy of seismic construction, radiation shielding, the Waste Form Compliance Plan, and the Waste Form Qualification Report. Regular site visits by the NRC established positive communication between the parties resulting in useful and prudent design modifications to the Vitrification Facility. NRC representatives were also brought in to evaluate the Safety Analysis Report for Vitrification Operations and High-Level Waste Interim Storage. Additional support was enlisted from a joint Technical Review Group, comprised of 35 individuals from DOE, private and commercial industry, and the NRC to verify safety analysis report accuracy and completeness.

TECHNOLOGY TRANSFER

Technology transferred and lessons learned, shared among DOE sites, have served to improve our operations and reduce costs by minimizing redundant research, development, and testing activities. The WVDP has benefitted from Pacific Northwest Labs' (PNL) glass technology, ceramic melter, and canister turntable designs. PNL also developed the titanium-treated

zeolite used in the WVDP's ion exchange process for separating radioactive elements from the chemical content of the high-level radioactive waste tank. From the DOE's Hanford site, the WVDP has adapted their hot cell, window and cell penetration, and remote handling tool designs. Hanford also fabricated all of the WVDP's Waste Tank Farm and In-cell remotely removable connecting pipes or jumpers. From the DOE's Savannah River site, closed-circuit television camera designs and nondestructive waste tank examination methods were transferred and utilized. WVDP modified and is using a remotely operated canister transfer cart design developed at the DOE's Oak Ridge facility. Industrial lighting for radioactive cells was revised and installed in the WVDP's Vitrification Facility from designs developed by the DOE's Idaho site.

Technology developed for use at the WVDP has similarly been made available to other DOE locations. Ceramic melter operating information and instructional programs have been given to the Savannah River Defense Waste Processing Facility (DWPF). The DWPF also received from the WVDP infrared level detection technology used for determining the radioactive glass level inside a stainless steel canister and Distributed Control System programming parameters and instructions. The DOE's Hanford site was the recipient of waste tank mobilization pump designs, pump operating information, and proven liquid waste pretreatment technologies.

Recently, through the WVDP's Invention Disclosure Program, a device for bolt fastening/capturing and a device for removing entrained condensate from horizontal pipe have been published by the US Patent Office and made available by the DOE for licensing to the private sector. Several other DOE-funded inventions, developed for use at the WVDP, will be available in the near future for transfer to general industry through publication by the US Patent Office and licensing agreements with the DOE.

DOE APPROVAL UNITS

In keeping with current approaches to transfer approval authority from DOE-Headquarters to units more familiar with and accessible to the WVDP, several document approvals were transferred from higher DOE offices. Two prominent examples of this strategy include: approval authority for the Safety Analysis Report for Vitrification Operations and High-Level Waste Interim Storage was transferred from DOE-Headquarters to DOE-Ohio; and approval of the Vitrification Operational Readiness Review Plan of Action was transferred from the Secretary of Energy to the Assistant Secretary for Environmental Management.

In both instances, transferring approval authority from higher levels within the DOE resulted in a more accurate determination of each document's validity and served to expedite review cycle times.

BENEFITS OF THE WVDP TO THE DOE COMPLEX

Clean-up operations at the West Valley site and strategies developed for use in this site's restoration have contributed to the overall environmental recovery mission of the DOE. What has been learned on a site-wide level; from waste remediation efforts, testing activities, safety programs, and waste minimization practices; is applicable to other DOE facilities and may serve to reduce costs and improve implementation schedules for these programs elsewhere. By the DOE's establishment of a total safety culture, the WVDP has been able to reduce lost work days for all personnel to levels lower than general industry and DOE averages. Subsequently, costs for interrupted work due to accidental injuries have been decreased.

CONCLUSIONS

Today, the WVDP stands on the verge of being one of the first jointly operated (DOE/NYSERDA), high-level radioactive waste vitrification facilities to operate in the United States. Solidification of radioactive waste into a stable, durable form is scheduled to commence in June 1996 and conclude in October 1998, with a total production of about 300 canisters. Canistered waste will remain on site in interim storage until a federal repository is designated. A photo of the WVDP as it appears today is shown in Fig. 4.

Our past success has been predicated on commitment to management involvement, worker safety, external agency oversight, open communication with stakeholders and the public, and technology transfer. Our continued success will build upon these fundamentals and proportionately expand to include new, innovative management techniques designed to correspond with the DOE's strategic alignment to focus energies on contract reforms and more prudent use of taxpayer dollars.

Fig. 4

REFERENCES

1. The figure cited is from the Survey of Occupational Injuries and Illnesses - 1993, US Department of Labor, Bureau of Labor Statistics, issued February 1995.

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DESCRIPTION OF THE WEST VALLEY VITRIFICATION PROJECT

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ABSTRACT

The Vitrification Facility at the West Valley Demonstration Project (WVDP), West Valley, NY will be used to solidify residual high-level waste that was previously generated by an on-site nuclear fuel reprocessing plant. The Vitrification Facility has been constructed and radioactive operations are scheduled to commence in 1996. The facility, including both the pretreatment and vitrification processes, are described in this paper.

INTRODUCTION

By a Congressional Act in 1980, the West Valley Demonstration Project was created to, among other things, solidify the high-level radioactive waste that resulted from the former nuclear fuel reprocessing plant operations. Using existing and newly constructed facilities, vitrification and related systems were designed and installed to accomplish this Project mission.

Initial liquid pretreatment operations, completed in 1995, resulted in the decontamination of 6,400 cubic ft3 of liquid high-level waste and the production of 19,877 drums of cemented low-level waste. The Vitrification Facility is expected to produce about 300 stainless steel canisters 0.61 m in diameter by 3.05 m tall filled with the high-level waste encapsulated in borosilicate glass. The vitrification system uses a joule-heated melter that produces glass at a rate up to 30 kg/hr. The canisters will be stored on site until a federal repository is designated.

The paper describes the processes used for high-level waste pretreatment and vitrification, and the methods used to mix and combine the waste in one high-level waste tank prior to vitrification. The entire

Vitrification Facility is described and includes the vitrification remote process cell, vitrification process equipment, Cold Chemical Facility, Canister Storage Cell, computerized Distributed Control System (DCS) that monitors processing operations, and melter off-gas treatment system. Lessons learned throughout the Project are discussed.

PRETREATMENT AND VITRIFICATION PROCESS

Wastes of varying composition and radioactivity levels were stored in separate underground tanks at the West Valley, NY site. Tanks 8D-1 and 8D-2 are duplicate carbon steel tanks, approximately 21.3 m in diameter and 8.23 m in height. Tank 8D-2 contained about 2,420 m3 of neutralized PUREX waste, 170 m3 of that amount consisting of sludge. Tank 8D-1 was used as a spare for Tank 8D-2; condensate from Tank 8D-2's ventilation system gravity-drained to Tank 8D-1. Tanks 8D-3 and 8D-4 are also duplicate tanks made of stainless steel, and are approximately 3.7 m in diameter and 4.8 m in height. Tank 8D-4 contained approximately 35 m3 of THOREX nitric acid waste. Tank 8D-3 was originally used as a spare.

The neutralized PUREX waste stored in Tank 8D-2 required separate treatment methods for the liquid (supernatant) top layer and bottom sludge layer. The supernatant layer was decanted through zeolite-filled, ion-exchange columns suspended in Tank 8D-1 to remove cesium. The resulting decontaminated liquid was mixed with cement and solidified into about 10,000 square drums. The drums are stored above ground in a shielded cell on site.

Process water was then added to Tank 8D-2 to wash the sludge of crystallized salts and sulfates. Large pumps were inserted through risers in the tank to mobilize the sludge and mix it with the process water. Heavy particles were left to settle and the wash water was again decanted through the ion-exchange columns and solidified into cement. Two sludge washes were performed. Sodium hydroxide was next added to Tank 8D-2 to neutralize the incoming THOREX waste transferred from Tank 8D-4. Another wash was performed after THOREX transfer and neutralization to remove excess sulfates. These sludge washes resulted in the production of another 9,877 drums of cemented low-level waste.

Spent zeolite dumped in the bottom of Tank 8D-1 was passed through a grinder and returned to Tank 8D-2.

For high-level waste processing the radioactive slurry in Tank 8D-2 will be transferred to the Vitrification Facility for solidification in borosilicate glass. Slurry will exit the Tank Farm area via double-walled piping inside a concrete underground trench. Once inside the Vitrification Facility, the waste stream will be mixed with glass-forming chemicals delivered from the Cold Chemical Building, concentrated, and fed to a 52-ton ceramic melter. The melter will heat the mixture to 1150oC bonding the glass-forming chemicals to the waste. This molten mixture will then be poured into stainless steel canisters, remotely transferred to a lid welding station, a decontamination station, and positioned on a transfer cart for transport to the High-Level Waste Interim Storage Facility. Approximately 300 canisters will be produced during the Vitrification campaign and stored on site until a federal repository is ready to accept the waste. A diagram depicting low-level and high-level waste processing appears in Fig. 1.

Fig. 1

DESCRIPTION OF VITRIFICATION FACILITIES

Vitrification Cell

Radioactive glass production to specific requirements can only be accomplished by the integrated operation of numerous components, systems, and facilities. A brief description of the vitrification facilities and major components follows:

The Vitrification Cell holds the majority of the vitrification process equipment. The cell is a shielded, seismically designed, confinement boundary with interior dimensions of approximately 10.37 m (34 ft) by 16.6 m (53 ft) long by 13.11 m (43 ft) high. Cell walls are four foot-thick reinforced concrete and consist of prefabricated wall modules placed between previously cast-in-place concrete columns. Modules are constructed of a structural framework to support internal piping and penetrations for six shield windows and utility access. A solid stainless steel plate faces the radioactive cell interior. The modules were installed and filled with concrete, which formed the nonradioactive side of the finished module. A total of seven wall modules were installed and welded together to form an integral structure that serves as a radiological and seismic barrier. The cell floor is at elevation 100.00' or grade level, and the cell exterior roof is at elevation 145.00'. There are no intermediate floors in the Vitrification Cell. Surrounding the cell are three floors of operating aisles equipped with instrument racks, to control the delivery of cell services, and supporting utility rooms. Structurally, the cell was designed to withstand a design basis earthquake equivalent to a ground acceleration of 0.1g horizontal load, with a vertical component equal to two-thirds of the horizontal acceleration, and a design basis tornado based on detailed analysis of tornado occurrences in Western New York State.

Cell Components

Installed in the Vitrification Cell are the primary process tanks including a Concentrator Feed Makeup Tank and a Melter Feed Hold Tank. A 52-ton ceramic melter, Submerged Bed Scrubber, four-position canister turntable, canister cooling rack, canister lid welding station, decontamination station, and one of two off-gas systems also reside in the cell. The process tanks heat and concentrate the glass-forming chemicals, delivered from the Cold Chemical Building, and the waste delivered from the Waste Tank Farm prior to transfer into the melter. The melter is a stainless steel, water-jacketed box complete with corrosion-resistant interior and separate chambers for glass melting and glass pouring. Various types of thermal-resistant refractory brick line the melter interior. Joule heating of the slurry is performed by three metallic electrodes with regulating current. The canister turntable rotates empty canisters to a position beneath the melter pour spout and filled canisters to cooling positions. Viewing the glass pour level of the canister is performed by an out-of-cell-located infrared level detection system. The Submerged Bed Scrubber, designed to quench and scrub off-gas particulate from the melter, has the ability to circulate and remove entrapped liquids and solids to avoid buildup in the scrubber bed. The In-cell Off-gas System consisting of preheaters, HEPA filters, and mist eliminators removes the bulk of radioactive particulate from the off-gas stream. The lid welder remotely welds stainless steel lids onto filled canisters prior to decontamination activities. The Decontamination Station is used to submerge each filled and capped canister in a nitric acid-cerium (+4) solution to etch off a thin (3-5 micron) layer of the canister's exterior which may contain submicron particles of fixed contaminants. A recent photo of the Vitrification Cell appears in Fig. 2.

Fig. 2
Cell components are accessible via a crane system comprised of a 4.5-ton process crane, a 25-ton backup crane, an impact wrench, a bridge trolley and rails. Most major components in the cell are remotely removable using only crane-mounted equipment. Utility services to components are hooked up to in-cell components by crane-removable jumpers. High-dexterity jobs such as glass shard sampling are performed using manipulators. Floor rails enable the movement of a remotely operated canister transfer cart in and out of the cell through a large shield door. Viewing of cell operations and canister movement is accomplished by closed-circuit television.

Adjacent Shielded Structures

South of the Vitrification Cell is a shielded transfer tunnel, a sort of air lock with shield doors on either end, used to separate the cell atmosphere from the former plant's Equipment Decontamination Room (EDR). Located above the tunnel is the Crane Maintenance Room (CMR) and Crane Maintenance Room Operating Aisle (CMROA). The CMR will be used to service the cell cranes. South of the EDR is a former chemical process cell that will be used as the High-level Waste Interim Storage (HLWIS) Facility for approximately 300 canisters. The transfer cart will travel on floor rails between these facilities. Figure 3 outlines the canister travel path from the Vitrification Cell to the High-level Waste Interim Storage Facility.

Interior dimensions of the transfer tunnel are 4.88 m (16 ft) wide by 9.68 m (31 ft) long. A hatch in the ceiling of the tunnel allows for removal of inoperative equipment into the CMR. The tunnel has concrete walls and a sloped floor lined with 304L stainless steel to allow for decontamination. HVAC systems will maintain a negative pressure on the tunnel and direct the flow of contaminated air into the Vitrification Cell. A closed-circuit television camera mounted on the tunnel wall provides remote viewing capabilities.

The EDR was also part of the original plant and required considerable decontamination prior to use. This room will be used as an entry port from the Load-in Building for empty canisters, transfer cart maintenance, and recharging the cart's battery pack. The cell's concrete interior is epoxy coated and has dimensions of approximately 8.75 m (29 ft) wide by 13.33 m (43.7 ft) long by 7.62 m (25 ft) high. The walls are approximately 0.91 m (3 ft) thick reinforced concrete and the floor is 0.36 m (14 in) thick concrete. A ceiling hatch opens to the Chemical Crane Room located above for additional crane maintenance.

Storage of stainless steel canisters will be south of the EDR in the former Chemical Process Cell of the main fuel reprocessing plant. The HLWIS Facility is a shielded cell 28 m (92 ft) long by 6.70 m (22 ft) wide by 13 m (43 ft) high. Walls are 1.75 m (5 ft 9 in.) thick and the ceiling is approximately 1.52 m (5 ft) thick. Four shield windows permit viewing of the cell interior from the outside viewing aisle. The transfer cart enters the facility through another shield door. Individual canisters will be unloaded from the cart, by a 16-ton crane with grapple attachment, into storage racks. An additional crane also resides in the cell. A total of 11 storage racks, with a capacity of 36 canisters each, were installed in a two-tiered, interlocking system. Cranes requiring maintenance can be brought into the Chemical Crane Room located above the EDR. The cell has two closed-circuit television cameras for viewing unloading operations.

Load-in Building

The Load-in Building is located west of the EDR and will be used as the primary access for moving empty canisters into and filled canisters out of the Vitrification Cell. The building's concrete base mat and steel structures are designed to withstand a design basis earthquake of 0.1g. The floor is at grade elevation 100', and the roof rises to elevation 153'. A 15-ton crane provides canister and equipment transfer capabilities. Provisions have been made for a 50-ton crane to accommodate future load-out of the filled canisters into transportation casks. Canisters are loaded from the Load-in Building to the EDR horizontally by conveyor. A removable shield plug blocks the entry port into the EDR. When the shield plug is removed, canisters can enter the EDR (see Fig. 3). A tipping fixture uprights the canisters for placement onto a transfer cart. An area of the Load-in Building will also be used as a chemical staging area for nonradioactive chemicals used in melter feed preparation.

Fig. 3

Secondary Filter Room

The Secondary Filter Room, south and adjacent to the Vitrification Cell, houses two large filter units that provide filtering of the air exhausted from the Vitrification Facility. The room is a reinforced concrete structure with approximate dimensions of 6.70 m (22 ft) wide by 9.76 m (32 ft) long by 6.70 m (22 ft) high. Walls are reinforced concrete and vary in thickness from 0.61 m (2 ft) to 1.22 m (4 ft). An additional 4.27 m (14 ft) by 4.88 m (16 ft) area can be used for the maintenance of equipment.

Diesel Generator Room

The Diesel Generator Room, west and adjacent to the Secondary Filter Room, houses a 600 kW generator, switchgear, distribution cabling, and controls to provide backup power to selected loads in the event of a temporary or extended power outage. The room is designed to withstand the site's Design Basis Tornado and Design Basis Earthquake. Its dimensions are 12.20 m (40 ft) by 8.23 m (27 ft). Floor elevation is at grade level and the roof elevation is 111.50'.

01-14 Building

The four-story 01-14 Building was part of the original plant and reused as part of the vitrification facilities in keeping with the Project's objective to use existing buildings as much as practical. Off-gases produced from operation of the melter contain radioactive and nonradioactive components. The off-gases are treated by the scrubbing and filtration systems for removal in excess of 99.99 percent of the radioactive particulates. Subsequently, the off-gases are treated in the NOx Abatement System, that is designed to remove 91 percent of the oxides of nitrogen prior to exiting the Main Plant stack. The area used for vitrification-related off-gas components is 21.34 m (70 ft) by 9.15 m (30 ft) by 18.29 m (60 ft) high.

The 01-14 Building also houses the Cement Solidification System (CSS) that used to produce low-level cemented waste drums during earlier processing activities.

Vitrification Main Control Room

The Vitrification Main Control Room is located at elevation 114.6' of the Vitrification Facility. The computer area floor is raised to elevation 117.5' to accommodate cable access and routing. Refer to Fig. 4 for a photo of the Vitrification Main Control Room. The room is sealed for fire protection and access is only permitted by card reader. The control room

has two operator stations each with Distributed Control System monitors, Infrared Level Detection monitors for canister level detection, closed-circuit television monitors, and keyboards. HVAC monitoring panels, fire protection monitoring panels, and various other facility and utility indicators are located next to the operator stations. The shift engineer and shift supervisor stations are located on the upper level of the Vitrification Main Control Room. The shift engineer's station is equipped with a duplicate Distributed Control System monitor, Infrared Level Detection monitor, and closed-circuit television screens. The shift supervisor's station has the same equipment with the exception of the Infrared Level Detection monitor. Laser printers, VAX terminals, telephones, radios, and supporting system documentation are also present on both sides of the Vitrification Main Control Room.

Fig. 4

Distributed Control System

The data collection and control system for the vitrification process is a computerized system called the Distributed Control System (DCS). The system has four redundant work stations and is based in the Main Control Room of the Vitrification Facility. The DCS allows for monitoring, control, and supervision of vitrification processes. Control of process equipment can also be accomplished from remote control panels located in operating aisles adjacent to the Vitrification Cell. Data collection and control functions are performed for the: Cold Chemical Building; In-cell and Ex-cell Off-gas Systems; Ammonia Supply and NOx Analyzers; HVAC; Sludge Mobilization System; Cooling and Utility Water Systems; Steam; Utility and Instrument Air Systems; Canister Turntable; Waste Header and Vessel Vent Systems; Primary Vitrification Process System; Canister Decontamination; and Electrical Backup Systems.

Computer monitors display control functions in graphical form along with key operating parameters such as vessel levels, temperatures, valve positions, pressures, and flows presented in real time. Process inputs are audibly and visually alarmed. Certain functions such as set point changes have restricted access and electronic permissives are used to coordinate activities prior to implementation.

The DCS power panel is fed from an uninterruptible power supply that is a self-contained battery power source with diesel generator backup.

Vitrification Facility HVAC System and Considerations

HVAC zones in the Vitrification Facility are maintained by barriers such as walls, aisleways, and doors. The walls, aisleways, and doors prevent the spread of contamination between rooms. Differential pressure ranges are maintained with reference to atmosphere or adjacent zones throughout the facility. Zone I designated areas are those that may contain radioactive materials during normal operations. The Vitrification Cell, Transfer Tunnel, and Crane Maintenance Room are examples of Zone I areas. Zone II designates operating areas and other potentially contaminated areas surrounding Zone I. The EDR, Chemical Crane Room, and Load-in Building are examples of Zone II areas. Zone III HVAC areas are expected to be free of contamination. Ex-cell aisleways and the Control Room are examples of Zone III areas. Air flow is directed from Zone III designated areas into Zone II areas and from Zone II into Zone I, or from areas of lesser contamination potential into areas of higher contamination potential.

The HVAC System is comprised of a filtered air supply and exhaust, in-cell coolers, exhaust fans, Diesel Generator Room ventilation, Chiller

Equipment Room ventilation, a stairway pressurization system, HVAC Control Room supply and exhaust control dampers, electric unit heaters, the Chilled Water System, and Cold Chemical Building heating and ventilating. System criteria, as discussed above, is to maintain the release of radioactivity and airborne particulate below design limits.

CONCLUSION

Construction of the vitrification facilities was completed in 1995. Precommissioning and commissioning testing of components, and the functional checkout of systems were completed in early 1995. Performance testing of systems to Test Instruction Procedures was completed in December 1995. Integrated operations runs with a selected combination of systems began in September 1995 and are scheduled to be completed in May 1996. Preliminary preparations for radioactive tie-ins between the Waste Tank Farm and the vitrification facilities were initiated in November 1995; actual tie-ins are scheduled to be completed in April 1996. Vitrification of high-level waste is expected to begin in June 1996 with the pouring of the first production radioactive glass canister. Approximately 300 canisters will be produced during the expected two and one-half year long vitrification campaign.

Successful high-level radioactive waste solidification into borosilicate glass will be a significant milestone for the US Department of Energy, West Valley Demonstration Project, New York State Energy Research and Development Authority, and West Valley Nuclear Services, Co., Inc.

17-3

HIGH-LEVEL RADIOACTIVE WASTE PRETREATMENT AT THE WEST VALLEY DEMONSTRATION PROJECT

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ABSTRACT

The West Valley Demonstration Project (WVDP) has completed the pretreatment of the high-level radioactive waste (HLW) prior to the start of waste vitrification. The HLW originated from two million liters of plutonium/uranium extraction (PUREX) and thorium extraction (THOREX) wastes remaining from Nuclear Fuel Services (NFS) commercial and defense nuclear fuel reprocessing operations at the Western New York Nuclear Service Center (WNYNSC) from 1966 to 1972. The pretreatment process and subsequent vitrification of the HLW were authorized in 1980 by passage of the WVDP Act. This gave the U. S. Department of Energy (DOE) the responsibility to conduct a HLW management project at the WNYNSC to demonstrate viable decontamination and decommissioning techniques and stabilize the HLW stored on the site from prior reprocessing operations. HLW pretreatment operations, utilizing existing facilities to the maximum practical extent, were conducted to separate the radioactive portion of the waste from the salts found in the waste to minimize the quantity of vitrified waste that will be produced. The pretreatment process removed cesium as well as other radionuclides from the liquid wastes and captured these radioactive materials onto zeolites. The decontaminated salt solutions were volume-reduced and then mixed with portland cement and other admixtures. Approximately 19,900 square, 270-liter drums were filled with the cement-wastes produced from the pretreatment process. These drums are being stored in a shielded facility on the site until their final disposition is determined.

Over 6.4 million liters of liquid HLW were processed through the pretreatment system. PUREX supernatant was processed first, followed by two PUREX sludge wash solutions. A third wash of PUREX/THOREX sludge was then processed after the neutralized THOREX waste was mixed with the PUREX waste. Approximately 6.6 million curies of radioactive cesium-137 in the HLW liquid were removed and retained on 65,300 kg of zeolites. With pretreatment complete, the zeolite material has been mobilized, size-reduced, and blended with the PUREX and THOREX sludges in a single feed tank that will supply HLW slurry to the Vitrification Facility.

INTRODUCTION

Approximately two million liters of liquid HLW were produced by NFS in the reprocessing of commercial and defense nuclear fuel from 1966 to 1972 at the WNYNSC in West Valley, NY. The wastes consisted of alkaline PUREX sludge and supernatant, and a small quantity of acidic THOREX liquid. The PUREX waste was generated from reprocessing 640 metric tons of uranium oxide fuel to recover both uranium and plutonium. Sodium hydroxide was added to neutralize this acidic waste for safe storage in the 2.8 million liter, underground carbon steel storage tank, designated Tank 8D-2. Neutralization and concentration of the tank's contents caused insoluble hydroxides and other salts, notably sodium sulfate (Na_2SO_4), to precipitate out of the liquid supernatant and form sludge layers in the bottom of Tank 8D-2. Both hard and soft sludge layers were found to exist through sampling programs (1). Prior to the initiation of HLW pretreatment, approximately 100,000 kg of sludge (Sp.Gr.=3.3) and two million liters of supernatant (Sp.Gr.=1.32) were contained in Tank 8D-2. The major chemical and radionuclide constituents are listed in Tables I and II, respectively.

Table I

Table II

THOREX waste liquid was produced by reprocessing one core of mixed uranium-thorium fuel from the Indian Point No. 1 Nuclear Plant. This acidic HLW was stored in a 50,000 liter, underground, stainless steel storage tank, identified as Tank 8D-4. Approximately 31,000 liters (Sp.Gr.=1.84) of this waste, having the chemical composition shown in Table III and the major radionuclide content presented in Table IV, were present at the beginning of HLW pretreatment.

Table III

Table IV

PRETREATMENT PROCESS

Early in the Project, it was decided to pretreat the liquid HLW. This was necessary to remove those salts which have a detrimental effect on the final vitrified HLW form, notably sodium sulfate. Without pretreatment, the quantity of vitrified waste would have increased by at least tenfold to maintain the sulfate at acceptable levels in the glass.

Various pretreatment methods were studied and evaluated with the assistance of Battelle PNL, EBASCO, and others. Evaluation criteria included: process decontamination performance; equipment and process complexity; impacts on the vitrification system and the Liquid Waste Treatment System (LWTS), which volume-reduces the decontaminated HLW liquid; and safety and environmental considerations. The use of an inorganic/zeolite ion exchange process received the top ranking and was selected as the reference pretreatment technology (2).

INTEGRATED RADWASTE TREATMENT SYSTEM

The Integrated Radwaste Treatment System (IRTS) was specified to carry out the HLW pretreatment utilizing four separate subsystems: Supernatant Treatment System (STS), Liquid Waste Treatment System (LWTS), Cement Solidification System (CSS), and Drum Cell (DC). These subsystems were constructed using existing facilities and equipment whenever possible to minimize Project costs. The STS uses zeolite to separate the highly radioactive constituents in the HLW from the salt solution. The decontaminated salt solution is volume-reduced in the LWTS by evaporation to a concentration of 20 to 40 wt% Total Dissolved Solids (TDS), suitable for solidification with portland cement. The CSS mixes the decontaminated liquid concentrate with portland cement and other admixtures and remotely fills 270-liter square drums with cement-waste discharged from a high-shear mixer. These drums are remotely loaded onto a shielded, overland truck transporter that delivers up to eight drums at a time to the remotely operated DC. At the DC the drums are automatically off-loaded and placed into a compact storage array inside a shield structure enclosed by a weather shelter.

Supernatant Treatment System Facility Description

A simplified process flow diagram of the STS is shown in Fig. 1. Supernatant, and then later sludge wash, are transferred from Tank 8D-2, the 2.8 million liter underground HLW carbon steel storage tank, to the STS by a submersible vertical turbine pump (50-G-001) suspended in the tank. The pump has a floating suction to minimize the potential for sludge pickup and it is supported from the vault roof. Optional filtration (50-F-001) is provided to prevent process contamination by removing sludge particles suspended in the supernatant or sludge wash. The filter is capable of being pulsed and blown back with compressed air to clean the accumulated particles from the porous tube filtering surface.

The 6,400 liter Supernatant Feed Tank (50-D-001) serves as a surge tank for intermediate collecting and feeding of supernatant to the ion exchange columns. Supernatant, which is ready for ion exchange processing, is transferred from Supernatant Feed Tank 50-D-001 through Supernatant Cooler 50-E-001 by a canned centrifugal pump (50-G-002) at a rate of 8 to 30 lpm. The supernatant is cooled to less than 12C to improve the cesium removal efficiency and then is pumped downflow through three or four ion exchange columns (50-C-001, 50-C-002, 50-C-003, and 50-C-004) in series. Each ion exchange column contains 1,700 liters (1,630 kg) of zeolite.

Lab analysis is performed to determine the cesium loading in each ion exchange column. When the first column is fully loaded (saturated with cesium), supernatant processing is stopped. All columns are then flushed with demineralized water and the system is placed in recirculation through the second, third, and fourth ion exchange columns for the remainder of the shutdown to cool the column contents. The fully loaded zeolite in the first or lead column is then sluiced to the bottom of Tank 8D-1 with process water and is filled with fresh zeolite before the column goes back on-line in the last or polishing position. The loaded zeolite is temporarily stored under corrosion-inhibited water in Tank 8D-1 at approximately 50C until it is combined with the HLW sludge in Tank 8D-2 and delivered to the vitrification system.

Following ion exchange, the decontaminated supernatant is filtered through a sand filter to remove any suspended zeolite fines. The filtered and decontaminated supernatant is then transferred to existing

underground spare THOREX Waste Tank 8D-3. This tank has a working volume of 34,100 liters for STS processing and serves as both an intermediate storage tank and as a sampling tank. Sample analyses are performed to verify the cesium concentration and decontamination factor of each batch of decontaminated supernatant that is produced. Decontaminated supernatant is transferred to the LWTS from Tank 8D-3 in batches for volume reduction by evaporation.

Fig. 1

Liquid Waste Treatment System

The function of the LWTS is to reduce the volume of the liquid waste by boiling off excess water in a steam-fired evaporator/reboiler, which is run in thermosyphon mode. The major components of the system are located within the Main Process Building of the formal nuclear fuel reprocessing facility (3).

Decontaminated waste from the STS effluent tank is batch-transferred by a submerged, vertical turbine pump installed in Tank 8D-3 and an in-line booster pump to the 57,000 liter evaporator feed tank.

After sampling, the waste is slowly pumped at flow rates up to 40 liters/minute to the high-efficiency evaporator where the liquid is concentrated as desired for future solidification in cement, typically from 20 to 40 wt% TDS.

The distillate from the condenser is processed through a zeolite ion exchange column to remove cesium and strontium. This liquid effluent is then sent to the site's Liquid Waste Treatment Facility (LLWTF) for additional treatment prior to discharge to the environment. Gaseous effluents from the evaporation process are collected by the Vessel Off-gas (VOG) system, HEPA filtered, and then discharged through the Main Plant stack.

The concentrates produced in the evaporator are pumped through a water-cooled heat exchanger and fed to Tanks 5D-15A1 or 5D-15A2, having capacities of 19,000 and 38,000 liters, respectively. The concentrates are then batch-pumped to the 1,900 liter Waste Dispensing Vessel in the CSS. Instrumentation placed at strategic points throughout the system allow the process parameters to be monitored from the LWTS Control Room in real time. Feed and product tanks are sampled to ensure that the liquid waste stream is within process parameters. Typical activities of the concentrated liquid waste range from 0.01 to 0.5 mCi/ml cesium-137, with an upper design limit of 50 mCi/ml.

Cement Solidification System

The fully automated CSS receives concentrated liquid radioactive mixed waste and produces stabilized cement-waste which meets 10CFR61 criteria for low-level waste. The radioactive portions of the system are contained in the shielded confines of the 01-14 Building, which housed the acid recovery system during prior fuel reprocessing operations. A storage and transfer system provides silo storage for 70 cu. meters of the cement/calcium nitrate blend. This material is pneumatically conveyed into the 01-14 Building where a weight feeder batches it to the cement mixer (4).

Liquid waste is stored in the Waste Dispensing Vessel and recirculated through the piping system with a positive displacement pump to maintain a homogeneous mixture. Process control systems monitor, limit, and meter the liquid waste, additives, and the portland cement/calcium nitrate blend into a high-shear mixer. A data acquisition system records the weight of each additive, bar-coded drum number, and other process

information for record-keeping. After mixing, the cement waste is discharged into a 270-liter, polyethylene-lined square drum. Two mixer batches are required to fill each drum. The drum fill neck is then remotely capped and the drums move on roller conveyors to a remote smearing station. After cleanliness is verified, the drums are automatically sequenced onto the shielded, truck transporter which hauls up to eight drums at a time to the DC.

The CSS facility was designed for fully automatic processing of drums having radiation fields up to 1 R/hr at contact. Due to required maintenance activities, ALARA (As Low As Reasonably Achievable) typically limits operations to 100 mR/hr per drum, which is easily maintained by the excellent decontamination provided by the STS.

Drum Cell

The DC serves as a storage facility for the cement-waste produced by the IRTS, pending approval of the Environmental Impact Statement (EIS) and the Record of Decision (ROD) for final site closure. The facility was designed and constructed so that it can be converted to an above-grade tumulus disposal facility.

The facility consists of 50 cm thick concrete shield walls around a small drum unloading area and a 15 m wide by 105 m long by 4.6 m high drum storage area. A larger, sheet metal, weather structure encloses the shielded areas and also houses the Control Room and a crane maintenance area. The facility was originally designed to store 17,700 270-liter drums of cement-waste, stored 9 layers high (5). Upgrades to the remote drum lifter and other hardware/software modifications have expanded the drum capacity to 11 layers high, with a capacity of 21,200 drums.

Operations within the DC are fully automated. Drums are sequenced off the transporter onto a roller conveyor system that employs a bar code scanner to record the drum number into the data base, upends the drum on its side, and then tips the drum on its side edge. The remotely operated crane picks up each drum from the top and bottom edges and places it into the drum storage array, keeping the lower activity drums on top to provide shielding. The global X,Y,Z coordinates of each drum are logged into the data base, along with its serial number, for documentation and potential retrievability.

IRTS OPERATION

Supernatant Processing

The IRTS was operated from May, 1988 until November, 1990 processing 2.34 million liters of PUREX supernatant having cesium-137 concentrations ranging from 1,100 to 2,800 Ci/ml. Approximately 5.20 million curies of cesium-137 at a decontamination effectiveness of over 99.99% were removed from this liquid and adsorbed on 42,500kg of UOP IONSIVR IE-96 zeolite, which was stored under liquid in Tank 8D-1. The decontaminated supernatant, concentrated to a nominal 40% TDS, was solidified into 10,393 270-liter square drums. The cesium-137 activity in these drums totalled 302 curies. Through various test programs, this cement-waste has been shown to meet all U.S. Nuclear Regulatory Commission (NRC) requirements for a stabilized low-level radioactive waste form (6). Additional test programs also demonstrated that heavy metals contained in the liquid waste; primarily barium, cadmium, chromium, mercury, and selenium; are immobilized so that the solid waste is classified as Resource Conservation and Recovery Act (RCRA) nonhazardous. These drums are currently stored on site in the DC.

Sludge Washing

After completing laboratory simulation tests, preparations, and operational readiness reviews; both internal and DOE/NRC; the PUREX sludge in Tank 8D-2 was washed from October, 1991 to January, 1992. Washing consisted of adding a sodium hydroxide solution and additional water to increase the alkalinity of the liquid waste from an indicated pH of 10 to a pH of 12.5. This was done in conjunction with sequential operation and lowering of the five 150-hp mobilization pumps in the HLW tank to thoroughly mix the contents. This washing process dissolved the hard layer of sludge present in the tank, solubilized sulfate and other undissolved salts present in the sludge, and mixed the interstitial liquid trapped in the sludge with the wash solution. The alkalinity increase precipitated plutonium and uranium compounds formed so that these concentrations in the sludge wash were reduced by factors of 10 and 20, respectively. This was desirable since much more of these long-lived radionuclides would be vitrified with the sludge and not carried through to the cement-waste product.

Sludge Wash No. 1 Processing

Sludge wash no. 1 processing through the IRTS began in April, 1992 and finished in May, 1994. Over 1.55 million liters of the sludge wash no. 1 solution containing 910,000 curies of cesium-137 with concentrations between 520 and 850 Ci/ml were decontaminated using UOP IONSIVR IE-96 and TIE-96 zeolites. The TIE-96 zeolite was developed jointly by Battelle PNL, UOP, and West Valley Nuclear Services (WVNS), with its first commercial use at the WVDP during sludge wash no. 1 processing through the STS. In addition to cesium, the TIE-96 zeolite removes plutonium and strontium with typical decontamination factors of 10 to 100. By using this new zeolite, the amounts of these long-lived radionuclides that are solidified in cement-waste are minimized and the quantities vitrified are maximized. Six-thousand five-hundred kilograms of the TIE-96 zeolite and 4,900 kg of IE-96 zeolite were used during sludge wash no. 1 processing. The cesium activity passing through into the cement waste was 201 curies. Two new stabilized cement-waste forms were developed to solidify the new waste stream to newly revised NRC stability requirements. Portland Type V cement was used in one of the waste forms, instead of the Type I cement, to resist the impact of the increased sulfate content in the sludge wash and to increase the waste concentration from a nominal 20 wt% TDS to 30 wt% TDS. Approximately 7,280 270-liter square drums of cement-waste were produced from the decontaminated sludge wash no. 1 concentrate: 4,160 with Type I and 3,120 with Type V portland cement. Both waste forms were developed to meet applicable U.S. Environmental Protection Agency (EPA) and New York State Department of Environmental Conservation (NYSDEC) regulations for immobilization of heavy metals and the updated stability requirements contained in the 1991 NRC Branch Technical Position on Waste Form. Nearly all of the sludge wash cement-waste produced meets 10CFR61 criteria for a Class A low-level waste due to three factors: 1) precipitation of the plutonium in the alkaline sludge wash 2) plutonium and strontium removal provided by the TIE-96 zeolite and 3) the slightly lower waste concentration employed to adequately produce a stabilized cement waste with the much higher sulfate levels and limit precipitation of solids in the LWTS evaporator.

Sludge Wash No. 2 and Processing

A second sludge wash of the PUREX sludge was preformed during May and June, 1994 to further reduce the amount of sulfates in the HLW prior to vitrification. As with the first sludge wash, sodium hydroxide and water

were added to Tank 8D-2 while the mobilization pumps mixed the contents of the tank. Following the second wash, the wash solution was again processed through the IRTS from June to August, 1994. Approximately 1.35 million liters of wash no. 2 solution, containing 126,000 curies of cesium-137 at a concentration of 100 mCi/ml, were decontaminated in the STS with both types of UOP zeolites used earlier. Only 1,600 kg of zeolite were used during this processing. The cesium activity passing into the 750 drums of cement waste produced from the decontaminated sludge wash concentrate was limited to 21 curies. All drums meet 10CFR61 criteria for a Class A LLW.

THOREX Transfer and Neutralization

Following the completion of sludge washing, final preparations were made to complete the installation of the HLW Transfer System which links all three HLW tanks and the Vitrification Facility together with double-contained piping run in underground concrete trenches and pits. Vertical, turbine-type transfer pumps were installed in Tanks 8D-1, 8D-2, and 8D-4. Readiness assessments for the THOREX transfer and neutralization, THOREX-wash processing, and zeolite transfer were conducted by both WVNS and DOE, with oversight by the NRC. Tank 8D-2 was prepared for the acidic THOREX addition during November and December, 1994 by increasing its alkalinity with sodium hydroxide to approximately a pH of 13. Corrosion probes and a NOx monitor were employed to monitor and limit tank corrosion during the THOREX addition. Dilution air was added into the HLW tank vapor region to also minimize corrosion during the transfers.

The acidic THOREX was transferred from Tank 8D-4 to Tank 8D-2 and neutralized during January, 1995. The THOREX was brought over in three separate transfers. A first transfer of 18,000 liters, a second transfer of 38,600 liters of THOREX with water added to reduce its corrosion potential in Tank 8D-2, and a third transfer of 21,100 liters of a second THOREX dilution. In total, 98% of the THOREX waste was removed from Tank 8D-4 with the remainder left in the tank heel. The NOx monitor and corrosion probes in Tank 8D-2 indicated no significant increase in corrosion rate during the neutralization process. Following neutralization, sodium nitrite was added to Tank 8D-2 to minimize pitting corrosion that could result from the large amount of nitrates in the THOREX.

THOREX/PUREX Wash Processing

After mixing the contents of Tank 8D-2: washed PUREX sludge, sludge wash, THOREX precipitates, and THOREX solubles; the liquid THOREX/PUREX wash was processed through the IRTS. From January to May, 1995, 1.19 million liters of the THOREX/PUREX wash solution, containing 300,000 curies of cesium-137 at a 250 Ci/ml concentration, were decontaminated using 1,600 kg of UOP IE-96 and 1,600 kg of TIE-96 zeolite. The decontaminated THOREX/PUREX wash was concentrated to 20 to 29 wt% TDS in preparation for solidification in cement.

A new Type V portland cement-waste form for the THOREX/PUREX wash was developed and tested. Test data indicated that the waste form immobilizes the heavy metals; specifically barium, cadmium, chromium, mercury, selenium, and silver; so that the waste is classified as RCRA nonhazardous. Additional qualification tests were performed to demonstrate that the resulting cement-waste meets NRC requirements for LLW stability. Actual post-production waste analyses indicated that nearly all of the drums of solidified THOREX/PUREX wash meet 10CFR61 criteria for Class A LLW. This is attributed again to the effectiveness

of the alkaline wash of Tank 8D-2 to precipitate plutonium and the adsorption capability of the UOP TIE-96 zeolite in removing strontium and plutonium from the wash solution, in addition to cesium removal. Approximately 1,450 270-liter drums containing 46 curies of cesium-137 were produced from the decontaminated THOREX/PUREX wash concentrates, bringing the total number of drums stored on site in the DC to approximately 19,900.

Zeolite Mobilization and Transfer

The last pretreatment operation has been ongoing at the WVDP from July, 1995 to January, 1996. The 65,300 kg of zeolite stored in spare HLW Tank 8D-1 have been mobilized with five 150-hp mixing pumps installed in the tank. Once fluidized, the zeolite slurry was pumped from Tank 8D-1 through an in-line grinder that size-reduced the 20 to 50 mesh (840 to 300 microns) zeolite size to approximately 50 microns or less before the slurry was added into Tank 8D-2 and mixed with the PUREX and THOREX sludge. Once a zeolite transfer was completed, Tank 8D-2 liquid was then decanted back to Tank 8D-1 to aid in mobilizing and transferring the next batch of zeolite slurry to Tank 8D-2. Fifteen zeolite transfers have been accomplished with an estimated removal of 85-91% of the original zeolite stored in Tank 8D-1. To accomplish this, over 14 million liters of zeolite slurry have been transferred from Tank 8D-1 to Tank 8D-2. The amount of zeolite transferred has been estimated by the use of radiation probes placed along the transfer piping within the trench and video inspections of the heel remaining in Tank 8D-1. The amount of zeolite removed from Tank 8D-1 greatly exceeds the required quantity to begin HLW vitrification. Although zeolite mobilization and transfer are complete, additional flushes of Tank 8D-1 are scheduled to be performed in parallel with HLW vitrification to further remove zeolite from the tank heel.

SUMMARY AND CONCLUSIONS

The WVDP has completed its high-level radioactive waste pretreatment program begun in 1988. The pretreatment process consisted of four distinct phases: PUREX supernatant processing, sludge washing, THOREX waste neutralization and washing, and zeolite mobilization and transfer. Pretreatment of the HLW was specified to separate the majority of the radioactive species from the nonradioactive salts, specifically those of sulfate, in order to reduce the total volume of HLW glass.

The pretreatment process for the first three phases was similar. The HLW supernatant from the PUREX waste, sludge washing, and neutralized THOREX waste was processed through the STS where the cesium-137, and later the plutonium and strontium-90, radionuclides were adsorbed onto the zeolite. This system effectively removed these radionuclides, thereby providing decontamination factors of at least 1,000 for cesium-137 and approximately 10 to 100 for plutonium and strontium-90. The spent zeolite was stored in a spare HLW tank until later in the pretreatment process. The resulting decontaminated salt solutions were then volume-reduced by a steam-fed evaporator in the LWTS. The concentrated, radioactive mixed waste was blended with portland cements and special additives in the CSS that employed high-shear cement mixers. Specially constructed 270-liter square drums were filled with the cement-waste that met the NRC stabilization criteria for Class B and C LLW, as well as the EPA and NYSDEC Toxicity Characteristic Leaching Procedure (TCLP) requirements for heavy metals. Approximately 19,900 drums of four different stabilized cement-waste recipes; each meeting 10CFR61 criteria for Class A, B, or C

LLW; have been produced and are currently stored in the remotely operated DC awaiting final disposition. Table V presents IRTS processing totals.

Table V

The last pretreatment phase consisted of mobilizing approximately 85-91% of the 63,300 kg of zeolite stored under water in the spare HLW tank, size-reducing the particles from the original 570 m size (20 to 50 mesh) to less than 50 m, and transferring the slurry into the original HLW tank containing the washed sludges. The size-reduced zeolites have been blended with the sludges to form a homogenous HLW feed for the Vitrification Facility. The projected composition of the resulting HLW liquid phase is shown in Table VI.

Table VI

Many challenges presented themselves during the pretreatment of the HLW which led to the development of major technical innovations, including:

Development of cement-wastes that met increasingly more stringent stabilization requirements.

First use of square drums to minimize the size of the Drum Cell.

Design, fabrication, and use of specialized remote tooling to repair process equipment within HLW tanks where radiation fields range over 1,000 R/hr.

Development, commercialization, and use of the UOP IONSIVR TIE-96 zeolite that removes plutonium and strontium, as well as cesium. This allowed production of a LLW cement-waste having much lower concentrations of these radionuclides.

The use of fully remote drum handling and bar code tracking systems in the CSS and Drum Cell.

Safely neutralizing the acidic THOREX waste in the primary carbon steel HLW tank using in-tank corrosion probes and a NOx analyzer to monitor the process and ensure minimal corrosion.

Increasing the capacity of the existing Drum Cell from 17,700 to 21,200 drums by modifying hardware and software systems.

Removal of the zeolite from the spare HLW tank after up to seven years of storage in an alkaline environment.

Through innovative engineering and disciplined pretreatment operations, these and other challenges have been met and solutions achieved. Over 6.6 million curies of cesium-137 were removed from the HLW liquid and adsorbed onto the zeolite. The original two million liters of PUREX waste and 31,000 liters of THOREX waste have been processed to produce a slurry of approximately 1.1 million liters that contains washed sludge and size-reduced zeolite. The decontaminated salt solutions produced 19,900 cement drums, collectively containing 570 curies, for an average cesium removal efficiency of over 99.99%.

The WVDP Pretreatment Program has been completed and the accomplishments over the last seven years illustrate that HLW processing can be safely and successfully managed. In addition, the pretreatment processing is a success in that the WVDP cement-wastes have met commercial NRC stabilization requirements, and the heavy metals in the wastes have been immobilized below EPA and NYSDEC TCLP limits, producing RCRA nonhazardous wastes.

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UNIQUE DESIGN FEATURES OF THE WEST VALLEY DEMONSTRATION PROJECT VITRIFICATION PROJECT

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ABSTRACT

The West Valley Demonstration Project (WVDP) has designed, built, tested, and operated a process for vitrifying high-level radioactive waste (HLW) in borosilicate glass. Components, systems, and facilities were designed to take advantage of commercial nuclear experience, control cost and schedule, maximize utilization of existing facilities and systems, and integrate functional requirements with the Project mission to vitrify the HLW as safely, quickly, and cost effectively as possible.

Unique design features which were developed for overcoming constraints that could have impacted the success of the Project are discussed. Among these constraints were limited space, the utilization of existing facilities, short operating life, cost minimization, and conversion from a pilot plant test facility to a remotely operated radioactive production facility.

BACKGROUND

At the WVDP, final preparations are underway for solidification of high-level radioactive waste (HLW) into borosilicate glass in the Vitrification Facility. The fundamental functional requirements for the WVDP Vitrification Project are comparable to those for vitrification projects at Savannah River's Defense Waste Processing Facility (DWPF), Japan's Tokai Vitrification Facility, and England's Sellafield site. However, differences in radioactive waste composition, operating schedules, budget constraints, facility size, and reuse of existing facilities, when combined with the ingenuity and creativity of design engineers, have resulted in the creation of many unique design features at the WVDP Vitrification Project. This paper describes a few of these unique design features.

VITRIFICATION FACILITY

It is often said that "Necessity is the Mother of Invention." In this regard, the Vitrification Facility provides fertile grounds for the growing and harvesting of fresh new ideas. Constraints placed on the design by a process that must be remotely operated, coupled with a number of diverse tasks that are involved in the production of vitrified HLW,

require the design engineer to develop innovative solutions to meet a wide variety of functional requirements.

The structure of the Vitrification Facility presents a case in point. Constructed in 1984 as a test stand for hands-on testing of vitrification equipment components, it was designed to be converted into a shielded, remotely operated facility for solidifying highly radioactive waste in borosilicate glass. The conversion process was started in 1989, after five years of nonradioactive testing. As part of this conversion, seven wall modules, which include all piping and electrical penetrations, were fabricated by an off-site vendor. The wall modules varied in size, with the typical module measuring 4.7 m wide by 7.1 m high by 1.2 m thick. Wall modules were installed into the Vitrification Facility using the same 25-ton overhead bridge crane that is being used for remotely operating the vitrification process. Using the inside stainless steel liner of the wall module as a concrete form, plasticized concrete mix was then pumped into the wall modules for shielding.

The use of these module units permitted testing of the melter in conjunction with the conversion of the Component Test Stand (CTS) into the Vitrification Facility shielded structure. In addition, accurate location of penetrations could be assured to support the fabrication of remote connection pipes or jumpers. This approach reduced the overall construction schedule by over six months.

Beyond the Vitrification Facility itself, there are a number of unique design features associated with the vitrification process equipment. A sampling of these unique design features follows:

Canister Lid Welder

The purpose of the canister lid welder is to achieve leak-tight canister closure after the canister is filled with HLW glass. A process called pulsed Gas Tungsten Arc Welding is used to weld the 16-inch diameter lid onto the canister without the use of filler metal. The welding machine, which is typically used for fully automated welding in the automotive industry, was adapted for remote operations in the WVDP Vitrification Facility (see Fig. 1). The effects of varying the primary process parameters of current, voltage, and travel time were characterized during early developmental testing by welding lids onto canisters or canister mockups and then hydrostatically burst-testing the canisters. Five welded canisters that were dropped from a height of 7 meters at Battelle Pacific Northwest Laboratory successfully passed subsequent helium leak testing.

Different remote tools have been developed for aiding the canister lid weld process. One of these tools, the flange conditioning tool, consists of an end mill and a wire brush, both of which are mounted on a carriage assembly. The flange conditioning tool can be used to clean the weld surface, perform local excavation for repair of nonconforming welds, or mill a groove for welding on a secondary canister lid, if a satisfactory weld of a primary lid is not possible.

Fig. 1

Evacuated Canister

The purpose of the evacuated canister is to allow the remote removal of radioactive molten glass from the melter during a planned melter shutdown. An L-shaped stainless steel dip tube (four inches in diameter), sealed at one end with an aluminum plug, is inserted into the molten glass in the melter. The aluminum plug melts after 3 to 8 minutes and the molten glass is then sucked into the canister by vacuum. The evacuated canister fills to approximately 85 percent full in 15 minutes with molten

glass that is initially at a temperature of 1150C. Surface temperatures on the outside of the evacuated canister approach 900C during the filling operation. A photo of the evacuated canister appears in Fig. 2.

Fig. 2

Slurry-fed Ceramic Melter

High-level waste slurry is fed to the ceramic melter where it is first dried and calcined and then gradually mixed with molten borosilicate glass. Slurry-fed ceramic melters at DWPF; Tokai, Japan; Karlsrhue, Germany; and PAMELA in Belgium all have refractory-lined melting cavities that are joule-heated. However, these melters differ in the method of pouring glass (side versus bottom), number and type of pipe penetrations, method of power/temperature control, and method of viewing the melter cavity.

The first WVDP melter used during cold testing had Inconel pipe nozzles. Due to corrosion experienced during testing, ceramic (alumina) liners were added to the current melter design. During functional checkout, these liners cracked due to a combination of physical stresses, thermal shock, and thermal expansion. They were subsequently replaced with a second, redundant Inconel liner that employs a passive air purge to sweep corrosive gasses away from the Inconel melter pipe nozzles.

Viewing the inside of the melter is important whenever feeding slurry. The melter viewing system, which is functionally similar to a periscope, is used to determine the extent the molten glass pool is covered by a floating island of drying slurry and calcined solids called a "cold cap." Adjustments of slurry feed rate are made based on the size of the cold cap. A similar device is used to view the glass during pours from the melter into each canister. Based on watching the orientation of the glass pour stream on their TV monitors, operators can make adjustments in the rate of glass pouring and, if necessary, terminate an off-normal pour. Figure 3 shows the melter viewing system.

Another method used to control glass height in the canister is an Infrared Level Detection System. This system indirectly monitors infrared emissions from a thermally hot canister during and immediately after glass pouring. This same system has also been adapted for use at the DWPF based on successful test results at the WVDP.

Fig. 3

Off-gas Treatment

Originally designed as a passive scrubber for venting reactor containments, the Submerged Bed Scrubber has been adapted for use in the WVDP Vitrification Facility. Hot gasses from the melter are introduced below a bed packed with spherical ceramic beads. The hot gases are immediately quenched (cooled) by the scrubber solution. As the remaining noncombustible gasses rise, they also cause the scrubber solution to rise by an air-lift type effect. The emerging scrubber solution overflows into a collection tank and recirculates by gravity back to the packed bed. Accumulated solids are periodically removed by creating a swirling action and then jetting the solids back to the front end of the vitrification process for recycling to the melter.

Various oxides of nitrogen (NOx) are produced as the solids in the melter feed decompose in the melter. Although radioactive contamination in the off-gas is effectively removed by multiple stages of scrubbing and filtration, including the Submerged Bed Scrubber, the nonradioactive NOx gases still remain. As a final step in off-gas treatment, the NOx is reacted with ammonia at a temperature of 400C in the presence of a

catalyst to form harmless nitrogen and water vapor. Two parallel, redundant NOx reactors were installed in an existing shielded facility to reduce Project costs. The removal efficiency of the NOx reactors, based on actual operating conditions, exceeds 90 percent and fully meets regulatory requirements.

Remote Handling

Operating and maintaining the vitrification process equipment using only cranehooks, impact wrenches, and a limited number of manipulators presents many opportunities for custom-designing equipment to perform highly specialized tasks.

The Vitrification Facility has an "extra pair of hands," the Telerobotic Manipulator or In-cell Robot, that will be used in a support role to assist the cranes, impact wrenches, and manipulators in remotely repairing and decontaminating equipment. The In-cell Robot consists of a stainless steel torso housing on a hydraulic power unit and two 6-1/2 foot-long titanium arms. A photo of the robot appears in Fig. 4. The robot is deployed in the Vitrification Cell using the overhead bridge crane. The robot's gripper jaw hands can be exchanged with a drill motor or other end effectors. An umbilical cable attaches the robot to the control panel located outside the shielded cell. Movements by the operator are duplicated by the robot's corresponding arms/hands. Movements can also be preprogrammed into the computer, saved, and then repeated.

Fig. 4

Canister Decontamination

The exterior of the stainless steel canisters require decontamination prior to shipment to a federal repository. The canister decontamination system at the WVDP uses a dilute solution of nitric acid and cerium (+4) nitrate to chemically etch the exterior surface of the filled and sealed canisters. Laboratory and full-scale testing of this unique system has confirmed the effectiveness of the chemical reactions in removing a predictable amount of the canister's surface, along with any embedded surface contamination. The amount of material removed from the canister surface is controlled by altering the amount of cerium (+4) in the initial volume of decontamination solution.

While the cerium (+4) decontamination solution, followed by an acid rinse with 0.5 molar nitric acid, was effective in removing 3 to 5 microns of the surface material and completing the cleaning process, consistent cleaning of the canister dished-bottom was not initially accomplished. By modifying the surface finish on the bottom of the canisters, acceptable cleaning was obtained based on both appearance and a slight increase in the amount of material loss.

Canister Transfer

Selecting the site for the location of the Vitrification Facility was relatively straight forward. It lies in close proximity to the HLW Tank Farm and aligns with the existing plant's Chemical Process Cell (CPC) where canisters of vitrified HLW will be stored. A shielded tunnel connects the Vitrification Facility and the existing plant. But, there remained the question, how are canisters of vitrified HLW transported from the Vitrification Facility into the CPC? Both the Vitrification Facility and the CPC are remotely operated canyons. That is, they are equipped with remotely operated overhead cranes. One logical alternative considered was to extend the crane rails from the Vitrification Facility into the CPC. However, separation distance (22.86 m), space constraints,

and the prohibitive cost of extending the Vitrification Cell shield walls ruled out this option. A shielded cask transport truck, similar to that designed for use at the DWPF, was also considered but ruled out based on cost factors. Ultimately, a radio-controlled, transfer cart was designed, fabricated, and tested for the transfer of canisters of vitrified HLW from the Vitrification Facility to the CPC, as well as for transferring empty canisters into the Vitrification Facility.

The transfer cart is designed to move a load of four filled canisters with a combined weight of over 10 tons from the Vitrification Facility to the CPC. Additional provisions for handling loads up to 25 tons are provided by trailer carts to handle vessels and load test weights. The transfer cart travels a rail system that connects with the existing rail system in the CPC. The transfer cart's movement is interlocked with the operation of three separate shield doors to assure that the transfer cart cannot be moved when the shield doors are being operated and vice versa. The cart is battery-powered with no trailing umbilicals. The battery pack is automatically recharged, without removing it from the cart, as the transfer cart passes through the charging location. The transfer cart electrical/control system is modularized for ease of remote changeout.

Canister Load-in

The Canister Load-in Facility was designed to introduce clean, empty canisters into the Vitrification Facility. Clean canisters are removed from a custom-designed cargo trailer with pull-out storage racks. They are then placed horizontally on a roller conveyor and manually pushed through an opening in the shield wall. As the canister passes through the shield wall, it engages a tipper device. The canister is slowly uprighted from a horizontal to a vertical position. The rate of motion of the canister in the tipper is mechanically controlled by counterweights and pneumatic cylinders. After the canister is uprighted, a 10-ton overhead crane picks up the empty canister and places it in the transfer cart.

High-level Waste Interim Storage

The High-level Waste Interim Storage (HLWIS) Facility, formerly called the CPC, was part of the original reprocessing plant. It has been redesigned and retrofitted for temporary storage of the estimated 300 canisters of vitrified HLW that will be produced in the Vitrification Facility. On-site interim storage of the vitrified HLW is necessary until the designation of a federal repository. The HLWIS Facility provides significant radiation shielding and cooling to maintain the centerline temperature of the borosilicate glass in the canisters below the repository limit of 400C. A systematic study was performed to evaluate the options for HLW interim storage and identify a cost effective alternative that meets WVDP's needs and requirements. Cost estimates for designing and constructing a new shielded storage facility were compared with the costs of upgrading operations and maintenance of the existing plant buildings. The CPC meets technical requirements for radiation protection, remote handling, transfer of filled canisters into the CPC, and eventual shipment of canisters. Adapting the CPC for interim storage of vitrified HLW canisters was a cost effective and technically feasible alternative for the WVDP.

Twenty-two canister rack storage modules and 1 cooler support module were fabricated at an off-site vendor. Each canister storage rack, consisting of a bottom module and top module, is 6.09 m long, 1.52 m wide, 5.18 m high and weighs approximately 8.5 metric tons. The racks were preassembled off site to assure that they could be remotely installed in

the CPC. After preparations in the CPC were completed, the modules were moved into the CPC on a cart and remotely stacked two-high using a 16-ton overhead crane. All of the racks were remotely joined, after placement, with high-strength pins using a specially designed impact wrench attached to the power manipulator tubes to form a continuous structure.

Each HLW glass canister initially gives off approximately 300 watts of decay heat. The canister storage rack modules are designed to facilitate passive cooling of the canisters through natural convection to maintain the glass centerline temperature below 400C. The sides of the racks are open and vent holes are cut underneath each canister's storage location. As an alternative, two cooler modules are also installed in the CPC. These cooler modules are designed to force cool air throughout the canister rack assembly to cool the canisters and ensure a uniform temperature distribution in the CPC.

A 16-ton overhead crane is used to remotely remove canisters from the transfer cart and install the canisters in the canister storage racks. To allow recovery from a failed crane hoist, a Load Lowering Device was designed and fabricated. This device allows a canister to be safely lowered in the event the hoist becomes inoperative. The Load Lowering Device works entirely on mechanical principles. There are no electric control devices. The rate of descent of the canister is governed by a gear reducer coupled to a fan blade. As the canister descends, the fan rotates. The resistance the fan blade encounters in moving the air governs the fan speed which, in turn, controls the rate of canister descent.

CONCLUSION

Radioactive vitrification operations are scheduled to begin in June 1996, with the entire production campaign scheduled to last about two and one-half years. All functions of the components and facilities directly related to glass production will be performed remotely. The features discussed in this paper are just a few examples of the unique designs developed to integrate these converted facilities for radioactive operations and canistered waste storage.

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VITRIFICATION SYSTEMS TESTING TO SUPPORT RADIOACTIVE GLASS PRODUCTION AT THE WEST VALLEY DEMONSTRATION PROJECT

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ABSTRACT

Vitrification technology to immobilize the High-level Wastes (HLW) in a stable and nondispersible form has been successfully demonstrated, using

nonradioactive waste simulants, at the West Valley Demonstration Project (WVDP). This conclusion is based on a five-year test campaign, using prototypes of the major, full-scale process vessels and components planned for use during radioactive operations, and smaller scale flowsheet experiments. This experience provides high confidence that the radioactive glass production demonstration will be safely completed.

A series of potential waste-form compositions, based on borosilicate glass formulations, were tested against the Department of Energy Office of Environmental Management Waste Form Acceptance Product Specifications (EM-WAPS). (1) Based on the results of these tests, a target composition (termed Ref. 6) was selected for the radioactive vitrification campaign. This composition meets all of the acceptance requirements for eventual disposal in a deep, geologic repository.

This paper discusses the bases for the selection of the target glass composition and the verification that the process flowsheet produces an acceptable product. The data are based on both the full-scale tests, that produced approximately 150,000 kg of glass during 37 individual tests using simulated high-level waste, and small melter experiments. The one-eighth scale vitrification system was operated for eight months to further enhance understanding of the process chemistry, develop a feed make-up strategy and establish redox controls for the melter.

INTRODUCTION

Vitrification technology, ranging from the waste-form composition to the equipment needed to produce canistered waste forms, has been developed and tested for immobilization of the HLW stored at the West Valley Demonstration Project (WVDP). This waste was generated by the commercial reprocessing of 640 tonnes of reactor fuel at the Western New York Nuclear Services Center.(2) At the inception of the WVDP, two wastes were stored in underground tanks. The majority, 2,300 m3, was neutralized PUREX waste. Another 40 m3 of partially processed THOREX waste was also stored. The PUREX waste was washed to minimize the sodium and sulfate salts to be vitrified.(3) The original PUREX supernatant and wash solutions were decontaminated using a zeolite ion-exchange process to retain the cesium. Presently, the PUREX, THOREX and zeolite wastes have been combined, ready for vitrification.

As the previtrification waste treatment activities progressed, the facilities and equipment needed to solidify the waste in glass were installed.(4) In parallel, a borosilicate glass matrix, tailored for deep geologic disposal of the West Valley Site wastes, was developed and tested. The glass waste-form development and nonradioactive vitrification process demonstration activities are described in this paper.

WASTE-FORM DEVELOPMENT

As discussed above, the WVDP high-level wastes have been pretreated and combined, forming a single stream to be vitrified. The estimated composition of the individual wastes and the blended stream are shown in Table I. In addition to waste chemistry, the radionuclide content has also been characterized. The estimated activity of the combined wastes, decayed to a 1996 basis, is listed in Table II. These waste composition and activity data were the basis for developing a glass matrix to immobilize the West Valley HLW.

Table I

Table II

Two sets of glass properties formed the acceptance criteria as the waste-form composition evolved: the waste glass had to be processed using

demonstrated technology and it must have high leach resistance as measured by the Product Consistency Test (PCT) protocol. (5) The processability criterion required that the waste form be produced in a joule-heated melter using Inconela electrodes. This implies an oxidizing glass melted at a maximum average temperature of 1150C.

To assure that the glass was consistent with this technology, viscosity, electrical resistivity and liquidus temperature limits were established. These glass property limitations are shown in Table III.

Table III

To achieve the second glass development measure, PCT performance, the candidate waste glass leach rate must be lower than a glass standard. More specifically, the target glass composition performance has to be statistically demonstrated to be superior to the EA6 glass using the PCT. This performance criterion is stipulated by the EM-WAPS. (1) In order to assure a high quality waste form, this criterion was internally increased to require the glass to exhibit stable, low leach rate performance for a period of at least 28 days rather that the seven day period specified by the PCT procedure.

Using these waste-form composition selection criteria, a series of reference glasses evolved, (7) as shown in Table IV. The Ref. 6 glass formulation was ultimately selected as the final waste form using the criteria discussed previously. The measured processing characteristics of the Ref. 6 composition are listed in Table III. The major chemical additions needed to convert the blended HLW into the Ref. 6 waste-form are listed in Table I.

Table IV

VITRIFICATION PROCESS DESCRIPTION

The vitrification process can be separated into three phases: tailoring the waste slurry by adding glass-forming chemicals to achieve the desired composition, melting the waste slurry and casting the glass into canisters, and closing the canistered waste-forms. The overall process is shown in Fig. 1 and each of these phases are discussed in this section.

Fig. 1

The initial step in the vitrification process is pumping a batch of waste slurry from the storage tank to the Concentrator Feed Makeup Tank (CFMT) in the Vitrification Facility. Approximately 100 batch transfers will be required to process the HLW inventory. At the CFMT, the waste is sampled and concentrated by evaporating water from the slurry. Chemicals are added to the CFMT, based on the waste sample analysis results, to adjust the waste slurry to the Ref. 6 glass composition. Following confirmation that the desired slurry chemistry has been achieved, the slurried melter feed will be transferred to the Melter Feed Hold Tank (MFHT).

The second phase is initiated by pumping the waste slurry from the MFHT to the melter. At the melter, the remaining water is evaporated from the slurry and the waste fuses with the glass-forming chemicals producing the Ref. 6 product. Molten glass is periodically poured from the melter into 3 m tall by 0.6 m diameter stainless steel canisters. After the glass solidifies, the canisters are routed for closure, decontamination and on-site storage.

In the final process stage, the canisters are closed by welding a lid over the top opening. The welding technique is an automated, autogenous, pulsed gas tungsten arc welding process. (8) Following a visual quality verification inspection of the closure weld, the canisters will be decontaminated using a acidic, cerium-based, chemical oxidation

procedure. (9) After decontamination, the canistered waste-forms will be stored on site until they can be shipped off site to a Federal Facility for storage or disposal.

VITRIFICATION PROCESS TESTING

The WVDP vitrification process testing has been performed at scales ranging from full-sized production equipment to examine the interaction between the various unit operations, to bench-top procedures for studying process chemistry and waste-form performance issues. This section summarizes several of the test programs needed to develop the vitrification flowsheet or generate waste acceptance data required by the Department of Energy (DOE).

The full-scale vitrification process was tested using nonradioactive, simulated wastes at the West Valley Site from 1984 to 1989. (10) This testing developed the operating parameters and generated the waste-form acceptance data required by the DOE Office of Civilian Radioactive Waste Management. (11) The full-scale campaign, summarized in Table V, represents a significant step in the understanding of HLW vitrification processes. In fact, the quantity of glass melted during this testing represents approximately one-third of the total waste glass production planned during radioactive demonstration. The later stages of this testing sequence demonstrated that the Vitrification System consistently produces canistered waste-forms with predictable chemistry and properties. This was a milestone toward establishing that the radioactive waste glass product will meet or exceed all of the DOE acceptance criteria and was key to enabling the WVDP to be the first site to obtain DOE acceptance of its Waste Form Qualification Report (WQR) in August, 1995. As the WQR presents the data needed to demonstrate compliance with the DOE waste-form acceptance requirements, its completion is one of the most important accomplishments in the progression toward authorization for the radioactive demonstration campaign.

Table V

The full-scale testing also confirmed that controlling the oxidation state of the melt pool is an important process parameter for wastes containing multivalent elements. The oxidation state in the melt is inferred by measuring the ratio of ferrous to ferric ions in the glass. If the melt becomes too oxidizing (Fe+2/Fe+3 0.01), a stable foam layer develops at the glass surface, diminishing the glass production rate. Alternatively, if the glass becomes overly reduced (Fe+2/Fe+3 > 1), electrically conductive phases precipitate from the glass that interfere with the electrical current flow within the glass pool.

To avoid these extremes, a desired operating range of 0.05 Fe+2/Fe+3 0.5 was established to achieve optimum processing conditions. As there is no on-line method for determining this ratio under radioactive operating conditions, a function was developed to relate the glass oxidation state to the concentrations of the oxidizing and reducing species in the melter feed slurry.(12) This function, termed the Index of Feed Oxidation (IFO), is defined as:

Eq. 1

where [NO3] is the feed concentration of nitrates, f is the slurry solids fraction and [TOC] is the feed concentration of total organic carbon. The correlation between the IFO and the logarithm of the iron ratio data measured during the full-scale campaign is shown in Fig. 2.

Fig. 2

Nonradioactive waste glass melting operations resumed in early 1995, using a one-eighth scale melter, to augment the training of the radioactive campaign operating staff. One objective of these tests was to confirm the IFO functional response using an updated simulated waste. These tests showed that the glass oxidation state responded more rapidly to IFO changes than before, as shown in Fig. 2. The difference was traced to the presence of nitrite salts in the revised waste simulant. The nitrite salts are now included in the simulated waste to reflect the decision to add sodium nitrite as a waste tank corrosion inhibitor. Using the revised iron ratio response curve, and accounting for differences observed previously between the full-scale and mini-melter systems (13), a target IFO value of 3.0 was selected for the full-scale Vitrification Facility startup tests. This IFO value has consistently produced glass with a Fe+2/Fe+3 ratio of 0.1 to 0.15, which is well within the desired oxidation state range.

Laboratory experience has shown that both reducing the oxidation state of waste glasses or slowly cooling them can affect the PCT measurement. Crucible tests were performed with Ref. 6 glass to verify that neither of these variables significantly affect the PCT results, relative to the EA glass standard. Results from PCT testing of Ref. 6 glass, reduced to an iron ratio of 0.3, and heat-treated for extended periods at 800, 700 and 600C are listed in Table VI. The actual canister centerline temperatures are in the 800C temperature range for less than a hour, the 700C range for approximately three hours and the 600C range for five hours. These heat treatments produce less than 4.5 volume percent crystals in the glass, as measured by x-ray diffraction. The table clearly shows that these processes have only a minor effect on the PCT response of the Ref. 6 glass with respect to the EA glass acceptability standard.

Table VI

The PCT testing described thus far was performed using the correct glass chemical formulation, generally with nonradioactive isotopes of the elements. Given the vitreous nature of glasses, potential damage to the waste-form matrix from radiation exposure is not expected to impact the PCT performance. To enable the DOE to perform relevant glass performance studies with the actual WVDP waste-form, WVDP waste samples have been melted to form Ref. 6 glass at the Pacific Northwest Laboratory Materials Characterization Center. This glass will be shipped to the Argonne National Laboratory for inclusion in their long-term glass performance testing program.

CONCLUSION

In summary, we have successfully demonstrated the performance of the technology and process for transforming simulated waste slurry into a canistered, borosilicate glass waste-form. Based on the one-eighth scale melter data, the IFO of 3.0 was determined to be an optimum condition for melter operation. This determination was confirmed using the full-scale system during Vitrification System startup testing.

The WVDP testing has also shown that the target glass composition, Ref. 6, meets all of the acceptance requirements for a geologic repository. Crystallization remains low in this formulation under very slow cooling conditions and this minor devitrification does not affect waste-form leachability. The PCT leaching data on the target glass are almost an order of magnitude lower than the DOE acceptance standard specified in the WAPS. Under all process conditions, the viscosity and electrical resistivity of the glass is within the processing range and has been

shown to provide adequate melting and production rates in the full-scale production melter.

This data has enabled official DOE acceptance of the WVDP Waste Qualification Report. Following successful completion of the Operational Readiness Review Process, the West Valley Vitrification System will initiate fully radioactive operations.

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START-UP/TESTING OF VITRIFICATION AT THE WEST VALLEY DEMONSTRATION PROJECT

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ABSTRACT
The West Valley Demonstration Project (WVDP) vitrification start-up effort, like other start-up efforts, transformed many challenges into success stories. An integrated schedule, in conjunction with regular meetings to resolve issues, kept the start-up effort on track. The use of summary tables greatly facilitated overviews of the test program and enhanced its credibility. The WVDP used operators to perform the testing. This was the right decision for the WVDP due to the high degree of finesse required to operate the Vitrification Facility. The WVDP committed early in the test program to perform around-the-clock testing. This was a great way to jump start the test program. Portions of testing and construction were performed in parallel. Up front planning to facilitate parallel construction and testing was proven to be effective at the WVDP.

INTRODUCTION

The initial operation of major equipment and systems is always a challenge and the integration of systems to result in a fully operational facility is usually an even greater challenge. This paper will not dwell on the challenges of start-up; it will, however, present information with regard to the WVDP vitrification start-up program. Hopefully, readers can utilize this information to emulate some of the processes that went well at the WVDP. Specific solutions to problems will be presented. Start-up management will be discussed as well as the testing documentation. Finally, a critique of the key decisions will also be presented.

SPECIFIC PROBLEMS AND SOLUTIONS

This section reviews specific solutions to problems at the WVDP.

Melter Dry-out Improvements

Prior to heating up a ceramic melter, the refractory must be "baked" free of moisture. The previous WVDP melter was "baked" by flowing steam through the cooling jacket of the melter. The baking process for the current WVDP melter was substantially improved. The new process injected hot/dry air into the main melter cavity. This sounds like a minor difference, but it resulted in a 50% schedule improvement for the dry out -- a savings of two precious weeks in the schedule.

Like the Word "Temporary"

Commencing the start-up program as early as possible requires an evaluation of the cost and time for temporary equipment versus waiting for the permanent equipment. The WVDP discovered several opportunities in which substantial testing progress could be made with simple temporary measures. A very basic, temporary blower was used to simulate the permanent Off-gas blowers months before the permanent Off-gas blowers were ready for operation. The temporary blower provided the same flow rate and negative pressure as the permanent Off-gas blowers; therefore, the temporary blower was sufficient to commission the In-cell Off-gas equipment, verify tank-to-tank transfers at the design negative pressures, perform initial slurry testing, verify turntable in-leakage, and commission melter vacuum controls. This saved over a month in the schedule. Some small, temporary programs were used to check instrumentation and electrical inputs and outputs very early in the program. Load banks were used to checkout much of the electrical power system early in the program. For example, the start-up heater power system, discharge heater power system, main melter power supply, Off-gas heaters, and diesel generator were all commissioned early using load banks.

Alarm Management During Testing

One of the great challenges of start-up is to provide operators with accurate, relevant information. Obviously, alarm management is a very important part of the challenge. If left unchecked, nuisance alarms can easily deter proper response to important alarms. The WVDP implemented a program to inhibit alarms for nonoperating systems that utilized the control system's delete feature, as appropriate, and conditionalized alarms to the maximum extent practical. These efforts maintained the alarms in the Control Room at a level that permitted proper response. Inhibiting alarms was cautiously implemented. The establishment of correct criteria for obtaining permission to inhibit an alarm or group of alarms, mandating a daily review of all inhibited alarms, and properly using temporary plant modification procedures results in a safe, sound program.

Using the control system's delete feature was approached with even more caution than inhibiting alarms; but once the feature is properly understood, it makes perfect sense. The easiest way to explain the delete feature is with an example. Say, a pump fails and a low-pressure alarm results. The operator acknowledges the alarm and makes the appropriate alarm response. Upon investigation, the operator discovers that the pump requires repair. The operator then initiates the delete feature for the low-pressure alarm. The alarm will automatically be reset when an acceptable pressure is sensed in the system or, in other words, when the pump is fixed and is operational.

Conditionalizing alarms is activating alarms in accordance with the operating status of the primary component in the system. For example, if the HVAC fans are shut down, there will be a low-pressure differential across the filters. The purpose of this alarm is to indicate potential problems with the filter. At zero flow through the filters, the pressure differential will always be zero. Therefore, the alarm would be of no value and should not distract the operator from other important information.

Through much of the test program, the WVDP was able to manage alarms to a degree that a day's (24 hours) listing of alarms was less than a page.

Level Detection in a Slurry

This problem was actually solved during an earlier melter testing period at the WVDP. Current testing verified that the solution was and remains a good one for an age-old problem. The engineers involved in solving this problem researched many sophisticated level-detection devices before settling on a simple device. The level-detection instruments in the slurry tanks at the WVDP are bubblers. As might be expected, the major problem with bubblers in a thick slurry is that they become plugged. The solution is simple: unplug them while they are easy to unplug. The WVDP bubblers are serviced with air, water, and steam blowdown capability. The air and water blowdowns are automatically performed at predetermined frequencies. Steam is available when more aggressive cleaning is required.

Managing a Short-term Hazardous Waste Stream

In the WVDP vitrification system, the waste is concentrated (boiled) prior to glassmaking operations. The 15,000 liters of condensate produced during each batch of waste will be directed to the installed, low-level waste treatment system once radioactive system tie-ins are made. During testing, this condensate needed to be collected and disposed of using an alternative method. Even though the fluid was condensate, it was

characterized as hazardous wastebased upon pH and the potential for hazardous metals. The first step in implementing the alternative method for waste handling was to modify an existing tent so that it could accommodate two 30,000-liter tanker truck trailers. A working tanker and two spares were determined to be necessary due to the aggressive testing schedule and the need to ship out waste as soon as possible. An inflatable berm was purchased and installed, and a temporary delivery system was designed, installed, and connected to the tankers. The company that provided the tankers was also a waste disposal company; therefore, the contract was written to include provisions for them to transfer the contents to another tanker for off-site shipment. This solution was simple and quickly implemented, and just as simple and quick to decommission.

Progressive Levels of Difficulty

Progressing from simple tasks to more difficult tasks is a basic practice. It worked so well at the WVDP that it is worth discussing. We have heard over and over again, learn to walk before you run. The WVDP test program adapted this philosophy and started with the basics, then progressed to increasingly difficult operations. Calibrations, component testing, and commissioning of utilities led to system testing with water and control system testing. Once systems passed the water testing, testing with slurry was initiated. After slurry testing was completed, integrated testing was initiated. Simple, but effective!

MANAGEMENT SYSTEMS

Agility is one of the key words for success in the '90s. The need for agility was realized throughout the start-up effort at the WVDP. Just about the time a new strategy was implemented and starting to work smoothly, circumstances required that it be thrown out for an even newer strategy. Strategies required changing as the Facility ownership transitioned from construction to testing and changed again as Facility ownership transitioned from testing to operations.

The initial start-up efforts were performed by a small engineering group. This group had previously supported full-scale melter testing. Several test engineers were hired from U.S. Navy programs to supplement this small group. This single group directed the start-up testing of the Sludge Mobilization System (THOREX transfer and zeolite transfer were accomplished using this system) and vitrification's Cold Chemical Facility. As the start-up effort began to grow, the testing organization was then substantially strengthened with the addition of managers, engineers, and operators from other areas of the site. This enhanced organization performed water testing, slurry testing, melter start-up, Ex-cell Off-gas system testing, HVAC testing, and integrated operations. A few changes were made internally to this organization to support changing needs. These changes were generally made as the needs for field engineering versus the needs for producing testing paperwork changed. The design organization spearheaded significant modification efforts during this period. The testing/operations organizations supported the design organization's efforts with a team of engineers to facilitate lock-and-tag support and a team of engineers devoted to work group coordination. The majority of the modification work was performed by a time-and-materials contractor, though smaller, specialized contracts were written to handle several of the unique modifications.

One of the major tools used by the WVDP management team was coordination meetings. The meetings, in general, had very specific objectives and

appeared or disappeared as needed. Most of the meetings had a tie-back to the integrated schedule. In fact, my opinion is that the stronger the connection between the meeting and the integrated schedule, the more effective the meeting. The following discussion will timeline the key meetings. The WVDPstart-up effort can be divided into four major periods. They are:

1. Heavy construction, light testing
2. Heavy testing
3. Transition to operations
4. Operations proficiency, cell closure, hot tie-ins.

During the heavy construction, light testing period, a Plan-of-the-Day meeting and a Weekly Production meeting were the major meetings used by WVDP management to monitor and plan the start-up activities. The activities statused in the Plan-of-the-Day meeting were the activities from the integrated schedule due to be completed in the near term. The Weekly Production meeting looked ahead several weeks to identify and resolve items with unacceptable schedule float. The integrated schedule was updated weekly in support of the Weekly Production meeting. A weekly testing/operations interface meeting was also held to maintain a monthly testing plan. A senior and middle management meeting was held twice a week to facilitate resolutions to tough issues. This meeting was affectionately called the Barrier Busters meeting.

The heavy testing period required the formation of one additional meeting. A Preplan-of-the-Day meeting was developed to capture the overnight testing activities, plant status, items requiring repair, and to status construction interfaces. Also, the weekly testing/operations interface meeting transitioned into a daily construction, maintenance, operations, and testing interface meeting. This meeting gathered information to develop the worksheet for the next 24-hour period. Activities were listed in order of priority. Activities listed in this meeting were only those items requiring operator action. That is, testing and lock-and-tag activities. Construction submitted service requests during this meeting as the initiator to any operations activity required to support the construction activity. Test engineering support in the field was increased by a factor of four during this period to provide the required level of technical knowledge and to facilitate prompt resolutions to problems.

The transition-to-operations period continued in all the aforementioned meetings with the exception of the Weekly Production meeting. Changes occurred too rapidly for a weekly meeting to be beneficial; therefore, weekly meetings, such as the Weekly Production Meeting, were cancelled. Also, the Preplan-of-the-Day meeting was transitioned from a testing meeting to an operations meeting. An integrated operations run plan has become the controlling document for plant operations.

As this paper is being prepared, the transition-to-operations period is ending. Some activities of the operations proficiency, cell closure, and hot tie-ins period have started. A run plan will continue to control the operation of the plant. Longer range planning meetings are resuming.

In summary, the management tools transitioned from longer term during heavy construction and the initial testing efforts, to very short term during heavy testing, and then back to longer term for the operators' proficiency, cell closure, and hot tie-ins period. In other words, as the predictability of events decreased, the speed in which the organization addressed potential delays increased. For example, the number of

technical people in the field increased substantially during heavy testing. The time at which technical support was needed could not be predicted and the area of the plant which required technical support could not be predicted; therefore 24 hour-a-day technical support for all areas of the plant was provided.

TESTING DOCUMENTATION

System description documents were written for all the major vitrification systems. Included as part of these documents were the system/component test requirements. Test Instruction Procedures (TIPs) were written, based upon system description requirements, to direct and complete each performance test. The WVDP was aware of the fact that the test program required a high degree of credibility and outside reviewers would want some assurance that: all of the test requirements specified by the system description documents were met; the data from the tests were reviewed by the test engineer, system's engineer, and the Joint Test Committee; and all test results were found to be acceptable. To help achieve this credibility objective, the start-up test organization required a test results table be included at the end of each TIP. This table lists the test requirements, test results, name of the test engineer, name of the system engineer, and Joint Test Committee concurrence that the test results were satisfactory. Taking the credibility objective a step further, the start-up test organization developed a matrix that listed all the test requirements, the TIPs, and the dates that the TIP results tables were approved by the Joint Test Committee. All personnel assigned to review the test program were provided copies of the testing matrix and any test results tables as were requested. The start-up test program was found to be satisfactory by all of the reviewers.

HINDSIGHT IS 20/20

The following decisions were key to the success of the WVDP start-up program:

A. Using the WVDP operators to perform start-up testing.

B. Starting 24-hours-a-day, 7-days-a-week testing, even with a cushion of over one month of float in the schedule.

C. Testing in parallel with construction.

This section presents a critique of these decisions.

The major obstacle with using the WVDP operators for testing was that they were required to perform two critical priorities essentially simultaneously: training and testing. Testing is so demanding that tolerating delays due to Operations personnel performing training objectives becomes very difficult. On the other hand, it was apparent that the experience that Operations personnel gained during testing and by working through some of the testing problems was very valuable. Training cannot be complete until operators can demonstrate that they are proficient using complete and accurate operating procedures. Complete and accurate operating procedures, however, are not always available until system testing is complete and the procedure has been validated using the actual system. Therefore, a very strong argument can be made for testing with a contractor and then training the operators on a working machine with actual working documents. The missing part of the discussion, so far, is the finesse items: manipulator, crane, and Distributed Control System (DCS) operations. Productivity and efficiency on these items require more than a few pass-throughs under ideal or close-to-ideal conditions. Based on the WVDP start-up experience, start-up of a facility that does not require a lot of finesse to test and operate could be

effectively performed by a contractor. A facility, like the WVDP Vitrification Facility, which requires a lot of finesse should be tested and operated initially by the Operations personnel assigned to the facility.

Early in the WVDP test program, Operations announced a commitment to transform to around-the-clock shift coverage. Initially, start-up test engineering was concerned about this decision. How could our engineers support that much testing in the field when the need for test procedure writers was at a maximum? Could enough test paper be produced fast enough to avoid the phrase, "We're waiting on paper?" Test engineering started sprinting; that is, the cycle time for writing and reviewing test procedures was cut in half. This set the pace for the test program. What a way to jump start a program!

Performing work in parallel is one of the most common strategies for shortening a schedule. Also, allowances must be made to let Construction fix things during testing. Therefore, construction and testing must be performed in parallel and the only question remaining is how to accomplish both effectively. My answer to this question is that planning for parallel construction and testing needs to start at a project's conceptual design phase. At the WVDP, the sequencing of the Cold Chemical Facility, the main Vitrification Facility, and the Ex-cell Off-gas system facilitated parallel testing and construction activities. All projects fight to control the seemingly never-ending need for change and the management of punchlist items. The WVDP formed a middle management group to control changes. This group evaluated all changes to baseline documents against specific criteria and any change that did not meet the criteria was either cancelled or sent back for reevaluation.

CONCLUSION

Base the management of start-up activities on a good integrated schedule. Provide an easy method for reviewers to review your test program. Provide a fast pacesetter for your test program. Innovative minds will find a way to keep up. Facilitate parallel construction and testing at the conceptual design stage. And, finally, if your facility requires a great deal of finesse to operate, heavily involve the operators in start-up activities.

Session 18 -- FIFTY YEARS OF NUCLEAR WASTE MANAGEMENT POLICY AND PRACTICE I

Co-chairs: Stephen P. Cowan, USDOE
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18-1

THE DEVELOPMENT OF WASTE MANAGEMENT POLICY - AEC TO DOE

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ABSTRACT

The goal of the Federal government's policy for the management of radioactive waste from its nuclear materials and weapon programs was, and is, to protect the health and safety of those working in these programs, the members of the public and the environment. What has changed since the days of the Manhattan Project are the specific technical solutions pursued, the implementation schedules and the projected cost to the American taxpayer. Although some of these changes are due to non-technical and political issues, many have been the result of knowledge gained from research and development programs, scientific over-optimism,

and an unrealistic appreciation of the cost and schedule impacts of required stakeholder involvement.

The fundamental policy goal established at the outset of this activity, namely the protection of people and the environment from the real and potentially harmful effects of these materials, is still valid. But the challenge that faces the U. S. government, and all who are committed to the technically sound solution of this national problem, is to see that its implementation continues to be based on good science and sound public policy. This will only be possible if we work patiently, and relentlessly, with those in our government who bear the program responsibility and all other stakeholders, reserving our passion for the attainment of the goal, not for any particular solution, facility or schedule.

This goal of this paper is to review the development of waste management policy from the early days of the Atomic Energy Commission to the present. The current generation of policy makers, and their critics, will be greatly aided in their tasks if they have a general understanding of what events have occurred over the past half century which have led us to the current situation.

The past fifty years have clearly demonstrated that it has been far easier to get agreement on general policies than to get political and public agreement to actually implement the resulting permanent solution. This has apparently been true because policy is an internal decision within the government/technical community while implementation is an external decision involving many more players and is often more heavily influenced by non-technical factors.

OVERVIEW

The need to protect people and the environment from the potentially harmful effects of the products of nuclear development was recognized from the outset of the nuclear age. In the earliest days of the Manhattan Engineering District (commonly known as the Manhattan Project), and the basic research and development activities which preceded it, the scientists and engineers gained a healthy appreciation of the hazards inherent in what they were doing and applied what they believe were prudent management policies, based on the best information available to them. With the limited amount of available information expanding rapidly, and with the pressures of the political, military and security aspects of the nuclear program ever present, they did an excellent job.

With the creation of the Atomic Energy Commission (AEC) in 1946, and the end of the frenetic wartime rush to develop the atomic bomb, a conscious, serious attempt was made to establish an effective policy basis and specific implementation plans for the management of the growing volume and diversified spectrum of wastes being disposed of or being placed into engineered storage. The "Cold War" period did not bring the hoped for period of calm. Instead, it brought a continuation of a sense of urgency to the programs responsible for developing nuclear weapons, peaceful uses of atomic energy, and other applications of a rapidly growing nuclear technology to the AEC, the Energy Research and Development Administration (ERDA) and the Department of Energy (DOE).

These agencies were also responsible for continuing and improving the waste management programs started earlier. Their tasks became more difficult as the volume and variety of waste steadily grew, and the public interest in their work and public expectations similarly increased. Over the past fifty years, and especially in the past few,

much more previously classified information has been released to the public. During World War II and throughout the Cold War period, almost all information related to nuclear weapon production was closely guarded and certain waste management data fell into this category.

The various agencies created and recreated internal organizations specifically charged with the responsibility to establish sound policies and effective programs to carry these policies out (Fig. 1). Each reorganization was intended to better focus the growing level of resources, both in number of personnel and funding, to speed the program along and to improve the safety and efficiency of the activities. The agencies also had to implement bureaucratic procedures, arising from Congressional actions to assure that environmental, safety and health aspects were given adequate consideration in policy and programmatic decisions, and also to take into account and respond to the concerns and interests of potentially affected stakeholders.

Fig. 1

It is of little benefit to anyone to now attempt to judge past actions by the current state of knowledge of the health, safety and environmental impacts arising from the management of these wastes and other nuclear materials. If judging is the aim, then conclusions must be reached based on what was known at the time decisions were made. The goal of this paper is to review the development of waste management policy from the early days of the AEC to the present state of affairs within the DOE. This paper will briefly discuss the policy which was established by the AEC, ERDA or DOE and comment on the context in which it was established. Primary attention will be placed on the management of high-level waste (HLW), but transuranic (TRU) and low-level waste (LLW) management will also be discussed. The formal definitions of these waste types are shown on Fig. 2.

Fig. 2

The discussion will not be complete, that is a task of genuinely Herculean proportion. Additionally, it is not of great interest or import to ongoing public discussions of what is currently planned by the DOE. What is needed by the current generation of policy makers, critics and other commentors is a more general understanding of what has transpired in the U.S. waste management policy world over the past fifty years. Reference will be made to selected AEC, ERDA, and DOE documents which chronicle the development of waste management policy throughout the past fifty years. Hopefully, this discussion will promote a better understanding of the policy evolution that has occurred and put it into context, so the government, public and private sectors can rationally approach the creation of future policy. It must be clearly understood that policy and the means of implementing it are subject to a constantly expanding information base and impacted by change. Public policy and implementation methods must be subjected to periodic re-examination and all involved groups must retain an open mind about changes which reflect the latest and best information available.

WASTE MANAGEMENT UNDER THE AEC

The policy and related implementation decisions concerning HLW were first made by the AEC and, before its establishment in 1946, by the Manhattan Project. In the very beginning in 1942 there were few detailed health and safety requirements, but the HLW was clearly recognized as being extremely hazardous and the program took what was, at that time, considered to be a very safe approach. A few examples are the

establishment in 1950 of the "as low as reasonably achievable" principle, the setting of specific exposure limits for normal operations, and the decision to store rather than to immediately dispose of the most potentially dangerous liquid waste streams.

HLW was also seen as a valuable resource, due to the amounts of valuable uranium and plutonium contained in it. The waste was stored as a neutralized or basic solution in underground steel tanks, normally located within concrete vaults, with what were then considered to be appropriate environmental safeguards. This was the practice that was followed as the Hanford and Savannah River facilities were developed. In the 1960's the Idaho facility initiated stainless steel tank storage of acidified HLW, processing it into a solid form, and storage in large steel bins within concrete structures.

Research and development programs were started to focus advanced waste processing approaches to move from interim storage in buried tanks or bins to an approved final disposal facility. Within the interim storage programs at the several sites, designs progressed from single shell tanks to increasingly more secure versions (e.g. cup and saucer designs) leading to the current double shell tanks designs.

The AEC was assisted in its waste management program from 1955 to 1965 by the Committee on Geologic Aspects of Radioactive Waste Disposal of the National Academy of Sciences - National Research Council (NAS-NRC). The NAS-NRC served as advisors to the AEC's Division of Reactor Development and Technology (RDT), which was responsible for the AEC's commercial nuclear power program. The NAS-NRC's responsibilities were the geologic aspects of RDT's R&D program; however, they broadened their charter and considered all aspects of ground disposal of radioactive waste and drew conclusions on overall waste management practices.

In May 1968, the General Accounting Office (GAO) reviewed the AEC's HLW program (1) and concluded that the AEC "needed to devote more vigorous attention to advancing the technology required to permit long-term storage at its Richland and Savannah River sites." GAO also suggested that the AEC establish a Headquarters office vested with the responsibility for policy making and overseeing radioactive waste management activities. Such an office was established in the Spring of 1970, and has continued through many reorganizations to this day in both the Nuclear Regulatory Commission (NRC) and the DOE.

AEC policy recognized from the outset that the practices for handling HLW need not be uniform at all sites. This policy was confirmed by a Task Force established by the AEC General Manager in 1968. That Task Force's conclusions, which were subsequently approved by the Commission, stated that, in view of the differing nature and status of the programs, there was no need for uniformity between programs at AEC installations and those prescribed at the commercial facilities that were licensed by the AEC (2).

High-Level Waste (HLW)

The AEC defined HLW by reference to its origin (i.e. liquid waste from the processing of irradiated spent fuel, solid waste derived from those liquids, irradiated spent fuel which won't be processed, or materials with an equivalent hazard potential). This "genetic" definition of HLW, based on the process by which it was created, has effectively continued to the present time. This waste was normally generated at AEC facilities during the chemical reprocessing of irradiated production reactor fuel in order to recover the weapon grade materials and unused uranium. Similar

waste was also generated by the processing of commercial reactor fuel to recovery unused uranium for recycling back into the commercial fuel cycle. The volume of HLW generated and placed in storage grew during the peak Cold War years and then decreased and has effectively stopped. We are now left to contend with this backlog or "legacy" waste.

The basic 1968 AEC policy (2) for managing HLW was "... that all liquid high-level radioactive waste should be suitably contained with adequate provision for control or recovery in the event of leaks or accidental spillage." It should be noted that this policy has not materially changed in almost thirty years. The AEC policy further stated that "Storage of such wastes as liquid in storage tanks shall not be regarded either as disposal or as an acceptable practice for long-term handling; rather, waste management programs should provide for either a) reduction of such wastes to solid form for long-term storage, or b) transfer of such wastes to long-term storage in deep underground locations, either approach to provide high assurance of isolation of wastes from the biosphere, be resistant to credible internal or external forces, and not be dependent upon mechanical cooling methods. Unless the Commission authorizes an exception, such long-term storage methods should not preclude removal of the wastes from such long-term storage locations for relocation should such ever become necessary or desirable."

The GAO reviewed the AEC program again in 1971 and concluded "Although AEC has assigned a high priority to radioactive waste management problems, GAO believes that the level of effort given to these programs should be increased in view of their extraordinarily complex characteristics. The problems and delays being experienced are attributable primarily to a need for definitive technology and such matters as the relative merits of alternative practices and proposals for interim and long-term storage."

These policies and planned implementation programs were contained in the 1973 Plan for the Management of AEC Generated Radioactive Waste (3). The plan guided the field offices in updating their individual waste management plans and their budget requests to implement the policies and planned programs defined in it. In addition to the policies discussed above (conversion to a solid form and isolation from the environment under all credible accident conditions with minimal reliance on perpetual maintenance and surveillance) the plan also established policies on the interim storage of liquid HLW in engineered storage systems, the requirement of retrievability, the need for spare tankage in case of leaks, and the need to reduce the volume in storage.

Low Level Waste (LLW)

The AEC defined low level waste (LLW) as basically all radioactive waste that didn't fit under the HLW definition (a definition by exception). Its disposal was to be by shallow land burial in a facility and geology which would isolate it from the general environment for its hazardous lifetime, assumed to be about 300 years. The facility was to be either on Federal or State owned property. Initially, the AEC accepted the small volume of LLW that was generated by the fledgling commercial nuclear industry. Starting in the early 1960s the AEC (and later the NRC) licensed six commercial low level waste disposal sites across the country, and the AEC even sent some of its waste to these sites. In the late 1970s ERDA stopped using the commercial sites and directed all ERDA LLW to its own disposal sites.

Transuranic (TRU) Waste

In 1970 the AEC defined Transuranic (TRU) waste, a new class of waste which required special management considerations. They defined TRU waste as LLW contaminated with long lived transuranic isotopes (primarily of plutonium and americium) at or above a 10 nanocuries per gram level. The primary isotope of interest was Plutonium-239 which had an approximately 24,000 year half life. The AEC directed that TRU waste should be packaged and stored for a minimum 20 year period in a retrievable manner for later disposal in a geologic repository which would isolate it from the environment over its hazardous lifetime.

WASTE MANAGEMENT UNDER ERDA

When ERDA was established by the Energy Reorganization Act of 1974, it assumed responsibility for the AEC's waste management policies and programs not given to the NRC, which was created at the same time. It continued the AEC policies and programs, but now with the increasing involvement of the Environmental Protection Agency (EPA) and the public, following implementation of the National Environmental Policy Act (NEPA) in 1970. NEPA required a much greater degree of openness and the disclosure of many aspects of waste management plans and operations which previously had been classified or closely controlled.

Major advances were made in the 1970's in implementing the programs for improving interim storage of HLW and the desired transition to the geologic disposal of stabilized, volume reduced HLW. NEPA-required documents were issued by each of the major ERDA sites detailing their specific plans. In 1976 ERDA issued "Alternatives for Managing Wastes from Reactors and Post-Fission Operations in the LWR Fuel Cycle" (4) which led to many fruitful technical and policy discussions, conferences and symposia.

High-Level Waste

No significant changes were made in either the policy or the routine implementation practices for HLW under ERDA. Progress was made in improving the safety of the tank stored waste through programs to reduce the volume and mobility of the waste by removal and evaporation of liquids and the isolation of the tanks to prevent the accidental addition of more liquids. Confirmed and suspected instances of tank leakage were dealt with and questionable tanks were emptied of removable waste. Additional double shell tanks were built and all newly generated wastes, and wastes removed from unsound tanks were routed to them for safe storage.

Low Level Waste

No significant changes were made in the definition of LLW or ERDA's management practices, but closer, cooperative ties were developed between ERDA's LLW operations and those of the commercial sector. In the 1960s and 1970s, both ERDA and the NRC, along with the EPA, supported the Waste Management Task Force of the Conference of Radiation Control Program Directors, a group of state officials responsible for regulation or oversight of the six commercial low level waste disposal sites. They provided a forum in which the states could benefit from the exchange of information from the experiences of the AEC and ERDA in LLW disposal since the Manhattan Project.

Transuranic Waste

Studies were made under ERDA to investigate and resubstantiate the threshold concentration level of transuranium isotopes in waste which defined it as TRU waste. After an extensive technical review, the concentration level was changed from 10 to 100 nanocuries per gram and

the formal definition change was issued by DOE in 1981. The management of TRU waste, along with commercial HLW, was discussed at length by ERDA in 1974 in WASH-1539, "Management of Commercial High-Level and Transuranium-Contaminated Radioactive Waste", a document that was widely reviewed and commented on by the technical community, industry and the public (5). Previous site-specific waste management activities certainly interested those communities and citizens in the immediate vicinity of the AEC/ERDA sites. However, these national programs (especially the nation-wide search for geologic repository sites which gained increased momentum in the middle 1970's) developed a broader audience of stakeholders interested in the ERDA's policies and programs for the long-term management of its still growing inventory of long-lived radioactive waste.

It was during the ERDA era (and the early days of the DOE) that the radioactive waste management program was broadened to include hazardous waste, and the complex problem of "mixed waste" first arose. Mixed waste is waste containing both radioactive materials (regulated under the Atomic Energy Act) and hazardous materials (regulated under the Resource Conservation and Recovery Act). ERDA was self-regulating for the radioactive component, but subject to regulation by the Environmental Protection Agency (or individual states) for the hazardous component. The jurisdictional issues associated with dual regulatory situation with mixed waste are still being sorted out.

WASTE MANAGEMENT UNDER DOE

In 1977, the Federal government again reorganized itself and formed the DOE. The search for a permanent organizational home, with the responsibility for establishing and executing DOE's waste management program, finally led in 1989 to the creation of DOE's Office of Environmental Restoration and Waste Management, later renamed the Office of Environmental Management (EM). The many attempts to document DOE's policies and plans to the public and other Stakeholders led to the creation of the annual Five-Year Plan, Site Specific Plans and Activity Data Sheets. The annual update to the Five-Year Plan has since been replaced by the Baseline Environmental Management Report (BEMR).

DOE also formed the Office of Civilian Radioactive Waste Management (RW) which was given the responsibility for the establishment of a disposal facility for HLW and spent fuel from the commercial sector. Eventually, the disposal facility will also accept DOE immobilized HLW. The RW program will be covered in detail in other papers.

During the same time period, the growing influence of the EPA and states, which had regulatory powers over some DOE operations, became major factors in the selection and implementation of the DOE waste management program. Under DOE, and earlier under ERDA, lawsuits became a popular enforcement tool and often led to government commitments to specific acts and remedy implementation schedules. The 1980 Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), the 1984 Resource Conservation and Recovery Act (RCRA), and the 1992 Federal Facility Compliance Act (FFCA) are examples of legislation which have often been the foci for actions against ERDA and DOE by EPA, the States and others groups and individuals.

DOE's response to these pressures has been to devote a much greater level of attention (and funding) being devoted to openness and proactive communications with stakeholders. DOE's attempts to understand and respond to the concerns of its critics, and the budget restrictions that

have developed as the EM program has matured, have contributed to some delays in implementation.

THE CURRENT SITUATION

After fifty years of dealing with radioactive waste management, these are the basic policies:

- HLW: The 1968 AEC policy (i.e. safe interim storage, volume reduced and immobilized waste form, leading to geologic disposal) is the basis for DOE's program.
- TRU: The 1970 AEC policy (i.e. retrievable storage leading to geologic disposal) is the basis for DOE's program.
- LLW: The AEC policy (i.e. near surface disposal in geologic media which prevents the transport of waste to the general environment, for its hazardous lifetime, on Federal or state land) is the basis for DOE's program.

The expected implementation dates for the geologic disposal of HLW and TRU waste have slipped significantly from the days of the AEC to the present and the estimated costs have grown enormously. 1996 should see the initial operations of two HLW vitrification plants at Savannah River and West Valley. The date for the opening of a geologic repository is another question, with perhaps as many political as technical issues involved. No one in 1960 would have believed the amount of money spent in the last 35 years to address the HLW disposal problem, and it's not finished yet.

The schedule for the geologic disposal of the government's TRU waste has also slipped and the estimated cost also continues to grow. The Waste Isolation Pilot Plant (WIPP) has yet to receive its first waste and may have to re-mine some of the underground space due to the schedule. No government officials now pledge to stay on the job until WIPP opens, as some did in the past. This is not said to criticize, but as a caution to all on underestimating society's inertia when it comes to making profound, long lasting decisions, such as the siting and operation of geologic disposal facilities for TRU waste which will long outlive the decision-makers.

The policy of permanently disposing of long-lived waste garners support from practically all reviewers. The decision to actually implement "permanent" solutions may cause many to pause and relook at these implementation plans. While everyone can understand, and to a small degree condone, this philosophy, eventually society must find the courage to act. If the policy is valid, society must not shirk from implementing it. The permanent geologic disposal policy is technically sound and also sound from a public policy perspective. It is now clearly time to move ahead and support the steps necessary to implement it. Forty years of R&D have increased our knowledge of how the waste and the environment will perform, and recent experience has shown us how to assure that responsible dialogue is conducted with the affected stakeholders.

BUDGET GROWTH AND SCHEDULE SLIPPAGE

In the 1974-1976 period two ERDA Task Forces were very instrumental in establishing a renewed interest in the entire waste management program issue. One led to the issuance of ERDA-33 "Nuclear Fuel Cycle", which resulted in bringing all aspects of the commercial nuclear fuel cycle under one organization head with the establishment of the Division of Nuclear Fuel Cycle and Production. The other led to the issuance of ERDA-48 "A National Plan for Energy Research, Development and Demonstration", which was required by the act which established ERDA and the NRC. This

report explicitly stated the urgency for resolution of radioactive waste management policies and practices for the commercial nuclear fuel cycle. The management of commercial wastes will be addressed by other papers at this conference.

The chronicling of the growth of the budget allocated to implement Federal policy for the management of its radioactive waste is not the primary purpose of this paper, but it is interesting to note what happened to the waste management budget in just the last twenty years. As discussed previously, very little has actually changed in the policies. What has happened has been an enormous attempt to implement both improved interim measures like storage and waste stabilization, conduct necessary R&D, and to initiate final disposal.

Twenty years is a good period to choose for this comparison. In 1976 a series of large, public meetings were held by ERDA to discuss the range of disposal options then thought to be worthy of consideration (including continental U.S. geologic options, sub-seabed disposal, ice cap disposal and even disposal in space). Several technical societies, such as the AIChE, the ACS and the ANS held topical meetings on various technical aspects of radioactive waste management at that time.

The FY 1976 waste management budget totaled $30.6M ($18.6M for ERDA waste and $12.0M for commercial waste). The FY 1977 budget represented an enormous increase to $90.3M ($30.3M for ERDA waste and $60.0M for commercial waste); almost a tripling of the budget in one year. The FY 1996 budget represents an almost one hundred fold increase. The FY 1976 budgets included funding for extensive R&D programs. The development program for four HLW processing technologies were to be completed between 1979 and 1981 and the resulting technologies were to be ready for commercial application between 1983 and 1985. In the HLW terminal storage program (now known as the Yucca Mountain Program), the site for the first of several pilot plants was to be selected by 1978 and begin operation by 1984. In the terminal storage for TRU waste (the WIPP program), the site was to be selected by 1977 and operations started by 1983.

Hindsight tells us that the technical issues facing both the Defense and the Commercial programs were underestimated and adequate thought was not given to how society was going to satisfy the valid interests of the many individuals and groups that these policies would touch. The ensuing twenty years has clarified the primary technical issues, and time has also helped develop an understanding of the value of openness and the rights of our citizenry.

CONCLUDING THOUGHTS

The technical policies and broad program goals for the long-term management of the waste from the Nation's nuclear weapon production and energy development activities have remained essentially constant for over thirty years. The plans for implementing those policies (i.e. the establishment of long-term disposal or storage facilities) have, unfortunately, languished and had tremendous increases in their estimated costs. 1996 will finally see two major advances in implementation of the program to dispose of HLW, the initiation of radioactive operations at the West Valley and Savannah River HLW vitrification plants. While a clear path or an early resolution to the commercial repository situation is not readily apparent, our Nation looks for steady movement leading to the start of waste receipts at the Waste Isolation Pilot Plant yet this decade.

What has been learned from the past fifty years of establishing and implementing policy is that it is far easier to obtain general agreement on what we want to do, than it is to obtain agreement on how to do it. The former decision is reached primarily on the basis of fact, good science and technical judgment. The latter decision has broader criteria to contend with than policy establishment. For the most part, policy is established within the system; decisions on implementation are made in the real world. Program leaders must recognize the fact that in addition to environmental criteria, valid criteria relating to economic, social, cultural and political factors will have an oftentimes controlling role to play. Sensitivity to these criteria will better prepare all of us to assist decision-makers in moving the system from considering implementation to actually doing it.

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18-2

RADIOACTIVE WASTE MANAGEMENT IN THE U.S. - 1950-1970
AN OVERVIEW OF THE AEC PROGRAM AND R&D HIGHLIGHTS

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ABSTRACT

After World War II, the "new world of atomic energy" and the development of peaceful uses of the atom were assigned to the newly formed Atomic Energy Commission (AEC). During the period of 1947-1970, one of the major AEC programs involved the development of nuclear power reactors under the Division of Reactor Development (DRD). A small group of sanitary engineers and scientists was formed within DRD to assist in the siting and operation of power reactors and the development of the Commission's R & D program in radioactive waste management and related environmental research.

During this period, numerous low-level waste (LLW) treatment systems were studied and used at AEC sites, with closely monitored discharges of liquid, solid, and gaseous effluents to the environment. During the 1960's, extensive laboratory and pilot plant studies were carried out at Oak Ridge and Hanford on advanced LLW treatment systems such as scavenging-precipitation ion exchange, foam separation, steam stripping, filtration, two-stage ion exchange, electrodeionization, water recycle, and asphalt solidification. Some of these systems have been used at commercial nuclear power stations. By the end of 1969, more than 20 years of high-level waste (HLW) tank storage experience had shown it to be a practicable means of interim handling. During 1950-1970, an extensive HLW solidification R&D program was carried out at 5 AEC sites; the program culminated during 1964-1972 with the installation of an engineering scale

Waste Solidification Engineering Prototype (WSEP) at Hanford. Three processes (the ORNL pot-calciner; the BNWL spray calciner-melter, and the BNL phosphate glass system) were demonstrated with full-scale radioactivity. Design concepts for a 5 tonne/day commercial reprocessing plant were developed for industrial application. Ground disposal research (1957-1970) for liquid and solid HLW was carried out in 1) bedded salt formations in Kansas; 2) impermeable bedrock at the Savannah River Plant; and 3) deep permeable formations (3,000-12,000 feet) containing connate brine.

INTRODUCTION

The nature of radioactivity as a potential hazard to man and his environment and the importance of satisfactory effluent control were recognized when the atomic energy program was conceived under the auspices of the Manhattan Engineer District in 1942. After World War II and the formation of the Atomic Energy Commission (AEC) in 1946, the initial AEC programs in radioactive waste management and related environmental research and development were planned and carried out under the Division of Engineering, which later became the Division of Reactor Development (DRD). The early programs, commencing in 1947, were formulated in the Sanitary Engineering Section (SES), under the direction and leadership of Mr. Arthur Gorman and Dr. Joseph Lieberman, Sanitary Engineers, with the author joining the program in 1957. Many exciting aspects of the program occurred during the 1960's. Because of time and space limitations, only highlights of radwaste activities across AEC for the 1950-70 period will be presented. Excerpts from the "author's archives", i.e., numerous papers, speeches, internal reports, and book chapters which were authored or co-authored during this time, form the major basis for this paper. The history of the 1947-1970 time frame provides an excellent foundation for the succeeding quarter century (1970-1995), when we consider where we are now, or where we should be.

EARLY DEVELOPMENTS (~1947-60)

We start our historical trek along "nostalgia trail" by noting that sanitary engineering activities in the early AEC days focused on three major areas: 1) radioactive waste handling and disposal, 2) water supply, and 3) environmental concerns. The following activities were directly related to all AEC operations, including the development of a new nuclear energy industry through commercial enterprises:

Siting: An important aspect of the early program was the function of technical consultation in the selection of sites for nuclear energy facilities at AEC and industrial sites. In the 1950's and 1960's, the sanitary engineering staff established working relationships through comprehensive, cooperative interagency programs with the U.S. Geological Survey (USGS), the U.S. Weather Bureau (USWB), the U.S. Coast and Geodetic Survey (C&GS), the Bureau of Mines (BuM), the U.S. Public Health Service (USPHS), the Corps of Engineers, and the Fish and Wildlife Service (USF&WS). The experience and technology of specialists in these agencies were combined with AEC and DRD staff in the selection of suitable sites for all types of nuclear installations such as reactors, chemical processing plants, research laboratories, etc. Example sites included the National Reactor Testing Station (NRTS); the Savannah River Plant (SRP); the Experimental Gas Cooled Reactor at Oak Ridge (EGCR); and the West Valley, NY, Chemical Processing Plant. A special note is made of the geologic and hydrologic programs which were conducted by the USGS. These studies provided essential geohydrologic data and reports as

required for the location, design, construction, and operation of all AEC facilities. The siting and satisfactory operation of these installations required quantitative data and knowledge also on the seismology and meteorology of the proposed site area.

The environmental scientists from the agencies noted above were assigned to various AEC sites and the DRD Environmental & Sanitary Engineering Branch (ESEB, formerly SES) to aid in the data collection required in the assessment of the transport, diffusion, and behavior of radionuclides in the hydrosphere, lithosphere, and atmosphere. Through the 1950's, this expertise was used in the evaluation of proposed commercial reactor sites, their waste handling systems, and all other environmental aspects. Preliminary and final safety analysis reports were reviewed for the Advisory Committee on Reactor Safeguards (ACRS), prior to licensing the construction and operation of electric utility power reactor facilities.

Early R&D: The initial phases of the DRD-ESEB radwaste R&D program are summarized in Fig. 1. In the early 1950's, the program consisted of approximately 20 projects with universities, federal agencies and AEC sites. The possibility of using biologic treatment methods to remove low levels of radioactivity, such as the trickling filter and activated sludge processes, used in municipal and industrial waste treatment, was studied at several universities. Even at this time, a major portion of the program was directed at the high level waste (HLW) problem.

As also shown in the figure, satisfactory working relationships had been established with several river advisory boards, such as the Ohio River Valley Water Sanitation Commission, the Mohawk River Advisory Group, the Savannah River Advisory Board, the Columbia River Advisory Group and the New England Interstate Water and Pollution Control Commission. These groups were primarily concerned with atomic energy installations located on streams within their jurisdiction and with liquid waste effluents discharged to these streams. It was the policy of AEC, and a function of the DRD-ESEB program, to provide these groups with design and operating information on the waste handling facilities at AEC installations, and the associated monitoring, for assuring that water quality was not being deleteriously affected. In addition, the R&D program was guided in its development by consultation with the top scientists and engineers in the U.S., mainly through interaction with several divisions of the National Academy of Sciences (NAS), the National Committee on Radiation Protection (NCRP), and the American Standards Association (ASA).

Program Objectives: In the 1950's, to assure that nuclear energy operations would not affect the health and safety of the public and its environment, the AEC-DRD radioactive waste management R&D program followed three major objectives:

1) Develop practical systems for the safe handling and ultimate disposal of gaseous, liquid and solid wastes with emphasis on waste emanating from chemical processing plants and certain types of reactors;

Fig. 1

2) Evaluate quantitatively dilution and/or concentration factors in nature in order to determine the degree of treatment actually required prior to release of effluents to the ground, atmosphere or surface waterways, thus taking into account a proper economic advantage of these environmental factors;

3) Obtain increased knowledge of the fundamental phenomena and processes involved in disposal of radioactive and toxic wastes to achieve more efficient and economical methods. Although the basic principles of

radioactive effluent control and environmental and public health protection in the nuclear industry were similar to those which applied to other chemical or heavy industries, there was one significant difference between this industry and others. From its inception, the nuclear industry was acutely aware of the potential hazardous effect of its wastes and focused detailed attention on these problems. Waste management in all atomic energy operations must control radiation hazards not only during operations but also after the waste products are discharged to the environment.

Radioactive Effluent Control Operations: Three basic approaches were used:

1) Dilute and disperse was used only with low-hazard-potential wastes, in which the radioactivity could be reduced to acceptable levels (either directly or following treatment) by dilution in nature--air or water. Quantitative physical, chemical, and biological data were obtained continually on such things as dispersion phenomena and reconcentration factors, demonstrating the validity of this approach without compromising health and safety standards. I cannot over stress the favorable siting characteristics of large controlled areas and low population densities, which permitted safe development of the nuclear energy industry within the AEC complex. The procedure has not been used in commercial electric power reactor effluent control.

2) Concentrate and contain is used for HLW originating from chemical processing of irradiated reactor fuel. The radioactivity was concentrated by volume reduction and then confined or isolated in a controlled area such as specially designed underground storage tanks, away from man and his natural resources. Containment is essential for HLW because small volumes would require excessive amounts of environmental dilution.

3) Delay and decay was used to discharge certain kinds of liquid wastes to the ground at suitable locations. Soil holdup or exchange capacity is used in this concept. Either direct discharge to the environment or discharge after conventional treatment were used.

In establishing engineering or operational criteria for satisfactory waste management systems, detailed quantitative consideration was always given to:

The specific nature and quantity of radioactive wastes being handled.

The characteristics of the receiving environment.

The interaction between the two.

The basic radiation protection guides or standards, as established by the Federal Radiation Council (FRC) or the National Council on Radiation Protection and Measurements (NCRP). I cannot overemphasize that these guides or standards were the overall "driving force".

Early Liquid Waste Management: Early development work centered on the handling and disposal of high volume, liquid low level wastes (LLW), airborne particulate wastes and evaluation of the effects of radioactivity in these wastes on public water supplies and domestic waste treatment facilities. Sources of liquid LLW included reactor cooling systems, laboratories, hospitals, fuel fabrication, laundering, and other operations.

Conventional liquid waste treatment systems were studied and used at the AEC sites during the 1950's. These included sedimentation in baffled tanks, seepage ponds, evaporation, decay storage, co- precipitation, ion exchange, and direct discharge to nature without pretreatment; the type of treatment required depended upon the character of waste involved and

the environmental conditions at the site. Examples of this latter operation were liquid waste discharges from Knolls Atomic Power Laboratory (KAPL) to the Mohawk River, the Pressurized Water Reactor (PWR) at Shippingport, PA, and the Savannah River Plant near Aiken, South Carolina. In the late 1950's, the USPHS carried out detailed monitoring studies below each of these plants to determine the fate of specific nuclides discharged in these effluents. These comprehensive studies included the uptake of activity by suspended and colloidal material, by bottom silts, by aquatic organisms, and by fish and plant life--all were assessed to assure that no environmental radiation hazards existed.
In ground disposal operations, the "delay and decay" approach to effluent control was extensively used for large volume, LLW at most AEC sites. For example, large government installations located in isolated or remote areas such as Hanford, Washington, or the National Reactor Testing Station (NRTS) in Idaho, were able to operate safely under less restrictive effluent standards than could be utilized by smaller facilities in urban areas. Various ground disposal systems included retention basins, cribs, swamps, trenches, seepage pits, and an injection well at NRTS. Extensive monitoring systems ranging from over 30 wells at NRTS to over 200 wells on the Hanford reservation were used to determine the distribution and migration of radionuclides in the lithosphere and to assure that the safe capacity of the environment was not exceeded. The USGS, with specific competency in radio-geohydrology, cooperated with on-site scientists and engineers in developing the safe parameters needed for carrying out these operations.

Early Land Burial: One of the more important operating problems, during the 1950's, involved the satisfactory disposal of solid radioactive wastes. Activity levels varied from a few times background to levels requiring remote handling or shielding. Conventional sanitary landfill procedures similar to municipal refuse disposal were used. Burial of such wastes were done at sea and in established burial areas at large AEC production and research sites. During the 1950's, land burial areas were operated at Oak Ridge, Savannah River, Idaho, Los Alamos, and Hanford. Large installations such as KAPL, Argonne National Laboratory (ANL), and Mound Laboratory, Miamisburg, Ohio, all shipped packaged solid wastes to Oak Ridge for disposal. Westinghouse Atomic Power Development (WAPD) at Bettis Field, Pittsburgh, PA, and Brookhaven National Laboratory (BNL) disposed of their solid packaged wastes off the Atlantic Coast through arrangements with the U.S. Navy.

With the end of the 1950's, there was an increasing need by both AEC contractor and licensee operations for establishment of a regional land burial area in the northeastern United States. In December, 1959, a decision was made that regional disposal sites other than the existing AEC installations could be established, as required, on state or Federal Government owned land. As we leave the 1950's, it is noted that geohydrologic surveys were being conducted in several areas of the country for selection of additional burial ground(s). It was hoped that this facility would become a commercial operation, since it was believed that this was one area where industrial participation was feasible and desirable.

Early Sea Disposal: Sea disposal of small quantities of radioactivity had been accomplished since 1946 through arrangements with the U.S. Navy off both the Atlantic and Pacific Coasts. Monitoring investigations of both the Atlantic and Pacific disposal areas revealed no readily

detectable radioactivity from the above operations. Because of increased interest by both waste disposers and regulatory agencies, the Committee on Oceanography of the National Academy of Sciences (NAS) at the request of the AEC, identified a number of inshore disposal areas off the Atlantic and Gulf Coasts. A similar survey of the Pacific Coast sea disposal areas was made during 1959.

Early Airborne Waste Management: In regard to airborne particulate material, it was discovered early that conventional dust control equipment could not meet the extremely small permissible concentrations of various nuclides in the atmosphere. An important milestone in the air cleaning field was the successful development of a high efficiency filter, capable of routinely removing 99.95% of particles 0.3 microns in diameter, and placing the unit into commercial production. Another notable achievement in the air cleaning field was the development of a silver nitrate packed tower unit which was capable of removing iodine131 from gas streams; satisfactory plant operation was obtained with efficiencies greater than 99.99%.

In related work, the scientific achievements of the U.S. Weather Bureau are recognized; this group investigated meteorologic parameters such as atmospheric diffusion and transport in quantitative terms at all major AEC operating sites; these data were used in the over-all design of gaseous effluent facilities, i.e. stack height and location, and also for nuclear safety assessments under normal and accidental operating conditions. Their contribution of "Meteorology and Atomic Energy" in 1954 was a textbook for environmental and radiological safety evaluations.

Tank Storage & Early HLW R&D: By the end of 1959, approximately 65 million gallons of HLW from the chemical processing of irradiated fuel were being safely stored in underground carbon steel or stainless steel tanks at Hanford, SRP, and NRTS. Because of inherent limitations on long-term tank storage as an ultimate disposal system, the AEC-DRD program began investigating, in the early 1950's, the conversion of high-level liquid wastes to a solid form. Laboratory investigations began in 1955 at ANL and later at the Idaho Chemical Processing Plant (ICPP) on a heated fluidized bed technique for conversion of aluminum nitrate wastes (from aqueous processing of uranium-aluminum alloy fuels) to an oxide form. At the end of the 50's, a prototype pilot plant was installed at ICPP to test the process on a large scale. In another pioneering effort, BNL studied the fixation of liquid wastes in solids by adsorbing the radioactivity from the liquid onto montmorillonite clay. The actual fixation of fission products in the clay mineral was accomplished by heating the clay to about 1700oF.

Early Deep Underground Disposal R&D: The AEC ground disposal R&D program received its initial stimulus in September, 1955, when a committee of geologists and geophysicists was established by the National Academy of Sciences-National Research Council. The AEC requested this group to study the possibilities of disposing of HLW on land and to indicate what research was needed to determine feasibility. Similarly, during 1957-58, the American Petroleum Institute (API) was requested to review the problems of deep well disposal and to suggest specific approaches to the problem. This study would guide research, development and field testing operations required for the establishment of a safe and economical high-level liquid waste disposal process. In 1958, the American Association of Petroleum Geologists (AAPG) responded to a request from the AEC to assist in the evaluation of geologic basins as

possible sites for the safe subsurface disposal of radioactive wastes. These three consultations with NAS, API and AAPG (1955-58) provided the support for AEC-DRD to carry out extensive laboratory and engineering field demonstration studies in the 1960's on 1) salt formations; and 2) deep permeable and impermeable formations. This R&D will be discussed further in the second half of this paper.

1959 JCAE Hearings: At this point, note that the planning and conduct of the radwaste R&D program was under a continuing review and assessment by the Joint Congressional Committee on Atomic Energy (JCAE), several committees of the NAS and various national radiation protection groups. Perhaps the most comprehensive set of hearings held on any one facet of the atomic energy program were the ones held during January, February, and July 1959, on Industrial Radioactive Waste Disposal. The author had the pleasure of working with JCAE staff in the organizing and conduct of these hearings. Over 150 experts in all phases of radioactive waste management presented oral and written testimony for these hearings. Representatives from all major AEC production, reactor development and laboratory research installations, universities, electric power, coal and oil industries, the NAS, National Council on Radiation Protection and Measurements, various government agencies such as the USWB, USGS, and the USPHS, and several state health departments all participated in the hearings. About 3200 pages of hearing record described and discussed all aspects of 1) radioactive waste management operations, 2) research and development on waste treatment and disposal and associated environmental studies in meteorology, geology, hydrology, and oceanography, 3) the future problem, including estimates of the magnitude and economics, 4) federal and state activities in regulating waste disposal, and 5) international activities in this area. Among the salient conclusions reached as a result of the exhaustive JCAE hearings were:

1) Radioactive waste management practices had not resulted in any harmful effects on the public, its environment or its resources.

2) The general problem of radioactive waste need not retard the future development of the nuclear energy industry with full protection of the public health and safety.

EXPANDED PROGRAM (1960-70)

The Congressional Hearings provided a "springboard" for an expanded R&D program during the 1960's. A number of laboratory research projects advanced to an engineering scale and several ground disposal projects reached the field demonstration phase. There were so many exciting things happening in the 1960's, that I will only highlight some of the operational and R&D achievements in both AEC production and research facilities, and the emerging nuclear power industry (1957-70).

Operational

New LLW and Intermediate Level Waste (ILW) Facilities: 1) New alpha waste treatment facility at Los Alamos Scientific Laboratory (LASL), (1962); 2) new LLW treatment plant for ANL (1964); 3) new LLW treatment plant for Oak Ridge National Laboratory (1964); 4) new storage tanks and evaporator facilities for the treatment of ILW from Oak Ridge National Laboratory (ORNL) (1966); 5) new hydraulic fracturing plant at ORNL for disposal of evaporator slurries and ILW sludges, a culmination of the successful R&D program (1966). This technique, which was obtained from the petroleum industry, received extensive development at ORNL during the early 1960's. The method consisted of injecting a waste-cement-clay mixture under high pressure into impermeable geologic formations (like

Conasauga Shale at Oak Ridge) where it solidified in the form of a thin horizontal sheet. With the advice of the NAS, the technique was applied in a different geologic setting during the 1969-70 time period; the project was carried out as a joint effort with ORNL and USGS working with the state of New York at the Nuclear Fuels Services (NFS) Chemical Processing Plant site in West Valley, NY.

Land Burial: The 1960 decade was marked by the entrance of commercial industry into land burial operations, i.e., industrial services were initiated for the collection, packaging, transport, and disposal of low-hazard-potential solid wastes. Two interim land burial sites, one at Oak Ridge and the other at NRTS, were established pending the later designation of permanent regional sites to serve various parts of the country. During the period from 1960-63, the AEC carried out an "interim land burial program" during which time the radioactive waste burial grounds at Oak Ridge and the NRTS were made available. In the three years during which the interim land burial program was in operation, over 7 million cubic feet of solid radioactive wastes were disposed of by land burial.

In August, 1963, the AEC withdrew from its interim low-level waste burial program for licensees, based upon the availability of commercial burial sites at Beatty, NV, and Morehead, KY. In November, 1963, NFS established a LLW disposal facility near West Valley, NY. Other commercial burial grounds were established in 1965 and 1967 at Richland, WA, and Sheffield, IL, respectively. In April, 1971, a sixth commercial land burial ground was established by Chem-Nuclear Services, Inc., in Barnwell County, SC.

Sea Disposal: In the early period of atomic energy development in the U.S. (1946-62), the AEC disposed of small quantities of solid, packaged wastes at designated locations in the Atlantic and Pacific Oceans. With the advent of commercial land burial operations in the early 1960's, a steady decline of sea disposal occurred. Because of the availability of commercial land burial services of low-level radioactive waste materials, the AEC requested all of its contractors in 1963 to evaluate the economics of both disposal methods for their specific operation. With approximately 3/4 of a sea disposal container volume required for concrete ballast to assure its sinking to the 1,000 fathom depth in the ocean, it was determined that land disposal economics would eliminate sea disposal operations.

Incineration: The disposal of low-level combustible solid waste by incineration was practiced at various AEC and power reactor facilities during 1955-70, although not as extensively as at foreign atomic energy installations. As a waste disposal method in the U.S., it had to compete with compression, baling, and shipment of solid waste off-site for land burial. In general, incineration systems used by AEC sites failed to provide the desired performance and were costly to maintain. Extensive developmental efforts were carried out at the Harvard Air Cleaning Laboratory, ANL, KAPL, BuM, and the Army Nuclear Defense Laboratory. Experience with experimental and operating field installations revealed problems in combustion, air cleaning, and corrosion--all which required careful study and control in order to achieve satisfactory operating results. At that time, incinerators were operated at the Rocky Flats and Hanford plants for recovery of plutonium from burnable scrap.

First HLW Solidifier: The Waste Calcining Facility at the NRTS became the world's first plant-scale facility for converting actual high-level

radioactive wastes to a safer, solid form (December, 1963). The plant satisfactorily operated for the rest of the 1960's; it was shut down for about one year while additional solids storage bins were constructed.

Power Reactor Waste Management: While power reactor waste management systems (1957-1970) differed from one plant to another, they included decay hold-up tanks, evaporators, ion exchange units, steam stripping, catalytic recombination of hydrogen and oxygen, fixation of solids and liquids in concrete, incineration, baling, and liquid and gas filtration. In many cases, large waste volumes were processed by storing the material for a period of time to permit decay of radioactivity, and by dilution of the effluents with condenser cooling water prior to discharge from the plant site. Very small amounts of radioactive material which remained were permitted under AEC regulations to be released in liquid and gaseous effluents after careful monitoring.

In order to evaluate the waste management experience of civilian nuclear power plants, Oak Ridge National Laboratory, as an integral part of the AEC's effluent control R&D program, conducted a survey of these practices and operating experience during the period of 1960-67. All plants had operated well within their operating limits, and generally, at a small fraction (less than 5%) of radiation protection standards. It was seen that the management of radioactive wastes produced at nuclear power stations had not been a major factor in influencing the early development and growth of nuclear power.

Research and Development

Some of the more exciting R&D efforts in the 1960's are identified in the following:

LLW Treatment: Programs involving the development, testing, and application of improved systems for the handling, treatment, and disposal of low and intermediate level wastes received extensive efforts during the 1960's. For purposes of this discussion, a low level effluent is arbitrarily defined as one requiring a decontamination factor (DF) in the range from 10 to 103, and an intermediate level effluent as one requiring a DF in the range from 103 to 105. At ORNL and Hanford, extensive laboratory and pilot plant studies were carried out on advanced waste water treatment systems such as scavenging-precipitation ion exchange, foam separation, steam stripping, filtration, two-stage ion exchange, electrodeionization, water recycle, and asphalt solidification. DF's of several thousand were obtained for strontium and cesium in these systems and the average activity concentrations in the various process effluents were only a few per cent of accepted health and safety standards. Parts of this technology were used in the design of commercial power reactor and future fuel reprocessing waste management facilities.

Stream Studies: Supporting stream studies on the Clinch River below Oak Ridge and the Columbia River below Hanford were conducted as an adjunct to the LLW treatment R&D program. During the 1960 decade, the program developed quantitative data and knowledge on dispersal phenomena, reconcentration factors, biological uptake, etc., to assist in the evaluation of waste discharge systems below AEC sites. In its environmental pollution R&D programs, the AEC pioneered the use of a "team approach", e.g. on the Clinch River, the combined engineering and scientific competencies of ORNL, the U.S. Geological Survey, the Tennessee Valley Authority, the Tennessee State Department of Public Health, and the Tennessee State Game and Fish Commission were all integrated into a cooperative river study program. In the 1960's,

additional comprehensive environmental assessments were made of stream conditions in the Savannah River below SRP, and the Mohawk River below KAPL.

Meteorologic Research: In another environmental area, the USWB, now the Environmental Science Services Administration (ESSA), under interagency agreement with DRD, expanded its meteorologic studies on atmospheric transport and diffusion as it applied to the safe disposal of radioactive effluents to the atmosphere. ESSA scientists conducted on-site research in the vicinity of Oak Ridge, NRTS and BNL. Atmospheric transport and diffusion data were used in environmental monitoring of these sites and also in reactor safety assessments. A new volume of "Meteorology and Atomic Energy"--1968 replaced the original publication of 1954, and updated the knowledge and techniques used in the estimation of downwind atmospheric transport and diffusion.

At this point, it is appropriate to summarize briefly the ground disposal research conducted in the 1960's, as recommended by the National Academy of Sciences in 1957. They are: 1) Disposal in salt; 2) Excavation in impermeable formations; and 3) Deep permeable formations (3,000-12,000 feet) containing connate brine.

Disposal in Salt: In response to the NAS suggestion, a major part of the AEC ground disposal R&D program during the 1960's was directed at establishing the suitability of using underground salt formations for the disposal of high-level solidified radioactive waste. Bedded salt formations were chosen as the most optimum disposal media because of their unique geologic characteristics. Salt formations are dry, impervious to water, and not associated with usable groundwater sources. Because of its plasticity, fractures in salt seal or close rapidly. Deposits of rock salt underlie some 400,000 square miles of the United States and may represent some of the few naturally occurring dry environments in the eastern part of the country. Extensive laboratory investigations at ORNL and field studies in the Carey Salt Mine, Hutchinson, KS, were most promising. Project Salt Vault (a demonstration disposal of high-level radioactive solids in a Lyons, KS, bedded salt mine, using Engineering Test Reactor fuel assemblies in lieu of actual solidified waste) successfully demonstrated waste-handling equipment and techniques similar to those required in an actual disposal operation (1965-67). This part of the "salt disposal story" ends at the end of 1969, with ORNL preparing a preliminary design for an operating salt disposal facility, which would handle HLW solids to be generated in the reprocessing of irradiated reactor fuels during the mid-1970's.

Deep Impermeable Formations: Research on the use of deep impermeable formations for the ultimate storage or disposal of HLW was extensively studied at the AEC's Savannah River Plant (SRP) during 1961-63. An intensive exploratory drilling program was conducted to determine the hydraulic and physical characteristics of the basement rock and the overlying aquifer as well as the compatibility of the rock with the waste to be stored. Results of the overall comprehensive study program were reviewed by the NAS Committee on Geologic Aspects of R/A Waste Disposal and outside consultants. Their recommendation was to seek Congressional funding in FY '70 for an exploratory drilling program to provide field verification testing and final engineering design; while the concept appeared technically and environmentally sound, it was canceled after contact between the governor of South Carolina and the U.S. JCAE.

Deep Well Injection: The third geologic concept involving deep well injection was first suggested by the NAS (1957) and the API (1958). The API group studied the problems associated with deep well disposal of HLW to determine whether the techniques used for oil-field brine disposal might be adapted to R/A waste disposal. At this time, the injection of liquid HLW was suggested as an alternative to solidification and transport of the solid waste containers to an off-site underground disposal site. Of course, a suitable deep well (a confined, thick, permeable sandstone in an area of no actual or potential oil production) was conceived as part of the siting criteria for commercial fuel reprocessing plant(s). With the success of the solidification program (in the 1960's), the deep well injection approach was discarded; however, there were some thoughts that the concept might have application to the larger-volume LLW streams which also would emanate from commercial reprocessing plants.

By the end of 1969, more than 20 years of continued tank storage experience had shown it to be a practicable means of interim handling. Over 80 million gallons of radioactive solutions and sludges were stored in nearly 200 underground tanks throughout the AEC complex. Although corrosion data indicated expected tank life-times of 40 to 50 years, there were reported instances of tank failure, all in carbon steel systems at Hanford and Savannah River. Stress-corrosion cracking and/or thermal stress of the reinforced concrete structures were established as causes of these tank leakages, and these factors were taken into account in improved design and construction of new interim tank storage. While tank storage practices were successful in preventing significant quantities of radioactive materials from escaping to the environment, these operations required continual surveillance and tank replacement. This need for surveillance, as well as the necessity of transfer of liquid waste from tank to tank over periods of hundreds of years, were compelling factors for the HLW solidification and deep underground storage programs in the 1960's.

HLW Solidification R&D: With the support of the JCAE in 1959, several processes for conversion of liquid waste to thermally and radiolytically stable solids were investigated. These processes included the use of fluidized beds (ANL & NRTS), a rotary ball kiln at BNL, ceramic sponges at LASL, heated pots at ORNL, and radiant heated spray columns at Pacific Northwest Laboratory (PNL). The addition of glass-forming materials, such as phosphoric acid, lead oxide, sodium tetraborate, etc., for the purpose of providing a relatively non-leachable final product was also intensively studied. The fluidized bed, pot, spray, and phosphate glass processes underwent extensive engineering development during the 1960's. Detailed descriptions of these processes are provided in the references. The Waste Calcining Facility (WCF) at NRTS demonstrated the use of the fluidized bed process during the period of 1963-70 (as noted earlier), principally on waste from the processing of test reactor fuels.

The highlight of the DRD-ESEB HLW solidification R&D program in the 1950's and 60's was the design, construction and operation of a Waste Solidification Engineering Prototype (WSEP) plant at the Hanford Battelle-Northwest Laboratories (BNWL). The engineering-scale technology for solidification of HLW from power reactor fuel reprocessing using the ORNL pot calciner; the BNWL spray calciner-melter; and the BNL phosphate glass system was demonstrated during 1966-72. Thirty-three demonstration runs were completed on the three processes; more than 50 million curies

of radionuclides were processed and collected in 6, 8, and 12-inch diameter containers. Results from WSEP were used to develop design concepts for a 5 tonne/day commercial reprocessing plant and waste management economics for application of the WSEP solidifiers in industrial plants. An overall comparison of the WSEP type solidification processes indicated an apparent slight preference for the spray solidification process over pot calcination, with the phosphate glass process having a fewer number of advantages for the cases studied.

Long Range Evaluation Studies: During the 1960's, the DRD effluent control R&D program utilized ORNL for conducting detailed engineering, economic, and safety evaluation studies to aid the AEC in its planning of future waste-management programs. In these studies, the objective was to evaluate the economics and hazards associated with the treatment of HLW and their storage in ultimate disposal sites as part of a nuclear power economy. These studies indicated that a series of operations consisting of interim liquid storage for five years, solidification, interim solid storage on-site for an additional five years, followed by shipment of the waste and disposal in a salt formation, could be carried out for approximately 0.03-.05 mills/kwh. It was not expected that the costs for waste management, including perpetual storage in a federal repository, would be a significant factor in either the fuel cycle or over-all costs of nuclear power (estimated as less than 1 percent of total power cost).

New HLW Regulatory Policy: As culmination of the R&D program of the 1950's and 1960's, note that in 1969-70, significant policy and regulatory changes occurred in the overall AEC regulatory program. Waste treatment and disposal technology which was developed through the DRD research, development and demonstration programs during the 1960's, was utilized as a basis for the establishment of new waste management policies and revised regulatory procedures. The planning and orderly development of this technology was considered most timely, since we were entering a period in our nation's development which was termed by many as a decade for "environmental action". Protection of the environment had become a major political concern. This resulted in numerous Congressional hearings, including the comprehensive hearings by the Joint Committee on Atomic Energy on the Environmental Effects of Producing Electric Power, which were held in late '69 and early '70.

Prior to these regulatory changes, AEC staff, with assistance from major contractors such as ORNL, PNL, DuPont, Atlantic Richfield, and the Idaho Nuclear Corporation conducted an intensive study in 1968-69 to determine the needs and bases for establishing a siting policy for commercial fuel reprocessing plants and related waste management facilities. General guidance on the handling of liquid HLW inventories at the plant, the conversion of HLW to a solid form, and transfer of these solids to a federally owned and controlled repository were described in the AEC regulatory policy statement of November 14, 1970 (Appendix F). With the continued growth of nuclear power and indications of substantial industrial interest in the construction of fuel reprocessing plants, both contributed to a general feeling of optimism that the fruits of 20 years of R&D would be realized with the operation of a HLW solidification unit and underground repository by the mid-1970's.

FINAL REFLECTIONS

As we conclude our trek down "memory lane", I would like to share some reflections with you. The 1950's and 60's were exciting times in the nuclear energy field, in general, but even more so in the radwaste R&D

area. New technology was being created and demonstrated, and it was most satisfying to be a small part of it. But on the "other side of the coin"--it has been most discouraging to not see the application of fully engineered HLW solidification technology, integrated with the first HLW repository. With the establishment of a HLW repository resisted by political opposition (the NIMBY syndrome), the past 25 years have been a major disappointment. At long last, plant-scale HLW solidification systems at West Valley and SRP are scheduled to go "hot" in early '96. However, the "dream" of seeing a full scale HLW underground repository in operation in Nevada or elsewhere may never be realized in my lifetime.

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DEVELOPING NUCLEAR WASTE POLICY AND REGULATION WITHIN THE CONTEXT OF CHANGING POLITICAL AND SOCIAL ISSUES

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ABSTRACT

Since the end of World War II, the development of nuclear waste management policy and regulation has been influenced by social and political events. Our use of nuclear materials during the 1940's was greatly influenced by World War II and the need to develop weapons. The nuclear establishment was primarily mission oriented - developing weapons to protect us from a perceived threat. Post war political priorities were focused on making the world a safer place; nuclear weapons development was seen as a key element of protection from a perceived future threat. Social priorities emphasized physical and emotional recovery from the war, security, and prosperity. Nuclear waste management, for the past 50 years, has reflected changing social needs and political situations. Post-war protectionism, escalating cold war, rising environmental awareness, economic prosperity and recessions, rapid technology development, increased public awareness and participation in political processes have all affected national waste management policies and practices and our ability to effectively and efficiently manage radioactive waste. In some programs, the result has been implementation of responsible decisions that protect public health and the environment. Other programs are still struggling. In the 1990s, the challenge remains to develop solutions that acknowledge social, economic, and political influences and provide flexibility to accommodate these changing factors.

REGULATORY BEGINNINGS

After World War II, legislation was developed that started our national policy toward atomic energy and eventually the management of nuclear waste. On August 1, 1946, the Atomic Energy Act was signed into law by President Harry S. Truman. This law gave the federal government exclusive authority over the development and application of atomic energy and provided for federal control over regulating the use of radioactive materials. The Atomic Energy Commission (AEC) was established to fulfill this authority. Within this Act, Congress also created the Joint Committee on Atomic Energy to provide Congressional oversight over the operations of the AEC for non-classified aspects of the program.

Even during these early days of operations, certain concerns were raised about waste management practices, especially at Hanford where high level liquid waste was being stored in tanks. Reactor cooling water passed directly into the Columbia River and other liquids were discharged directly into pits (cribs) and allowed to percolate into the soil and eventually into the Columbia River. In 1948, the AEC s Advisory Committee on Nuclear Safety noted these practices would lead to long-term contamination and should be discontinued. The AEC however, indicated that no money was available to explore other disposal methods.

During this early period the AEC adopted health and safety codes and standards. In June 1949, the AEC issued AEC Bulletin GM-133 which instructed its operations to comply with standards developed by the National Committee on Radiation Protection (NCRP) with respect to radiation exposure. The NCRP was actually founded in 1928 as part of the National Bureau of Standards and was originally known as the Advisory Committee on X-ray and Radium Protection. In 1946, the name was changed to the NCRP to reflect its expanded scope work.

In September 1949, the National Bureau of Standards issued Handbook 42 which set forth a maximum whole body dose standard of 300 mrem per week. On an annual basis this is the equivalent of a whole body dose of 15,600 mrem per year.

The 1950's brought an eventual lowering of the permissible doses from radiation, but not until 1958. On June 29, 1951, the National Bureau of Standards issued Handbook 47, which restated the 300 mrem per week radiation dose standard. On March 20, 1953, the National Bureau of Standards issued Handbook 52, which discussed not only permissible amounts of radio isotopes in the body, but in air and water as well. This standard continued the guideline of 300 mrem per week of radiation exposure. In September 1954, National Bureau of Standards Handbook 59 continued the same 300 mrem per week radiation exposure standard. Of significance in Handbook 59 was its discussion of non-occupational doses. Specifically, it required that, with respect to non-occupational exposure of minors, no minor was to receive radiation at a weekly rate higher than one-tenth the respective basic permissible weekly doses for critical organs. However, no changes were made with respect to the standard for non-occupational exposures to adults.

The 300 mrem per week standard was in place until 1958 when the AEC published the AEC Manual Chapter 0524. This manual set forth a new standard for radiation exposure for members of the general public, i.e., non-occupational exposure. This new non-occupational standard was significantly lower than the permitted occupational exposures, and set at 500 mrem per year for whole body exposure and 1,500 mrem per year for specific organ exposures.

In the 1950's two pieces of legislation were significant, the Atomic Energy Act of 1954 and its Amendment in 1957, incorporating the Price Anderson Act. The Atomic Energy Act of 1954 re-authorized the AEC and government control over atomic energy activities. This Act also reestablished authority for protecting the health and safety of the public. At this time, significant interest in commercial generating of electrical power fueled by the peaceful atom was developing. This commercial development required large-scale uranium exploration and mining. In Colorado and other western states, this mined ore was milled and enriched to provide fuel for the new reactors. The potential for accidents and liability for their occurrence became evident. Mining and milling operators, as well as nuclear plant operators pushed forward with legislation to protect their interests. The 1957 Price Anderson Act amendment to the Atomic Energy Act of 1954 provided for liability and compensation in the event of a nuclear accident, should it be transportation or operations related. It was specifically designed to protect the nuclear industry should an accident occur that was so large as to threaten the future of nuclear power.

The AEC was largely self regulating, especially on the classified side, and on a learning curve in waste management. Ocean dumping of low level

waste was common practice, as was trench burial. Usually the low-level waste was categorized according to hazard levels and packaged accordingly before it was placed in the trenches. High level waste was found to be more difficult to contain in some instances, especially during the separation experiments with plutonium at the national laboratories. Several accidents at some of these facilities caused off site releases to the environment. The AEC maintained limited environmental monitoring, primarily of effluents and health testing procedures to maintain compliance with the standards. These monitoring procedures usually took the form of specific AEC Orders.

During the 1960's the Cuban Blockade, Vietnam War and the Civil Rights movements largely distracted government administrations from nuclear activities. Antiwar sentiments were the focus of social issues; weapons testing and reactor building continued with little public opposition or awareness. The need to regulate the building of reactors dominated regulatory development work. Disposal of spent fuel was not yet an issue because the industry planned to reprocess its fuel. However during the late 1960's plans to dispose of high-level wastes from reprocessing emerged with the studies of the Lyons Kansas Project Salt Vault. Other nuclear activities continued under the Swords to Plowshares project, where large potential storage caverns were excavated in salt domes by nuclear detonations.

RISING ENVIRONMENTAL AWARENESS

During the late 1960's the environmental movement gained momentum and signaled the beginning of public involvement in waste management issues. Passage of the National Environmental Policy Act in 1969 required that federal government actions undergo thorough environmental review and that the public participate in the review process. NEPA mandated that all federal agencies consider the impacts to the environment that their actions may incur. The Council on Environmental Quality (CEQ), established to implement NEPA, promulgated regulations that defined environment in its broadest sense to include physical, biological, and human components. It also required a thorough evaluation of alternatives, coordination and consultation with other agencies, and public participation in the review and decision making process.

In the 1970's a number of events strengthened the environmental movement. The 1976 Resource Conservation and Recovery Act addressed the widespread problem of contamination resulting from municipal solid and hazardous waste. This act established a broad policy that governments and industry alike must minimize the amount of waste they produce and find environmentally safer ways to store and dispose of wastes.In 1974, the need for independent regulation of AEC activities led to its reorganization into the Energy Research and Development Agency (ERDA) and the formation of the US Nuclear Regulatory Commission (NRC). The NRC was responsible for developing and enforcing regulations for ensuring the public health and safety from all of the Atomic Energy Act activities. The US Department of Transportation was given the responsibility for developing and implementing safety standards dealing with transportation of nuclear materials. This 1974 reorganization lasted three years, until 1977, when ERDA became known as the US Department of Energy.

Several events in the late 1970's influenced public opinion of nuclear power and subsequent waste management policies and practices. The film China Syndrome and in its wake, the accident at Three Mile Island heightened public awareness and fear of nuclear energy and the potential

consequences of nuclear accidents. Another factor was President Carter's 1978 decision not to pursue development of commercial nuclear fuel technology or permit construction of commercial fast breeder reactors, a component of the existing nuclear fuel reprocessing program. This decision was based primarily on nuclear proliferation fear and fueled public fears of the potential for nuclear war and devastation. Carter's decision also resulted in different and more immediate needs for fuel storage and disposal facilities. Disposal efforts now had to be directed toward spent rather than reprocessed fuel. Power plant operators were forced to take a serious look at storage options for spent fuel. Several states passed legislation banning further development of nuclear power until the disposal problems could be solved.

In terms of low-level waste, states became aware of practices that had resulted in groundwater and soil pollution at several disposal facilities. By 1978, only three commercial low-level waste disposal sites remained in operation. These three sites were to receive all the nations commercial low-level waste resulting not only from nuclear power plants, but also from medical and industrial sources. The governors of these three states gave notice they were going to either close their sites in the near future, or cut down on the quantities of waste that they would be accepting. In this atmosphere of concern, Congress began considering legislation to deal with the radioactive waste disposal problems. With these evident changes, the NRC set about looking at criteria for deep geologic disposal of spent fuel, and land burial of low-level waste, even in the absence of legislation.

CONGRESSIONAL ACTION

In the 1980's, Congress passed several key laws which reflected the need for better management of our nuclear waste streams. The first was the Low-Level Radioactive Waste Policy Act of 1980. Under this law, each state was given the responsibility for ensuring adequate disposal capacity for waste generated within their borders. The act encouraged states to form compact agreements so that joint regional disposal facilities could be developed. States could also receive authority to regulate these facilities from the NRC as Agreement States. States were also given the ability to have a role in the regulation of transportation of nuclear waste. The 1980 Act laid out a time table for developing disposal capacity. This time table proved unworkable and the Act was subsequently amended in 1985 and again in 1987, to give states more time to provide such facilities. However, strict time schedules and penalties also were delineated.

By 1982, nine interstate compacts agreements to site low-level disposal facilities had been ratified by Congress. In December of 1982, the NRC promulgated 10CFR61, Licensing Requirements for Land Disposal of Radioactive Waste, which set the technical requirements for the newly formed compacts' siting efforts. Shortly afterward however, public distrust of land disposal techniques led to recognition that this out of sight, out of mind philosophy was not publicly acceptable. This public opinion steered disposal practices to above ground or above grade facilities, where this waste could be more easily monitored. This attitude led to the revision of 10CFR61 to provide guidance on siting above ground and above grade facilities.

The Nuclear Waste Policy Act of 1982 (NWPA) incited intense public debate, especially in states that appeared promising for hosting disposal facilities, such as Nevada and Washington. Until then, there had been no

legislation governing the search for technically and scientifically suitable sites for the disposal or storage of civilian high-level waste or spent fuel (League of Women Voters, 1993. The NWPA laid out several policies:

A high priority was given to waste disposal. Schedules for repository development and operations were delineated for two repositories, one in the east and one in the west. This provision was carefully negotiated by governors of western states, who felt that it was unfair to put all of the waste in the west when most of the waste was generated in the more industrial east.

Interim storage was to be permitted on a limited basis. Schedules were developed for siting and constructing a monitored retrievable storage (MRS) facility.

Interactions between the states, local governments, Indian tribes were defined. These interactions included such things as financial assistance to states to review activities defined by this act for health and safety and environmental impacts and rights to information. It also provided veto authority to the host state.

Specific federal agency functions were defined; the DOE was to be the repository developer, the US Environmental Protection Agency (EPA) was to set health standards, and the NRC was to implement the EPA standards and promulgate regulations on repository siting, operations and closure.

Rulemaking

In February 1981, the NRC issued their proposed rule, 10CFR60, Disposal of High-Level Radioactive Wastes in Geologic Repositories and final rule in June of 1983, in consultation with the EPA.

In May 1984, the EPA issued their proposed rule, 40CFR191, Environmental Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes. This proposed rule established release limits and dose requirements for deep geologic disposal of these wastes for two principal stages: the period of management and operations and the period after waste disposal had been completed. The EPA finalized this rule in September of 1985. Once this rule was finalized, the NRC had to then issue conforming amendments to 10CFR60, which was done in June of 1986.

10CFR60 and its conforming amendments established procedures and technical criteria for disposal of high-level waste in a geologic repository. These technical criteria were sometimes quite specific (subsystem performance requirements), yet attempted to give the Applicant (DOE) flexibility to develop the repository to meet the overall EPA standard.

In 1984, court challenges brought about a stricter adherence to environmental standards by the DOE when the Legal Environment Assistance Foundation and the Natural Resources Defense Council brought suit against the DOE. The law suit challenged the DOE's right to set and enforce their own regulations. The court found the DOE in violation of environmental standards and ruled that federal environmental laws, and certain state and local laws also apply to weapons production activities. This action led to more public scrutiny of DOE weapons facilities. In 1989, the DOE announced that cleanup and compliance with environmental laws would have a higher priority within the DOE. The office of Environmental Restoration and Waste Management was created to handle this responsibility.

Other laws such as the Clean Air Act and the Clean Water Act heightened public understanding of the contamination problems faced by the nuclear

and hazardous waste industries. These industries became the focus of environmentalist actions. Generally environmental standards became more strict and prescriptive.

NWPA Amendments

Concerned with the escalating costs of finding a repository site for spent fuel and high level wastes, Congress passed the Nuclear Waste Policy Act Amendments Act of 1987. This Act revised some of the policies which were established in the 1982 Act including:

Congress directed the DOE to abandon all sites under consideration except Yucca Mountain in Nevada. The DOE was to characterize this site in order to determine its suitability as a repository and to postpone consideration of a second repository.

Siting, construction and operation of a MRS was authorized and the Office of the Nuclear Waste Negotiator was created to identify, in participation with a state or tribal government, a MRS host.

External technical oversight was increased through the Nuclear Waste Technical Review Board, by onsite oversight by the host states and local governments, and by increased local government participation.

The action of dropping all other sites except Yucca Mountain lead to confrontation with the State of Nevada. Additionally in 1987, the existing EPA standard 40CFR191 was remanded and the EPA was ordered to reconsider the disposal provisions and consistency with other existing laws.

PUBLIC PARTICIPATION

The 1970s and 1980s saw an increase in public involvement in government actions and decisions. Social activism during the Viet Nam War, increased distrust of government as a result of Watergate, and disclosures of health and safety issues associated with hazardous waste disposal practices stimulated a public demand for opportunities to participate in government decision-making. Environmental legislation, led by the NEPA, responded by requiring that federal agencies consider and incorporate public input in the environmental review process. Other environmental legislation also included provisions for public involvement and implementation of community relations programs. It was primarily through the NEPA process that citizens became involved in nuclear waste management issues. Initially, this public involvement took the form of review and comment on environmental documentation, such as the Statutory Environmental Assessments under the Office of Nuclear Waste Isolation Program. Public hearings on the environmental documentation, required under NEPA, were most often contentious and, in fact, the format of such hearings promoted an adversarial relationship. There was little evidence that public comments affected decision-making and the "decide-announce-defend" process most often prevailed. This attitude increased public distrust and led to formation of well organized, well funded and active public interest groups. Federal agencies struggled to respond orienting activities toward public information and education, providing fact sheets and newsletters and holding public informational meeting in local communities. "Educating the public" became the primary focus of these efforts and the unspoken attitude was "then they will understand."

CHANGES IN THE 1990s

The 1990's has brought a new attitude. With remediation costs escalating, an uncertain economy, an increasing federal deficit and political pressure to balance the budget, prioritization was required. The thinking has switched from prescriptive to risk-based standards. Awareness that

current environmental policies may restrict our technological future has emerged with a concerted effort to revamp our environmental and nuclear waste policy laws and subsequent regulation. Likewise, the Yucca Mountain project costs were also escalating. With the existing standard remanded and DOE claiming the strictness of the standards for this predicament, the DOE subsequently requested the National Academy of Sciences to look at the existing standards for waste disposal and to determine if a risk based standard, specific for Yucca Mountain would be desirable. The Academy did question the existing standards in terms of length of time the facility should be regulated and in terms of population dose. These results will now be considered by the EPA to determine if a new standard will be promulgated and if so, will lead to further conforming regulations for the NRC for 10CFR60.

Long term effects of low-level waste are also being examined by the states. This work could lead to changes in the low-level waste regulatory structure, as well. Budget cuts have also affected the role of the NRC in low-level waste regulation. With a need to cut back, one option being considered within the NRC is to turn all regulation of low-level waste over to the states.

The Nuclear Waste Policy Act of 1982 and its 1987 Amendments are being discussed again within the Congress. Several bills have been drafted that could eliminate major state oversight and lead to less outside review of DOE activities. Recent budget problems have left no money for state oversight and led to plans by the DOE to cut the states out entirely. Changes in the development of Yucca Mountain may include less interaction with the NRC and perhaps not even applying for a License to Construct, after all these years of work and expenditures that exceed $6 billion to date.

The 1990s has seen new directions in public involvement, directions that have resulted in meaningful, result-oriented input by citizens. Public scoping meetings and public hearings on NEPA documents have been revamped and now stress public-agency interaction. A meeting facilitator has replaced the administrative court judge in these hearings. Site specific citizen advisory boards have had varying success providing review and oversight of DOE programs. This participation uses the skills and knowledge of citizens and acknowledges the value of stakeholder concerns in establishing priorities that result in technically sound and acceptable decisions. Unfortunately, these directions have not been consistently implemented within the DOE but have focused on the Environmental Management cleanup and remediation programs while waste management programs such as Yucca Mountain have eliminated state funding for technical oversight and public participation remains focused on informational activities.

In general the 1990's has been a period of uncertainty. Much rethinking has gone on with respect to our national waste management programs. While revamping is often necessary and constructive, confusion is not. After 50 years of regulation there is still no long-term strategies for spent fuel and high-level waste disposal. The DOE needs a consistent long-term approach to be efficient, a goal difficult to attain in light of changing political and social priorities. Perhaps the answer is to reach consensus on the framework for decision making rather than determine specific waste management solutions. Models of public participation used in DOE's Environmental Management Programs could provide the basis for developing a strategy for making waste management decisions. If agreement can be

reached on how to make decisions, then specific solutions can be flexible and responsive to site specific conditions and issues and changing social and political climates. A long-term strategy must be developed and implemented to finally result in disposal (or reprocessing) of radioactive wastes. Perhaps the greatest challenge is to consider and incorporate public, political and social issues in this strategy to enhance its implementability and lead to a truly long term solution.

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18-4

EVOLUTION OF SPENT NUCLEAR FUEL REPROCESSING TECHNOLOGY AND HIGH-LEVEL WASTE MANAGEMENT

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ABSTRACT

The volume and physical form of high-level waste resulting from the reprocessing of spent nuclear fuel is determined by several factors, the more important of which are fuel type, reprocessing technology used, and waste treatment following reprocessing. The first plutonium production fuels were natural uranium metal slugs clad with aluminum. The first reprocessing technology was the batch bismuth phosphate precipitation process that recovered only the plutonium from the irradiated fuel; the uranium was sent to waste with the fission products resulting in large waste volumes. Storage in carbon steel tanks required the use of sodium

hydroxide to neutralize the acid waste, further exacerbating the quantity of inert materials present. Technologies were developed to recover uranium as well as plutonium during reprocessing and to recover uranium from the high-level waste tanks; these were the continuous solvent extraction processes known as the Redox and TBP (tri-butyl phosphate) processes. The Redox process did require the addition of inert process chemicals that again contributed to the solids in the high-level waste tanks. The Redox process was soon supplanted by the more desirable Purex continuous solvent extraction process that required the addition of very little inert material. Later defense fuels and research and test reactor fuels used highly enriched uranium in fuels that were composed of uranium aluminum alloy clad with aluminum or uranium zirconium alloy clad with zirconium. Reprocessing techniques were developed for these fuels based on Purex technology. Usually it was not possible to separate the clad and alloying material from the fission products during reprocessing so these materials also contributed to the inert materials in the high-level waste in storage. Except at one site, the use of carbon steel for storing high-level waste continued requiring continued waste neutralization. Technologies have been developed for the recovery of power reactor fuels that can produce very concentrated high-level waste streams, so concentrated that the space required in a deep geologic repository could be determined by the fission product heat output rather than the volume of waste. The benefits of these advanced reprocessing technologies have yet to be realized because of the present status of nuclear power in the U.S.

PURPOSE AND SCOPE

The purpose of this paper is twofold, to examine: 1) the principal determinants of the characteristics of the nation's inventory of high-level waste as it was initially generated and stored and 2) how reprocessing technology initially selected to meet wartime mission needs evolved in response to the needs for improved, more economic and efficient waste management techniques. Concluding remarks include observations on the prospects for reprocessing and waste management in the United States. The evolution of reprocessing technologies is limited in scope to aqueous systems since all of the inventory of high-level waste in the U.S. has been derived from those systems.

INTRODUCTION

In the beginning of the U.S. nuclear program, in the early 1940's, the pressing program requirement was for plutonium in significant quantities. The plutonium was to bede contaminated from fission products, to be available as rapidly as possible with all operations carried out with due regard for the health and safety of the worker and the public. The reprocessing technology used was based more on established techniques rather than optimization with respect to, for example, waste management. In this regard, the program was highly successful; the requisite material was produced on a remarkably short schedule. The focus here will be on the development of those technologies that have been put to large-scale use in the U.S.; that is, aqueous technologies and, in particular, precipitation and solvent extraction processes. It is the use of these processes that has produced all the high-level waste in existence in the U.S.

DEFINITIONS AND OVERVIEW OF REPROCESSING TECHNOLOGIES

High-level Waste (HLW) Definition

It had been recognized from the beginning of the nation's nuclear programs that wastes from the reprocessing of spent nuclear fuels were "high-level wastes," that is, highly radioactive. The definition was made more formal for civilian fuel reprocessing wastes in Appendix F to Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) in 1970 and for all wastes generated from spent fuel reprocessing, defense and civilian, in the Nuclear Waste Policy Act of 1982 (NWPA) (1). Currently high-level waste is source-defined, as in the NWPA, as the highly-radioactive material resulting from the reprocessing of spent nuclear fuel including the liquid waste produced directly and any solid material derived therefrom that contains a combination of transuranic waste and fission products in concentrations requiring permanent isolation. It should be noted that spent nuclear fuel is not defined in the NWPA as high-level waste, but as spent nuclear fuel: nuclear fuel that has been withdrawn from a nuclear reactor following irradiation the constituent elements of which have not been separated by reprocessing.

Reprocessing Definition and Technologies

Most of the nuclear fuels that have been reprocessed in the U.S. are uranium, either low or highly enriched in the isotope U-235. After irradiation, the spent nuclear fuel that was low enriched contained plutonium and residual U-235 whereas spent nuclear fuel that was highly enriched contained a significant quantity of U-235 but very little plutonium. Nuclear fuel reprocessing is the separation of uranium and plutonium from fission products and transuranic (TRU) elements other than plutonium, including the separation of uranium and plutonium from each other. For thorium fuels, of which there are almost none (e.g., Ft. St. Vrain and one loading from Indian Point reactor), reprocessing means the separation of uranium (U-233) from thorium and fission products. Several basic procedures can be used to effect separations; precipitation and solvent extraction have been used at Department of Energy (DOE) sites. In addition, aqueous and non-aqueous technologies have been carried through various stages of development, demonstration, and practical use. Non-aqueous fluoride volatility and pyrometallurgical processes, such as melt refining and salt transport, have been investigated through pilot-scale.

Reprocessing Unit Operations

Reprocessing operations involve several unit operations that can be broadly classified as activities involving 1) the head-end, 2) fissile/fertile/fission product separations, and 3) primary waste treatment. The head-end activities include clad/fuel separations and fuel dissolution resulting in cladding wastes and most of the off-gas wastes (for some fuels, it is necessary to dissolve the clad and fuel together because of the similarity of the clad and fuel; e.g., aluminum-clad uranium-aluminum alloy fuel). Separations activities, precipitation or solvent extraction, which isolate fission products and non-Pu transuranium elements from the useful constituents, were the principal sources of high-level wastes. The primary treatment for storage of liquid high-level waste, as it left the reprocessing plant, was usually neutralization of the acid effluent with sodium hydroxide to allow storage in carbon steel tanks such as those at Hanford and Savannah River; sodium nitrite was also added for corrosion control. Neutralization actually resulted in an alkaline waste in which sludges developed over time. At the beginning of the plutonium production program, there was a severe shortage of stainless steel with which to build large (about one-half million gallons) high-level waste storage

tanks; therefore, carbon steel was used thus precluding storage of acid wastes. High-level wastes are now stored as acid wastes in cooled, stainless steel tanks at the Idaho Chemical Processing Plant (ICPP) because of the possibility of mercury corrosion of stainless steel at high temperatures.

HIGH-LEVEL WASTE IN STORAGE

Determinants of High-level Waste Composition

The characteristics of the high-level liquid waste produced in reprocessing and stored as a liquid in tanks depend primarily on five factors: 1) the fuel type and radiation history, 2) the reprocessing technology employed, 3) the primary waste treatment and storage conditions, 4) subsequent waste treatment and storage conditions, and 5) the cooling time since the end of irradiation and the storage time since reprocessing. Because the composition of fuels and radiation histories are generally fixed, it is through development of reprocessing and waste treatment technologies that the greatest improvement on waste management can be realized. Primary waste treatment refers to the treatment of newly generated HLW before it can be stored, usually as liquid waste in tanks. Subsequent waste treatments to minimize storage risks or to prepare the waste for ultimate disposal will be covered only briefly. The source of all wastes generated throughout the nuclear fuel cycle is shown in R. L. Philippone's paper.

Quantities of DOE High-level Wastes in Storage

Particular emphasis is placed here on the high-level waste generated from the reprocessing of Atomic Energy Commission (AEC), Energy Research and Development Administration (ERDA), and Department of Energy (DOE) defense and test reactor fuels since these wastes represent most of the high-level waste in storage in this country as shown in Table I (2). Most of the high-level waste volume is at the Hanford site, most of the high-level waste curies are at the newer Savannah River site, and most of the curies at the Idaho site have been incorporated in a calcine. In total in the U.S. there are about one hundred million gallons of high-level waste in storage amounting to close to a billion curies of radioactivity. DOE high-level wastes in storage exist in several physical forms because of the principal determinants of the waste; that is, as liquids, slurries, sludges and salt cakes.

Table I

EVOLUTION OF REPROCESSING TECHNOLOGY - CHEMICAL SYSTEMS

Reprocessing facilities have been constructed in the U.S. employing several different technologies as shown in Table II.

Table II

First Separations Process - Bismuth Phosphate

The separations process chosen to meet early program requirements was the Bismuth Phosphate batch precipitation process for the bulk separation of plutonium from everything else followed by steps to decontaminate and concentrate the plutonium further; the uranium was not recovered but sent to waste. The Hanford reactor plutonium-production fuels were aluminum-clad uranium metal slugs. In the head-end, the aluminum was dissolved in sodium hydroxide and the uranium metal in nitric acid. The caustic aluminate cladding removal waste could be used, in part, to neutralize the acid reprocessing waste. This process that began operation at Hanford in December 1944, while it met the objectives set for it and did respect health and safety requirements, did produce very large quantities of high-level waste that contained all the uranium initially present. Later,

recognizing that much of the U.S. supply of uranium was being housed in Hanford's high-level waste tanks, processes were developed to remedy this situation.

First Solvent Extraction Systems - Hanford Redox and TBP

In 1952 the Redox plant was completed and went on line to recover both uranium and plutonium as Hanford defense fuels were reprocessed and the unused wartime U-plant was refitted as the Metal Recovery Plant (also known as the TBP (tri-butyl phosphate) Plant) to recover uranium from the bismuth phosphate tank wastes. The Redox process was the first continuous solvent extraction process and the first process for the recovery of both plutonium and uranium from nuclear fuels employed anyplace in the world. Both the Redox and TBP processes were continuous solvent extraction processes; Redox employed methyl isobutyl ketone ("hexone") as the organic extractant for uranium and plutonium. The metal recovery process utilized tributyl phosphate in a kerosene-like diluent as the uranium extractant. Continuous solvent extraction processes had many advantages over a batch, precipitation process. Disadvantages of the Redox process include an extractant (hexone) that had an undesirably low flash point and was reactive toward high concentrations of nitric acid. The use of hexone required the addition of aluminum nitrate as a "salting agent"; that is, a source of nitrate ions needed to favor the extraction of uranium and plutonium into the organic hexone phase. This inert material followed the fission products into the waste tanks thus increasing the waste volumes that had to be dealt with later.

Subsequent Processes - Purex Process

The TBP process evolved into the mainline Purex U/Pu recovery process that went into operation at Hanford in 1955 and in 1954 and 1955 at the Savannah River Plant. This process had the advantage that nitric acid could be used as a "salting agent" with the potential of substantially reducing HLW volumes. Not all the potential volume reduction available from the Purex process was realized because the continued use of carbon steel tanks required neutralization of the acidic high-level waste from the reprocessing operations, thus introducing inert sodium nitrate into the stored liquid high-level waste from the reaction between sodium hydroxide and nitric acid. Subsequent evaporation and concentration of the liquid high-level waste, left a solid "salt cake" composed largely of sodium nitrate.

Reprocessing Technologies for Specific Spent Defense Fuels

Highly enriched Al-clad, U/Al-alloy fuels as used in driver fuels in defense materials production reactors and in research and test reactors were integrally dissolved, clad plus fuel, in mercury-catalyzed nitric acid; a modified Purex solvent extraction process was used that recovered the uranium but not the plutonium, which went to waste, since there was so little of it in irradiated fuels that were highly enriched in U-235. In the case of zircaloy-clad/zirconium-uranium alloy fuels, the clad and fuel were dissolved together in an ammonium fluoride-ammonium nitrate mixture (zirflex process) or dissolved in boiling hydrofluoric acid (STR process) (3). Before solvent extraction, aluminum nitrate was added to complex with, and to render less corrosive, the fluoride present in solution. At Idaho, some aluminum-bearing wastes were added to the zirconium (and fluoride)-bearing wastes, again to render the fluoride less corrosive. Again, a modified Purex process was used to recover the uranium.

Advanced Purex -Commercial Fuels

Commercial zircaloy-clad slightly enriched uranium oxide fuels have been reprocessed using a shear-leach Purex process. In the head-end of this process, the fuel is mechanically sheared into small 1 to 2-inch segments and the uranium oxide fuel dissolved with nitric acid, leaving the clad as contaminated hulls to be disposed as waste. The uranium solution is reprocessed using Purex modified to minimize the number of waste streams and the addition of inert materials. For example, plutonium oxidation-state adjustment (used to partition uranium and plutonium) can be performed electrolytically without the addition of any chemicals. Plutonium can be further decontaminated from fission products and concentrated using ion exchange. The high-level wastes can be stored in cooled, stainless steel tanks thus avoiding the need for addition of neutralizing agents such as sodium hydroxide. This both increases the safety of waste management and facilitates subsequent waste management activities such as vitrification. This description closely fits the Barnwell Nuclear Fuels Plant in South Carolina, a full-scale plant (5 metric tons of commercial spent fuel per day) that was largely completed in 1977 in South Carolina but never operated because of nuclear weapons proliferation considerations at the time. About 600 tons of power reactor and AEC fuels were reprocessed in the period 1966-1972 in an earlier, smaller commercial solvent extraction plant operated by Nuclear Fuels Services at West Valley, New York, that produced a relatively small quantity of high-level waste. A wet/dry hybrid, Aquafluor, was installed by the General Electric Company in a small plant, the Midwest Fuel Reprocessing Plant, near Morris, Illinois, which never operated because of technical difficulties.

EVOLUTION OF REPROCESSING TECHNOLOGY - MECHANICAL EQUIPMENT

There have been developments in mechanical equipment that parallel the developments in chemical technologies; that is, the development of the actual mechanical devices in which head-end and separations processes are conducted. Mechanical devices did not have as significant an influence on waste management as did the process chemistry. The mechanical separation of cladding from the fuel by shearing and leaching eliminated the clad from the high-level waste. The success of a particular chemical technology did depend on the availability of appropriate mechanical equipment.

Decladding and Dissolution

Initially, fuel was declad chemically and dissolved in the same batch dissolvers from which oxides of nitrogen and gaseous fission products were emitted, principally krypton and iodine and sometimes ruthenium as the tetroxide. With some fuels, as mentioned earlier, the clad and fuel alloy were batch-dissolved simultaneously. There have been many dissolver designs proposed and tested. One of the more interesting is a continuous, rotary countercurrent dissolver has been shown for commercial-type fuels to provide a high degree of control of dissolution and digestion rates and off-gas evolution. It has also been shown that the oxides of nitrogen can be converted back to the original nitrate form (also called fumeless dissolution) and that all the volatile fission products can be trapped. Separations The bismuth phosphate precipitation process used centrifuges for solid/liquid phase separation but that process was supplanted by liquid-liquid solvent extraction processes before any major equipment improvements were needed. Solvent extraction processes involve the transfer of chemical species between immiscible aqueous and organic phases. Such transfer is enhanced by a large interfacial area obtained by

intimately mixing the two phases. In order to effect countercurrent stage-wise operation of solvent extraction equipment, not only must there be mixing, there must be phase separation. In some aqueous-organic systems it is difficult to mix the phases but easy to separate them (e.g., water-hexone); in others it is easy to disperse the phases but difficult to separate them (e.g., water-tributyl phosphate/normal hydrocarbon). Phase mixing and separation depend on many things, the more important of which are interfacial tension characteristics and density differences. Solvent extraction equipment that has been used successfully includes packed columns, pulse columns, mixer-settlers, and centrifugal contactors. The first application of the Redox process employed packed columns some 50 feet in height. Fifty feet of column height means fifty feet of shielding height and a high capital cost for the facility. The pulse column, while shorter than a packed column, still requires considerable cell height, though with proper piping, the headroom required can be decreased by dividing the column into two or three concatenated sections located side by side. Mixer-settlers, in which the two phases are mixed in one chamber and separated by gravity in another, use much less headroom than columns. One of the more remarkable pieces of solvent extraction equipment ever devised is a form of mixer-settler called the centrifugal contactor that uses much less space. It provides for excellent mixing and outstanding phase separation, taking place as it does in a centrifugal field. It is stagewise very efficient, has a high throughput in compact equipment with a short residence time, and can be shut down or started up in a very short time. The compact equipment minimizes shielded cell space and, coupled with short residence time, minimizes in-plant process solution inventory that in turn facilitates materials safeguards. A packed column is a relatively simple device whereas pulse columns, mixer settlers, and centrifugal contactors are progressively more mechanically complex.

HIGH-LEVEL WASTE STORAGE, TREATMENT AND DISPOSAL

High-level Waste Volumes

Comparison of the relative volumes of high-level waste generated by the processes just discussed is enlightening as shown in Table III (4)(5). There was a dramatic decrease in waste volume in going from the bismuth phosphate process (that resulted in a uranium-bearing waste stream) to the Redox and Purex solvent extraction processes (that provided for the recovery of uranium). Because the continued use of carbon steel tanks (except at the Idaho site) required neutralization, inert materials (particularly sodium nitrate) continued to be added to the high-level waste, as noted earlier. The volumes estimated to be produced from commercial power reactor fuels using advanced Purex technologies are so small that extensive cooling would be required to prevent unwanted boiling and the wastes may be more concentrated than can be handled comfortably during vitrification and subsequent storage and disposal.

Table III

Further Treatment of High-level Waste

Over the years, the high-level waste at Hanford has been treated for various reasons. For example, to acquire more tank space, a process was developed at Hanford in the 1950s to scavenge radiocesium from tank waste liquids. In implementing this process, approximately 140 metric tons of ferrocyanide were added to 24 single shell tanks (6). The presence of ferrocyanide must be considered in managing these particular wastes. In the future, it is planned to treat the high-level waste at Hanford and

Savannah River to separate the high-level waste into two fractions; one largely free of inert materials but containing most of the radioactivity and the other, produced incidental to concentrating the high-level waste, containing most of the inert materials (e.g., sodium, aluminum, zirconium nitrates) but little of the radioactivity. The larger volume of incidental or low-activity material would be immobilized and disposed on-site and the smaller volume of high activity material would be converted to glass, placed in stainless steel canisters, and stored on-site pending removal for disposal in a deep geologic repository. It is estimated that at Savannah River, for example, by the year 2020, there will be over 300 megacuries of radioactivity in a little over 3,000 cubic meters of glass and about 60 kilocuries of radioactivity in about a million and a half cubic meters of saltstone, a cementitious material (7).

High-level Waste Disposal - Repository Criteria

Up to this point, the discussion has focused on the evolution of chemical technologies that have been devised to minimize volumes of waste in storage. But repository space requirements for disposal are determined by both volume and heat output. The Waste Acceptance System Requirements Document (WASRD) (8) published by the Office of Civilian Radioactive Waste Management (OCRWM) specifies that high-level waste destined for disposal in a repository be borosilicate glass sealed in right circular cylinders of austenitic stainless steel of height 3 meters, diameter 61 centimeters (interior volume 887 liters), a maximum of 2,500 kilograms mass, and a maximum heat output of 1,500 watts. For Savannah River, it is estimated that, in the year 2015, the heat output from high-level vitrified waste in canisters would average a little over 100 watts per canister (7), well below the maximum allowed. It is clearly an advantage to minimize the volume of the high-level waste glass as will be done at Savannah River and other DOE sites.

The situation would be substantially different if commercial power reactor fuels were to be reprocessed. The average heat output from all irradiated fuels in storage in year 2000 from U.S. commercial power reactors is estimated to be about 3,000 watts per metric ton of spent fuel (7). If this "average" fuel were to be reprocessed using advanced Purex technology followed by vitrification of the high-level waste, the heat output would exceed the 1,500 watts per canister limit by several fold, meaning that the final waste form would be heat, not volume, limited. In this case, the advantage of the Purex technology would be to allow adjustment to the composition of the high-level liquid waste to minimize the waste volume to the extent practicable and to facilitate durable glass formation on vitrification.

CONCLUSIONS

From beginnings in 1943, through expansions of the 1950's, to the decisions of 1992 to phase out reprocessing at DOE sites, there has been an ever growing concern for an increasing number of regulatory requirements and for the proper management of nuclear wastes including ultimate disposal. Part of this waste management challenge has been met with the development of reprocessing technologies more compatible with safe and economic storage, treatment and disposal of nuclear wastes. In the event that commercial reprocessing of power reactor fuels should ever again take place in the U.S., advanced technologies could be employed that would be optimized, with respect to health and safety, environmental concerns, accountability and safeguards, and economics for all aspects of fuel reprocessing and waste management. However, for the foreseeable

future, the leadership that the U.S. clearly once held in reprocessing technology is essentially lost. Beginning with the Atoms-for-Peace program that was announced by President Eisenhower in December 1953, other countries learned much in commercial applications of nuclear technology from the U.S. With time, other countries had much to offer to the U.S. and technology exchange agreements were initiated to further the transfer of information in both directions. In the past few years, all reprocessing technology development in the U.S. has been abolished along with the relevant international exchange agreements. Not only is our technological base being eroded, we have lost the ability to remain knowledgeable of, or have significant influence on, developments abroad. Should nuclear fuel reprocessing be needed in the U.S. in the future, and no one can say it will not be, the U.S. can always turn to France, the UK, or Japan (or all three) for a turnkey reprocessing plant and trained crews to operate it. Nor should we be surprised if, unbeknownst to us, some unlikely country (or countries) turns up with a nuclear fuel reprocessing capability.

No reprocessing of commercial fuels is now taking place in the U.S. but Purex-related systems are in current use elsewhere in the world as is well covered in R. L. Philippone's paper. The spent fuel from nuclear power reactors in storage in the U.S. (about 29,000 metric tons of uranium as of the end of CY 1994) that contains a considerable amount of potential [nuclear] energy (e.g., 250 metric tons of plutonium and 240 metric tons of U-235), is now destined for deep geologic disposal. This uranium, if available for reenrichment, would be equivalent to about 37,000 metric tons of natural uranium that, on enrichment, would produce about 5,400 metric tons of power reactor fuel enriched to 3.2% U-235 that would fuel the 99,000 electric megawatts of U.S. nuclear power for about two years. If licensed for such use, commercial reactors could also use the plutonium in mixed oxide fuels.

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ACKNOWLEDGEMENTS
The author wishes to acknowledge the help of the following who contributed significantly to this paper: R.D.Walton,Jr., ANL; H.F. Walter, DOE/EM; D.E. Bailey, CADMUS, M.E. McVey, Consultant, S. Schrock, ORNL.

18-5

THE CIVILIAN HIGH LEVEL RADIOACTIVE WASTE PROGRAM

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ABSTRACT
After years of false starts and neglect, Congress in 1982 finally prepared all encompassing legislation for the storage, transportation and disposal of high level radioactive waste and spent nuclear fuel produced by the civilian sector. The House Science and Technology Committee had the Congressional lead for development of the Nuclear Waste Policy Act of 1982 (Act). As part of the legislation the Department of Energy's Office of Civilian Radioactive Waste Management (OCRWM) was created. Over the years, since 1982, there have been numerous articles, testimony, statements at hearings, State and Indian Nation meetings and other discussions which have been critical of the Program. A particular concern has been the date for the Department to take title of the spent fuel for disposal purposes. The 1998 timeline has become a vanishing dream.
Four organizational entities can be primarily identified as contributing, in varying degrees, to the present state of the Program. These entities are: Congress; the Department of Energy (DOE); the Utility Industry and the media. Each of these entities who were influential on the course of the OCRWM Program are addressed in this paper; however, it must be understood that each interacted with the other and frequently influenced the course of action in often unexpected ways. Further, there were many other players in the process of influencing the Department's decision-making, including the affected States, Indian Nations, regulatory bodies, and other.
Once legislation had been signed into law by President Reagan, the Department of Energy (the Department) initiated the provisions of the Act, building upon work that had already been done. The initial step, rapidly implemented, was the reduction of nine candidate sites, previously identified, to five sites (all west of the Mississippi). The next step was the further reduction of the five sites to three sites. These final three sites were to be characterized in detail, thus permitting the Department to proceed with selection of the repository

site. The decision of the repository site would be supported by an Environmental Impact Statement and a recommendation to the President on the selected site. There would also be an opportunity for Congress to reject the site, if so petitioned by the State in which the site would be located.

If all other provisions of the Act were met, a license application would be prepared for submission to the Nuclear Regulatory Commission, for approval to proceed to construction. Selection of the three sites was achieved by using a multi-attribute utility method to rank the five sites and provide the decision-makers with another vehicle to support the selection. Concurrent with the first repository activity, but off-set by one to two years, was initiation of a second process for identification of a second repository to be located east of the Mississippi. The entire repository Program as mandated by the Act was to achieve equity in the selection of sites. As noted later in this paper, the identification of three sites seemed to be the trigger for the process mandated by the Act to start to come apart.

Of all the forces acting upon the Department, none has been of greater influence, for obvious reasons, than the Congress of the United States. For about 10 years, prior to 1982, Congress has tried to pass some all-encompassing legislation to provide legislative direction and legal coverage for permanent disposal of the high level waste and spent nuclear fuel accumulating at an ever increasing rate within the DOE complex and at the utility pools. In the waning days of 1982, Congress was finally successful and passed the Nuclear Waste Policy Act of 1982 (Act). However, from that point on, Congress continued to provide close management and direction of the Program, with politics playing a key role in the Department's decision making. As examples of Congress providing basic direction, albeit indirect through the Department's leadership, two events are significant. First, in the early stages of the Program, Congress made it clear that there were some sacred cows (sites), such as the salt domes along the Gulf coast, and language was added to the Act to preclude selection of certain of these sites. Language was also added that effectively prevented the start of shaft sinking at Hanford and delayed site assessment until the site was legislatively terminated. Probably the clearest signal of congressional management of the Program was the indefinite delay, announced by and attributed to the Department, of further investigations associated with the selection of a second site in an eastern state as required by the Act. Finally, in 1987, Congress passed legislation, the 1987 Amendments to the At, which profoundly changed the course set by the original Act. Basically, the Amendments took away the fundamental basis of the Act, which was equity, and left the Department with only one site to characterize and a second repository relegated to an uncertain future. There was no backup for an unsuitable site; all the effort was to be focused on an unproved site in a State that was, and remains, adamantly opposed to the site, and has taken numerous steps, within its own regulatory framework, and in the courts, to prevent and/or delay the siting process.

During this period from 1983 to the present, the Department of Energy has had only 3 permanent Directors. The first was appointed and confirmed in 1984, a year after the law was enacted and left the program in 1987. It wasn't until 1990 that the second Director was in place, and the third came into office following the 1994 Presidential elections. These extended periods with acting Directors in place were characterized by

status quo, since it was frequently difficult for the Acting Director to take significant new initiatives. The result could be described as stagnation in program direction and inability to deal decisively with issues inside DOE and with outside entities, i.e., Stakeholders (the states, Indian Nations, the utilities and Congress). The most significant progress (both positive and negative) during these periods followed the appointment of the permanent Directors, when significant initiatives were undertaken to move the Program forward. But again, the Department was frequently stymied by the inability to gain consistent funding, coupled with specific direction written into the reports (often the personal bias of Congressional staffers) which accompanied the funding legislation. Further, the ability of the Department to move forward was frequently thwarted by the Stakeholders who exerted considerable influence on the Department's decision-making process. A prime example of this influence was associated with the second repository effort. The Act provided for a second repository site to be identified in the East, thus providing equity with the selection of a site in the West. The second repository siting process was characterized by strict groundrules which severely limited the siting managers from gathering specific information on potential sites. The process was further complicated by the strong negative and well orchestrated reaction of the local Stakeholders to siting a repository in their backyard (NIMBY). Following some particularly contentious public meetings where the DOE representatives had to be protected from very hostile audiences, the Department abruptly announced that the second repository was not needed until some time into the next century. Interestingly, it was not long afterwards that Congress passed the Amendments Act which eliminated the siting process for the second site until well into the next century.

The Utilities, through their industry representatives, became increasingly critical of the Department and its progress. Over the course of several years, it became obvious that the Department would not be able to meet a 1998 date for taking title to the fuel, as the utilities believed was mandated by the Act. The Department, on the other hand, did not believe they had the obligation, pending the siting of either a repository or a monitored retrievable storage facility. In the utilities' view, the Act's requirement that large amounts of the utility money be transferred from the nuclear utilities to the Waste Fund in order to support the Program was not producing the expected results. Instead, the utilities saw the funds going into a perceived sink hole without any visible progress toward meeting the 1998 date. As a result, the utilities became increasingly strident in their criticism of the Department and lobbied Congress continually, seeking additional changes to the Act and hope for relief regarding payment of ratepayer funding in the Waste Fund. Currently, the utilities are supporting legislation that will basically transfer the burden of paying for storage from the utilities to the Government. This will, again, fundamentally change the approach initially established by the 1982 Act.

Finally, it is worthwhile to review the media's role. Of all the parties, the media would describe its role as merely a bystander reporting the facts. Unfortunately, this was not the case in the vast majority of newspaper and journal articles and broadcasting. Typical of the attitude and tone set by the media is the reported comment of a nationally syndicated columnist who states, when her facts were challenged, that the staff just did not understand. As she stated, "...This is theatre...there

is no story in the mundane reporting of things going well." These comments set the stage for what has been the tone whether in the written or broadcast format. Generally, the Department, notwithstanding these periodic negative shots across the bow has handled the situation with reserve and dignity. Nevertheless, the result of all the negative publicity is to set a view in the public's mind that there is no safe way to dispose of the "waste."

What has been the results of this experience over the last 16 years? Basically, the Program has become increasingly more institutionalized and moribund in its ability to move ahead. The present OCRWM management seems to have set a plan into motion that appears to have the attributes for success, basically, determining in the shortest possible time whether or not Yucca Mountain is a suitable site. But weighed against this plan are large budget cuts which impact the progress of the program and the myriad of bills that Congress has in the hopper to once again make changes which only the future can tell whether they will be successful.

Session 19 -- WIPP-1

Co-chairs: Brad Wolfe, Foster Wheeler Environmental Corp.

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19-2

UPDATE ON FINAL WIPP COMPLIANCE CRITERIA (40 CFR PART 194)

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ABSTRACT

The purpose of the meeting presentation is for the Environmental Protection Agency (EPA) to provide an up-to-date status of the Waste Isolation Pilot Plant (WIPP) Compliance Criteria rulemaking (40 CFR part 194). As of January 1996 (the time of this summary), the final rule has cleared the Office of Management and Budget (OMB) review process pursuant to Executive Order 12866. It is not appropriate for the Agency to discuss the final contents of 40 CFR 194 until the rule is signed by the EPA Administrator; however, this summary will discuss, in general terms, the status of the rulemaking. It is expected that the final rule will be signed prior to the meeting, so detailed information can be provided during the oral presentation.

INTRODUCTION

Since the Federal Register notice containing the Final Rule should be published in February 1996, the EPA should be in a position to discuss and articulate the explicit requirements in the final compliance criteria during the meeting. The EPA will also report on the last year's events that occurred in finalizing the criteria. These events will include: meeting of the National Advisory Council for Environmental Policy and Technology (NACEPT); lawsuits filed against the Agency regarding OMB review, timeliness of rule promulgation and the compliance application guidance; proposed legislation limiting the EPA's role in certifying compliance with the disposal regulations; and finally, last steps taken to finalize the rule.

BACKGROUND

The EPA regulates the release of radionuclides from the management, storage and disposal of radioactive waste in order to protect public health and the environment. The WIPP, which is under development by the Department of Energy (DOE), is a potential geologic disposal system for defense transuranic (TRU) radioactive waste. Pursuant to the 1992 WIPP Land Withdrawal Act, the EPA is required to perform several activities including, but not limited to: 1) finalizing safety standards for radioactive waste disposal; 2) issuing criteria for determining whether the WIPP complies with the radioactive waste disposal standards; 3) certifying whether the WIPP complies with the standards before waste disposal can begin; and, if the EPA certifies that the WIPP complies; 4) determining (or "re-certifying") whether the WIPP continues to be in compliance with the disposal standards every five years after initial receipt of waste at the site for disposal. In December 1993, the EPA completed the first of these activities by issuing the final radioactive waste disposal standards (40 CFR part 191), which place limits on the releases of radionuclides from waste management, storage and disposal facilities. Then, in January 1995, the EPA issued proposed compliance criteria (40 CFR part 194), intended to implement the 40 CFR part 191 disposal standards specifically at the WIPP.

In February 1996, after extensive consideration of public input, the Agency expects to issue the final compliance criteria. This program is unique because the EPA's approach taken at the WIPP may set precedent for future approaches taken at other radioactive disposal facilities. Many of the nuclear waste issues that are of considerable concern to the public will be addressed in this rulemaking.

The EPA's final compliance criteria are specific to the WIPP and its compliance with the disposal regulations found in subparts B and C of 40 CFR part 191. The primary goal of the compliance criteria is to make compliance at the WIPP as straightforward as possible. The criteria are aimed at clarifying the requirements of the radioactive waste disposal regulations and explaining procedural aspects involving the EPA's certification and re-certification of compliance at the WIPP. The final criteria are intended to support a "reasonable expectation" of compliance and include:

General Requirements, which include provisions for EPA inspections, quality assurance, computer models and codes, waste characterization, future states assumptions, expert judgment, and peer review.

Provisions addressing the Containment Requirements of 40 CFR part 191. These requirements address methodologies for considering inadvertent human intrusion and conducting probabilistic performance assessment.

Provisions addressing the Assurance Requirements of 40 CFR part 191. The Assurance Requirements are qualitative in nature and complement the numerical containment requirements. They are intended to provide added confidence in the performance of the disposal system or "defense in depth."

Provisions addressing the Individual and Ground-Water Protection Requirements of 40 CFR part 191. These requirements apply to undisturbed performance and limit exposures over 10,000 years, with the specific requirement of no more than 15 millirem per year per individual. The final criteria address methodologies for conducting compliance assessments against these requirements, appropriate assumptions, uncertainty considerations, and information needed to judge the results of such considerations.

Public Participation Requirements, which describe public opportunities for providing input to the certification rulemaking process.

APPROACH

The Agency's approach to WIPP's compliance demonstration will be presented during the oral presentation. For example, the EPA will describe its approach to assumptions regarding future human activities at the WIPP site (perhaps the most critical scenario in a WIPP performance assessment) that will have to be considered in demonstrating the disposal system's performance and its ability to contain the waste for 10,000 years. In addition, the EPA will discuss its consideration of the level of proof necessary to acquire confidence in the WIPP's ability to comply with the disposal standards. Additional confidence can be achieved through the successful implementation of the assurance requirements (e.g., institutional controls, monitoring, engineered barriers) of 40 CFR 191.

Throughout the rulemaking process, the EPA has been interacting with stakeholders on all aspects of the compliance criteria. The EPA is committed to achieving logical, reasonable compliance criteria that are protective of the public and the environment. To this end, the EPA has worked, and will continue to work, to gather input and promote understanding among key stakeholders. For example, the EPA: 1) circulated and obtained comments on an early, pre-proposal draft of the criteria; 2) participated in numerous technical meetings with the DOE; 3) hosted a technical workshop on several important compliance-related issues; 4) solicited comment on the proposed criteria and held three public hearings in the State of New Mexico; and 5) held a meeting of the NACEPT WIPP Review Committee to discuss critical compliance issues. Issues that have been complex and of intense concern include human intrusion, "credit" for passive institutional controls, monitoring, engineered barriers, peer review, waste characterization, and the qualification of "old" data. After promulgation of the final compliance criteria, the EPA will be responsible for ensuring that the criteria are properly addressed in the DOE's compliance certification application. And, upon review of the application, EPA will make a determination as to whether or not the WIPP complies with the Agency's radioactive waste disposal regulations. If so, the EPA will certify compliance and the facility will be allowed to open and accept waste for disposal.

19-3

THE WASTE ISOLATION PILOT PLANT REGULATORY COMPLIANCE PROGRAM*

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ABSTRACT

The passage of the WIPP Land Withdrawal Act of 1992 (LWA) (1) marked a turning point for the Waste Isolation Pilot Plant (WIPP) program. It established a Congressional mandate to open the WIPP in as short a time as possible, thereby initiating the process of addressing this nation's transuranic (TRU) waste problem. In addition, the LWA established the

Environmental Protection Agency (EPA) as the regulator for the WIPP and provided a schedular framework in which the EPA is required to work. Finally, the Congress provided for the oversight of certain WIPP activities by numerous other federal agencies, the National Academy of Sciences (NAS), the state of New Mexico, and the Environmental Evaluation Group (EEG).

The DOE responded to the LWA by shifting the priority at the WIPP from scientific investigations to regulatory compliance and the completion of prerequisites for the initiation of operations. Regulatory compliance activities have taken four main focuses: 1) preparing regulatory submittals; 2) aggressive schedules; 3) regulator interface; and 4) public interactions.

Four major compliance submittals are being prepared including a supplement to the Environmental Impact Statement (SEIS), a hazardous waste permit application, a no-migration variance petition (NMVP), and a compliance certification application (CCA). The WIPP Disposal Decision Plan (DDP) (2) was issued in May 1994 to show the DOE's aggressive commitment to opening the WIPP in 1998. Regulatory agencies that are dealt with on a day-to-day basis include the Environmental Protection Agency (EPA) and the New Mexico Environment Department (NMED). More than a dozen stakeholder meetings have been conducted to discuss various aspects of the WIPP program and to obtain input from the public. The WIPP stakeholder list includes 2650 individuals and organizations. The DOE has been highly successful in its strategy, meeting all milestones and holding frequent technical exchanges with the regulators. The DOE anticipates a continuation of this success as it continues toward an April 1998 opening of the WIPP.

INTRODUCTION

The DOE made a significant decision when it decided to abandon further on-site testing at the WIPP in favor of laboratory testing. This decision marked the transition from a facility for the collection of scientific information to an operating facility with the mission of disposing nearly 176,000 cubic meters of TRU waste. This transition was not easy. Experimental staffs had to be reduced as regulatory compliance staffs increased. Discussions with scientific organizations had to be augmented by discussions with regulatory organizations. These challenges notwithstanding, numerous factors contributed to the DOE's decision to proceed to operations. Among them were the LWA (1), the increasing emphasis on environmental remediation at DOE's weapons sites, and the general recognition that enough on-site testing had been done and further tests could be performed in laboratories.

THE TRANSITION TO DEMONSTRATING COMPLIANCE

The WIPP Land Withdrawal Act of 1992

When the LWA was passed in the waning days of the 102nd Congress, it received overwhelming approval. This had dual significance. First, it indicated that nuclear waste disposal is still a national priority and second, that Congress is anxious to see significant progress toward that end. While the LWA contained provisions for continued testing in support of the WIPP, that testing was required to be focused and relevant to demonstrating compliance with regulations applicable to the WIPP. One of the key features of the LWA is the schedules that Congress mandated. Key among these is Congress' intent that WIPP begin disposal operations within 7 years of initiating the on-site Test Phase. Even though the on-site Test Phase has been abandoned, the intent is clear--get on with

disposal. Recent attempts in Congress in both the House and Senate to amend the LWA have, as a common element, the establishment of a date certain for WIPP's opening--clearly reiterating Congress' intent to open WIPP.

Another important aspect of the LWA was the establishment of the EPA as the final approval authority for compliance with the long-term performance regulations in 40 CFR 191 Subparts B and C. Congress did this by requiring the EPA to establish criteria for the "certification" of the DOE's compliance with 40 CFR Part 191. This action has initiated activity by both the DOE and the EPA to ensure that the EPA has sufficient information to perform its duties.

Finally, the LWA requires numerous rulemaking processes including input from the public.

Regulatory Framework Documents

The first step the DOE took was the preparation and issuance of several documents to define the framework within which WIPP compliance will be demonstrated. These included the Regulatory Criteria Document (RCD) (3) which establishes general policy for compliance for TRU waste repositories and makes key interpretations of those portions of the environmental regulations that apply to the unique nature of the WIPP as a mined geologic repository for TRU and TRU mixed waste. In addition, the RCD provides a common basis for the implementation of similar requirements in the various environmental standards. The RCD specifically addresses the radiation standards in 40 CFR Part 191 (4), the hazardous waste standards in 40 CFR Part 264 (5), and the Land Disposal Restrictions in 40 CFR Part 268 (6).

The second regulatory framework document prepared by the DOE is the WIPP Regulatory Compliance Strategy and Management Plan for Demonstrating Compliance to Long-Term Disposal Standards (RCSMP) (7). The RCSMP detailed the WIPP specific activities needed to reach compliance and discussed the interrelationships among the various elements of the compliance and the experimental programs. This strategy called for preparation of numerous documents and compliance submittals that now appear on the WIPP DDP.

In addition, the Format and Content Guide for Title 40 CFR 191 and Title 40 CFR 268.6 Compliance Reports (FCG) (8) provides the framework for the documents that will be submitted to regulatory agencies. The DOE prepared the FCG as a means of assuring proper content and organization of the topics needed for a demonstration of compliance. Preparation was made with limited guidance from the EPA. The FCG has been useful in the preparation of the submittals that have been finished to date. The flexibility of the FCG has allowed the DOE to tailor the submittals to the specific regulations and regulators.

WIPP Disposal Decision Plan

The regulatory framework documents became the basis for preparing the WIPP DDP (2) which is shown in Fig. 1. The regulatory compliance schedule became the driver for the integration of other project activities such as experiments, waste characterization, public outreach programs and transportation system and facility readiness. The WIPP DDP is the tool for integrating these activities and for tracking progress. The DDP was most recently revised in October 1995 (Revision 2).

Fig. 1
Establishment of the Carlsbad Area Office

In order to provide proper focus on the compliance programs and to provide a single contact with regulatory agencies and with stakeholders, the DOE established an area office in Carlsbad in early December 1993. The Carlsbad Area Office (CAO) has total responsibility for the WIPP Program and for the implementation of the National TRU Waste Program. This combination provides complete interaction and coordination among the various WIPP project elements (DOE, Sandia National Laboratories, Westinghouse, Generator Sites) and the public, including the City of Carlsbad, a staunch supporter of the WIPP Project. The improved communication and coordination inherent with a centralized administrative function is paramount to the successful progress of the WIPP in light of its aggressive schedule.

Systems Prioritization Method

The Systems Prioritization Method (SPM) (9) was developed as a decision-aiding tool to aid in the process of identifying and selecting those experimental activities that will facilitate compliance. The SPM integrated standard decision-making tools with the performance assessment tool developed by Sandia National Laboratories to evaluate experimental activities based on their expected outcomes. The result was the specification of eight activities that could be completed within the desired time frame and which provide high confidence that compliance will be demonstrated.

ELEMENTS OF THE REGULATORY COMPLIANCE PROGRAM

The DOE regulatory compliance program comprises four major elements. These elements have been defined by asking three fundamental questions. First, what has to be done, by whom, and when? Second, How are they to be prepared? Third, who is the audience? Each of these questions is discussed in the subsequent text.

What? Whom? When? Schedule and Resources

The first task for the DOE was to identify all of the submittals that were needed for compliance and the times by which the submittals must be in the hands of regulators to assure timely processing. Five required submittals were identified. Given the unique nature of the documents and the time frames over which final information would be available, the DOE decided that the best way to handle these five submittals was through the preparation of seven separate documents as shown on the WIPP DDP (Fig. 1). In addition, the DOE identified two other support documents that would facilitate the compliance process. A summary of these documents is presented below.

The DOE addressed the resource issues by forming a compliance integration committee made up of managers from the CAO, Westinghouse, Sandia National Laboratories, and the DOE's technical support contractor. This committee addressed schedule and resource issues and conflicts, and interfaced with DOE upper management to assure timely availability of needed resources. In addition, the committee dealt with compliance issues and provided guidance on addressing issues in compliance documents. Since the inception of the accelerated compliance program, funding of compliance activities has received priority.

How? The Application Process

One of the greatest challenges facing the DOE with regard to regulatory permitting and certifying at the WIPP facility is the lack of regulatory precedent. For example, there are no other permitted mined geologic repositories. There are no other permitted TRU waste repositories. There are no other disposal facilities seeking a variance from the land

disposal restrictions. There are no permitted hazardous waste disposal facilities in the state of New Mexico. The EPA has never certified another federal agency's compliance with applicable regulatory requirements. This lack of precedence impedes the regulatory compliance program because both the WIPP and the regulators have to weigh each decision in light of the letter of the regulation, and where the letter of the regulation is unclear, in light of the spirit of the regulation. Unfortunately, in some instances, regulations conflict, or are inconsistent with each other.

In order to overcome some of the problems associated with the lack of precedent or with the conflict among regulations, the CAO has adopted an aggressive approach to compliance. One of the keys to the DOE's progress is the preparation of draft documents for review by regulators and the public. This allows for open discussion of issues in the context of the DOE's overall compliance strategy. While the number of comments becomes burdensome at times, the DOE treats each comment as a valid input into the application process.

With regard to application format, the DOE has used the FCG as the standard for applications. Since the FCG has roots in the NMVP process, it has proven to be acceptable to the EPA's Office of Solid Waste (OSW). Likewise, the EPA's Office of Radiation and Indoor Air (ORIA) considers the FCG an adequate framework for preparing the CCA. However, the actual content will have to conform to EPA's certification criteria to be promulgated as 40 CFR Part 194.

Whom?

The answer to this question is two fold--the regulators and the public. Both entities have a vested interest in the process and the outcome.

Regulator Interface

Interfacing with the regulators has been a major priority for the CAO. Three primary regulators involved are: the EPA ORIA, the EPA OSW, and the NMED, each with differing levels of familiarity with WIPP. Most of the CAO's efforts have, to date, focused on ORIA, since it has not been previously exposed to the WIPP. On the average, bimonthly technical interchange meetings have occurred and have covered major topics such as geology, hydrology, numerical model development, scenario screening, waste characterization, and others.

The CAO has tried to focus on current issues and their resolution. However, historical issues are also of interest to the regulators and, consequently, these have also been discussed. For example, deep dissolution was debated for many years at WIPP. Dissolution-related features in the vicinity were investigated to determine their likelihood of occurring at the WIPP site. Features such as the Wink sink, which is generally accepted to be human induced dissolution associated with oil and gas production, were also studied. None were determined to be likely at WIPP due to the lack of fresh water aquifers in the vicinity. Nonetheless, with new regulatory agencies involved, and new personnel, these old questions resurfaced and must be addressed.

Stakeholder Interface

Interface with stakeholders is a key aspect of the aggressive regulatory compliance strategy. Stakeholder input is mandated by federal and state laws and implementing procedures. The CAO concluded that the sooner the process began, the better for achieving the schedule and for minimizing "surprises" during public comment periods. The CAO defines WIPP stakeholders as those persons or organizations who have a vested interest

in the outcome of the permitting process. This includes the local population, many state and national organizations, individuals along transportation routes, populations at generator sites, and other interested parties. The WIPP stakeholder list includes 2650 individuals and organizations.

CAO stakeholder outreach has taken four forms. First, the DOE convenes general stakeholder outreach sessions to provide summaries of WIPP progress and to obtain general input. Second, the DOE has stakeholder meetings on specific topics to provide opportunities to inform and discuss specific technical issues of concern. Third, the DOE has formed stakeholder focus groups to obtain input on narrow issues. Fourth, stakeholders are asked to comment on numerous DOE documents prior to final issuance. In addition, stakeholders are kept informed about technical exchange meetings with the regulators and some stakeholders regularly participate in these meetings.

Of course, the success of the stakeholder outreach program lies in the CAO's ability to incorporate stakeholder views and comments into the WIPP program. While the CAO is still working toward a formal mechanism to do this, it is being done informally. During the SPM stakeholder meetings, numerous suggestions by stakeholders with regard to process, documentation, and content were provided and addressed by Sandia National Laboratories (9).

The CAO is committed to continue the stakeholder process. There are still plenty of topics to discuss, not the least of which will be the final conceptual models and the final data inputs used for compliance calculations.

ACCOMPLISHMENTS

The CAO has been able to maintain the aggressive schedule that it committed to in 1994. The following is a summary of the documents that have been published to date and the nature of the reviews that have been conducted.

Compliance Status Report

The Compliance Status Report (CSR) (10) was completed in March 1994. It provides the status of the compliance program. The CSR is broad in scope and covers both the radioactive waste and hazardous waste components. It identifies over 40 issue areas needing additional research or the preparation of final documentation. It served the purpose of providing the framework for starting discussions with regulators and stakeholders. The CSR generated over 280 written comments. In addition, the DOE and the EPA have had several technical exchanges to address these issues.

Project Technical Baseline

The Project Technical Baseline (PTB) (11) was conceived and prepared to be a compendium of established information regarding the WIPP site, with emphasis on information needed for environmental compliance documentation. It was issued early in the compliance program, as a draft in April 1994, to assure a level of consistency in documents being prepared for compliance. The latest revision is April 1995, although several changes are being processed at this time in preparation for the final NMVP and the CCA. The PTB is held under configuration control which means that it can only be changed with DOE's approval and after all users have had an opportunity to review the changes. It will be updated quarterly while applications are in process and probably less frequently after all regulatory submittals are complete. Documents that use the

information in the PTB include the NMVP, the RCRA Permit Application, the Draft CCA and the CCA, the Safety Analysis Report (SAR), and the SEIS.

Biennial Environmental Compliance Report

The first Biennial Environmental Compliance Report (BECR) (12) was issued October 1994 to satisfy the LWA requirement for a biennial assessment of compliance. It addresses 24 separate federal laws and state of New Mexico counterparts. The BECR is currently being reviewed by the EPA.

Draft Compliance Certification Application

The Draft CCA (13) was issued to ORIA in March 1995 in nine volumes. It describes the 40 CFR Part 191 compliance program status as of March 1995 for requirements that model undisturbed performance. The Draft CCA was supplemented in July 1995 (14) for requirements that model disturbed performance. The Draft CCA uses the projected outcome of the experimental activities identified in the SPM. The EPA has provided general comments and more detailed comments on the Draft CCA are expected in early 1996. The Draft CCA is based on 40 CFR Part 191 and does not incorporate the proposed 40 CFR Part 194 criteria since it was not available in final form at the time the Draft CCA was prepared.

Phase I No-migration Variance Petition

The Phase I NMVP (15) was submitted to OSW in May 1995 in seven volumes. It principally addresses emission issues. The EPA provided public notice of the Phase I NMVP, however, it received no public comments.

Revision 5 of the RCRA Permit Application

The DOE submitted Revision 5 of the RCRA Permit Application to the NMED on May 1995 (16) in ten volumes. The application has been re-scoped for disposal phase operations and closure. Discussions with NMED are ongoing. A notice of deficiency covering any remaining technical issues is expected in February 1996. The notable issues that the DOE and the NMED are discussing are waste characterization, RH-TRU waste, and closure.

FUTURE ACTIVITIES

In order to finally obtain permits and approvals, several key future activities are anticipated. These will constitute final submittals, and, to some extent are awaiting either final data on experiments or final rules from the EPA or both.

Phase II No-migration Variance Petition

The Phase II NMVP is due to the EPA in June 1996 after Sandia National Laboratories finalizes the conceptual models, the shaft seal design, and the numerical codes. It will contain the long-term performance calculations required to demonstrate no-migration for 10,000 years. The same calculational methods that were developed for the radioactive components of the waste will be used, along with the same conceptual model of the disposal system. A final decision by the EPA is expected by June 1997.

Hazardous Waste Permit

A draft hazardous waste permit is expected in the spring of 1996. The public will be provided an opportunity to comment on the NMED's permitting proposal. It is likely that public meetings and hearings will be scheduled. A final permit decision is expected before the end of 1996.

Supplemental Environmental Impact Statement

In the Supplemental Environment Impact Statement (SEIS) prepared for the Test Phase, the DOE committed to issue another SEIS to support the decision to begin waste disposal operations. Current plans call for the draft SEIS to be issued in May 1996. After a public comment period including public hearings in June 1996, the final SEIS is expected in

January 1997. A Record of Decision (ROD) is scheduled for issuance in March 1997. Among the issues being looked at are transportation, operational life, waste characterization, treatment to meet the waste acceptance criteria, backfill, and RH-TRU waste.

Final Compliance Certification Application

The DOE will prepare the CCA for submittal to the EPA in October 1996. However, the DOE needs the final data and conceptual models to prepare the CCA. These are due to be completed by March 1996. The CCA cannot be prepared prior to the issuance of final certification criteria, 40 CFR Part 194, by the EPA in February 1996. The proposed criteria were issued in January 1995 (17). The DOE has commented extensively on these proposed criteria (18, 19). In addition, the EPA is preparing a Compliance Application Guidance document to provide information relative to the level of detail expected in the CCA (20). An EPA certification is anticipated by October 1997.

SUMMARY

The DOE has successfully made the transition to demonstrating regulatory compliance at the WIPP. This transition is supported by an aggressive schedule that involves the DOE, its contractors, regulators, oversight organizations, and the public. The accelerated compliance process has been marked by numerous successes, including the resolution of issues and the identification of major topics to be addressed between now and the final submittal of applications. Major issues that are currently discussed include peer review, waste characterization, the use of engineered barriers, quality of old data, and computer code documentation. While many of these issues represent a significant amount of work, none appear to be obstacles to completing the compliance tasks in a timely manner.

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THE WIPP RCRA PART B PERMIT APPLICATION FOR TRU MIXED WASTE DISPOSAL*

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ABSTRACT

In August 1993, the New Mexico Environment Department (NMED) issued a draft permit for the Waste Isolation Pilot Plant (WIPP) to begin experiments with transuranic (TRU) mixed waste. Subsequently, the Department of Energy (DOE) decided to cancel the on-site test program, opting instead for laboratory testing. The Secretary of the NMED withdrew the draft permit in 1994, ordering the State's Hazardous and Radioactive Waste Bureau to work with the DOE on submittal of a revised permit application. Revision 5 of the WIPP's Resource Conservation and Recovery Act (RCRA) Part B Permit Application was submitted to the NMED in May 1995, focusing on disposal of 175,600 m3 of TRU mixed waste over a 25 year span plus ten years for closure.

A key portion of the application, the Waste Analysis Plan, shifted from requirements to characterize a relatively small volume of TRU mixed waste for on-site experiments, to describing a complete program that would apply to all DOE TRU waste generating facilities and meet the appropriate RCRA regulations (Title 40, Code of Federal Regulations, Parts 260-270). Waste characterization will be conducted on a waste stream basis, fitting into three broad categories: 1)homogeneous solids, 2)soil/gravel, and 3)debris wastes. Techniques used include radiography, visually examining waste from opened containers, radioassay, headspace gas sampling, physical sampling and analysis of homogeneous wastes, and review of documented acceptable knowledge. Acceptable knowledge of the original organics and metals used, and the operations that generated these waste streams is sufficient in most cases to determine if the waste has toxicity characteristics, hazardous constituents, polychlorinated biphenyls (PCBs), or RCRA regulated metals.

The repository is composed of eight panels and two panel equivalents (access drifts), with each being designated as an independent Hazardous Waste Management Unit (HWMU). Revision 5 of the Permit Application introduces partial closure of each underground HWMU when filled, and final closure of the entire repository following decontamination of the Waste Handling Building (WHB). Panel closure is accomplished by a multipart engineered panel closure system to be installed in the entries of each HWMU, keeping the emitted Volatile Organic Compound (VOC) concentrations well below environmental performance standards.

The NMED may issue a Notice of Deficiency (NOD) upon completion of the technical review and interactive discussions held with the DOE. The NMED will publish a draft permit and schedule a comment period and public hearings, following the DOE's response to the NOD.

INTRODUCTION

The WIPP site has been set aside from public use to host a deep geologic repository for TRU mixed waste. The WIPP has both surface and underground facilities constructed and ready to begin disposal operations. The underground repository has been partially mined 650 m beneath the surface within a thick bedded salt formation. The repository is eligible for permitting as a miscellaneous unit under state and federal regulations (RCRA). A permit is required under the New Mexico Hazardous Waste Act and its implementing regulations for hazardous waste management operations.

In September 1994, the NMED withdrew the draft hazardous waste permit at the WIPP facility (issued in August 1993) after the DOE elected to cancel the in situ tests that were covered by that draft permit, requesting the DOE to submit a revised permit application.

The revised Part B Permit Application is organized in accordance with the RCRA application checklist (1). The NMED issued numerous comments as

their technical review proceeded. Most of the comments were requests for more detailed information. The DOE responded to each comment and drafted new text and supporting information where appropriate. These revisions were discussed during meetings with the NMED prior to their issuing the NOD letter.

TRU mixed waste to be received at the WIPP exists in a variety of physical forms, ranging from unprocessed laboratory trash (e.g., tools, paper, glassware, gloves) to solidified wastewater treatment sludges resulting from plutonium reprocessing and fabrication as well as from research and development activities at various DOE defense program facilities. Most TRU mixed waste is contact-handled (CH) but some with a higher intensity of radioactivity is remote-handled (RH). In the future, significant quantities of TRU mixed waste may be generated from environmental restoration, decontamination, or decommissioning activities. TRU mixed waste containers are required to be vented through high-efficiency particulate air (HEPA)-grade filters to preclude pressurization caused by potential gas generation.

PERMIT APPLICATION

The permit application is organized into 13 chapters, with headings and subheadings in accordance with the RCRA application checklist (1).

Chapter A, "RCRA Part A Application Certification"

Chapter A consists of Part A of the application. The Part A contains the facility's owner and operator information, type of facility, capacity, process information, estimated annual waste receipts, and certification. The DOE is the WIPP facility owner/operator, and, in accordance with DOE policy, the Management and Operating Contractor, Westinghouse Waste Isolation Division, has signed the certification as co-operator.

Chapter B, "Facility Description"

The facility description includes some history of the site selection and the development leading up to construction. The basic requirements of the WIPP Land Withdrawal Act of 1992 are presented. Eight underground panels of seven rooms and two access drifts each comprise the repository and are to be permitted as HWMUs. The disposal area access drifts will probably also be used in the future as panels 9 and 10. As each panel is filled, it will be closed off to isolate that waste from the rest of the facility.

The WHB areas used for waste receipt, handling, inspection, etc., are included for permitting as container storage units (HWMUs), consisting of the CH Bay, the RH Bay and hot cell complex in the WHB, and the parking area south of the WHB. Waste is received in TRU Package Transporters (TRUPACT-II) and road or railroad casks (designed to meet Department of Transportation Type B requirements), which are unloaded inside the WHB. Both road and rail transportation accesses are available.

The CH waste containers will be inspected in the CH bay as they are removed from the TRUPACT-II, loaded onto a facility pallet, moved to the waste hoist conveyance, and lowered down the shaft to the repository horizon via the waste hoist conveyance, and transported to the HWMU where the waste is disposed. The road cask with RH waste is unloaded in the RH bay into the hot-cell complex remotely, where the waste container can be identified, checked for external contamination, and loaded into the facility cask for transport underground via the waste hoist conveyance. The facility cask is transported to the HWMU and set on the emplacement machine, which faces a drilled hole in the salt. Shield valves allow the waste container to be pushed into the hole without exposing the operators

to radiation. A shield plug goes into the hole afterward to assure safe personnel access to the room for CH waste stacking activities after the emplacement machine has been removed.

Other information in Chapter B is provided on location, topography, distance from bodies of water, prohibition of hydrocarbon wells on the site, vicinity of water wells, lack of nearby population centers, cattle grazing leases, site access control, and compliance with other portions of the regulations, including the floodplain and seismic standards. The only pathway for hazardous emissions is by air, either from the underground or the WHB exhaust.

Chapter C, "Waste Analysis Plan"

Generators are required to characterize all wastes planned for disposal at WIPP, using formal programs adhering to the requirements of the Waste Analysis Plan (WAP), the WIPP Waste Acceptance Criteria (WAC), and the TRU Waste Characterization Quality Assurance Program Plan (QAPP). Both sampling and analysis programs and acceptable (process) knowledge will be used in classifying the wastes. Some wastes have been in retrievable storage since 1970 and others will be generated in the future.

The previous permit application for test waste required characterizing 100 percent of the waste, although waste quantities were small and only two DOE generators were involved. The revision for disposal applies to all DOE TRU waste generator sites, possessing the full spectrum of waste streams. Waste characterization requirements for disposal have been developed to balance the quantity of data required to satisfy the regulator, yet prevent extensive increases in generator personnel exposure and excessive expense. Thus, the WAP establishes new WIPP TRU waste characterization requirements for all DOE generator sites planning to dispose of their TRU waste.

Waste characterization will be done on a waste stream basis. The waste streams fit into three broad categories: 1) homogeneous solids, 2) soil/gravel, and 3) debris wastes. Techniques to be used include radiography, visual examination of opened containers, headspace gas sampling, physical sampling and analysis of a statistically determined amount of homogeneous wastes, and review of documented acceptable knowledge.

Radiography is used to noninvasively examine waste containers (required for all stored waste and judiciously applied to newly generated waste) for physical form of the waste and to detect free liquids and other prohibited items.

Visual examination, an invasive technique, will be conducted on a statistical sample of waste containers from each waste stream and used to inspect waste contents and verify radiography results.

Headspace gas sampling and analysis are used to determine the concentrations of VOCs, hydrogen, and methane within the headspace of waste containers.

A statistically selected portion of homogeneous solids and soil/gravel wastes will be sampled for hazardous waste constituents and toxicity characteristic. Samples will be analyzed for VOCs, semi-VOCs, and metals.

Acceptable knowledge is used to classify debris waste. Knowledge of the original organics and metals used and the operations that generated these waste streams is sufficient to determine if the waste has a toxicity characteristic, hazardous constituents, PCBs, or RCRA-regulated metals. RCRA-regulated metals present in debris waste are associated with specific waste materials (e.g., lead in leaded rubber gloves).

Radiography, visual examination, and headspace gas sampling are used to collect data to provide confidence in acceptable knowledge.

Generator waste characterization also includes quality assurance (QA) requirements, applied through meeting the QAPP requirements by implementing site-specific QA Project Plans (QAPjP), which are submitted to the WIPP for review and approval. The WIPP personnel will perform audits of the generator site waste characterization programs to verify that implementation of the QAPjPs is consistent with the requirements of the QAPP and the WAP.

There are three phases to waste shipment screening and verification. The first step involves submittal of a Waste Stream Profile Form to the WIPP for review. The next phase occurs prior to shipment wherein the electronically transmitted data package is reviewed and accepted by WIPP personnel after extensive edit and range checks conducted by the WIPP Waste Information System (database). This acceptance is based on successful completion of the WIPP audits of the generator's QAPjP activities, and a determination that the data meet the acceptance criteria. The third phase occurs after the shipment has arrived, but before it is accepted. This includes 1) a determination of the completeness and accuracy of the EPA Hazardous Waste Manifest, 2) a determination of waste shipment completeness, 3) a determination of land disposal restriction notice completeness, and 4) an identification and resolution of any waste shipment irregularities.

Chapter D, "Facility and Process Information"

Facility and Process Information includes the TRU waste management facilities, equipment, and operations described in Chapter B, and compliance with the environmental performance standards for the WIPP.

The physical attributes of the WIPP site contribute to the ability of the facility to isolate TRU mixed waste and ensure that human health and the environment are protected. The average annual precipitation is only 30 cm, 96 percent of which is returned to the air through evapotranspiration. The geologic sequence at this site mainly consists of three evaporite-bearing formations, the Castile (deepest), Salado, and Rustler. The repository is located in the Salado formation, an approximately 610 m thick bedded halite, with some carbonates, anhydrites, and clay seams. It is regionally extensive, has extremely low permeability, behaves in a plastic manner under pressure, contains only fluids that are saturated with salt, and lies between the other two formations which contain highly impermeable layers that offer further confinement for the waste.

Four shafts connect the underground area with the surface. The Waste Shaft is located within the WHB. The Air Intake Shaft and the Salt Handling Shaft provide ventilation underground. The Exhaust Shaft serves as a common exhaust air duct for all underground areas.

The WHB was designed to withstand a design basis tornado with a maximum windspeed of 294 km/hr and a design basis earthquake acceleration of 0.1 g.

Release of hazardous waste or hazardous constituents to the air that may have adverse effects on human health or the environment is unlikely. VOCs constitute the greatest hazard, but the maximum concentration of any VOC release has been calculated to be at least two orders of magnitude below the environmental performance standards.

Chapter E, "Groundwater Monitoring"

In the past, groundwater monitoring at the WIPP has focused on the Culebra member of the Rustler Formation above the repository, as it represents the most permeable and most likely hydrologic contaminant migration pathway. No credible pathway has been established for contaminant transport to aquifers below the repository horizon, as there is no hydrologic communication between the two. It is the DOE's position that the migration of hazardous waste or hazardous constituents from the WIPP repository to the environment through groundwater is unlikely. The bedded-salt formation acts as an extremely low-permeability regional barrier isolating the repository from water-bearing units. Also the plasticity and geologic behavior of halite will tend to close any potential pathways created by excavation.

For hazardous waste or hazardous constituents to migrate from the repository to groundwater-bearing units, there must first be a pathway, such as a shaft. All four shafts extending to the repository horizon have been constructed to minimize the infiltration of water from the overlying water-bearing units into the repository during its operational life. After waste emplacement, the shafts will be filled with permanent low-permeability seals and plugs designed to inhibit migration of fluids to and from the repository.

Surface water is generally absent in the immediate vicinity of the WIPP facility.

Chapter F, "Procedures to Prevent Hazards"

Security, facility inspections, structures, equipment, procedures, and other measures taken to prevent hazards during the disposal operations are described.

Security requirements are met by 24-hour surveillance and a barrier to control entry to the facility at all times.

Equipment instrumental in preventing, detecting, or responding to environmental or human health hazards is inspected periodically for malfunctions, deterioration, potential for operator errors, and discharges which could lead to a release of hazardous waste constituents to the environment, or pose a threat to human health.

The WIPP facility has a variety of communications systems and emergency response equipment and possesses a continuous water supply to meet emergency situations. The intraplant communication systems include two-way communication by the public address (PA) system and its intercom phones and paging channels, an intraplant telephone system, mine phones, pagers and plectrons, portable two-way radios, and local and facility-wide alarms. External communications are provided by the commercial telephone system and two-way radios for summoning emergency assistance from off site or communicating with outside agencies.

The water supply system is for domestic use and fire control for the maximum credible fire. The underground has no water supply, but has fire extinguishers of various types and a rescue truck with chemical and foam extinguishers.

Diesel generators provide power in the event of utility power loss and the uninterruptible power supply is always available to supply important monitoring systems. Thus, during a power outage, the ventilation systems are powered by the diesel generators and all waste handling operations are shut down into a mode providing personnel and facility safety.

Chapter G, "RCRA Contingency Plan"

The Contingency Plan defines responsibilities, provides guidance for coordination of activities, and minimizes hazards to human health and the

environment from fires, explosions, or any sudden or nonsudden release of hazardous waste or hazardous waste constituents to air, soil, or surface water. In case of an emergency where a release results that may threaten human health or the environment, the RCRA Emergency Coordinator will activate the Contingency Plan. The provisions of the Contingency Plan apply to all HWMUs, the Waste Shaft, and supporting TRU mixed waste handling areas. Nonradioactive hazardous substances, hazardous materials, and hazardous wastes will also be managed in accordance with the Contingency Plan.

A RCRA Emergency Coordinator will be on site at the WIPP 24 hours a day, seven days a week, with the responsibility for coordinating emergency response measures. Persons qualified to act as the RCRA Emergency Coordinator are thoroughly familiar with the Contingency Plan, the TRU mixed waste and hazardous waste operations and activities at the facility, the locations of TRU mixed waste and hazardous waste activities, the locations on site where hazardous materials are stored and used, and the locations of waste staging and accumulation areas. Other personnel identified are 1) the Central Monitoring Room Operator, 2) the Emergency Response Team, 3) the Emergency Services Technicians, 4) the First Line Initial Response Team, 5) the Mine Rescue Team, 6) the Office Wardens, 7) the Chief and Assistant Chief Office Wardens, and 8) the Security Fire Support. The Contingency Plan will be activated only when the RCRA Emergency Coordinator determines that the activation criteria are met.

If notification of local authorities and or regulatory agencies is determined to be necessary, the RCRA Emergency Coordinator will assure that those notifications are made. The DOE policy is to provide accurate and timely information to the public by the most expeditious means possible concerning emergency situations at the WIPP site that may affect off-site personnel, public health and safety, and/or the environment. Supplementing the on-site capabilities for emergency response, the DOE has Memoranda of Understanding with off-site emergency response agencies for fire response, medical assistance, and law enforcement. Since the WIPP facility is owned and operated by the DOE, other off-site DOE emergency response organizations could be called in for support.

Any incident requiring activation of the Contingency Plan will be reported to the operating record in detail. Notifications required by regulations will be made to the required regulatory agencies within the allotted time span.

Chapter H, "Personnel Training"

The WIPP's personnel training program meets the requirements of the RCRA. Preparing personnel to operate the WIPP facility in a safe and environmentally sound manner is the primary objective of the training program. All on-site personnel are provided with an introduction to RCRA and emergency preparedness within 30 days of employment. Other relevant training is provided to employees and their supervisors whose jobs are such that their actions or failure to act could result in a spill or release or the immediate threat of a spill or release of hazardous waste. Required RCRA-related training is conducted by certified instructors and consists of classroom instruction and on-the-job training as appropriate. Certain positions require formal certification and are identified by Department Managers based on safety, complexity, and involvement with hazardous waste handling operations.

Hazardous waste management courses are offered at a frequency that ensures that new hires or transfers can receive relevant training within six months of assuming their new position. Employees do not work unsupervised in hazardous waste management positions until they have completed the requisite training.

The training program for emergency response ensures that personnel are able to respond appropriately and effectively to emergency situations by providing thorough training, including fire response elements, 40-hour miner training, Mine Safety and Health Administration requirements for medical and mine rescue, and lifesaving elements.

Chapter I, "Closure Plans, Post-Closure Plans and Financial Requirements"

This chapter was extensively revised to contain descriptions of the activities necessary to close the WIPP facility at the end of it's life and provide surveillance after closure. The previous revision detailed "clean closure" based on the possibility of having to retrieve and return all of the test waste. The WIPP is a federal project, so financial assurances for closure are not required to be addressed. The underground HWMUs (panels) will be closed individually as they are filled, with barriers constructed to limit accessibility of the hazardous constituents to the environment during continuing operations. Partial closure of each HWMU will be considered complete when the panel closure system is emplaced and operational, and the NMED has approved the closure. Possible backfilling of access drifts, plugging and sealing of the shafts, and dismantling of surface facilities will constitute final closure.

For the purposes of establishing a schedule for closure, the operating life may last 25 years, followed by a closure period of ten years for decontamination (both radiological and chemical), decommissioning, and final closure. The WIPP surface and subsurface facilities are planned to be dismantled (except for the hot cell portion of the WHB, which will remain as a portion of the Permanent Marker System) and either salvaged or disposed of. Paving and caliche will be removed and the surface area recontoured and revegetated. During the closure period, the DOE will continue to demonstrate compliance with applicable permit requirements. Facility monitoring procedures in place during operations will remain in place through final closure.

Decisions about closure activities may be based in part on analyses of potentially contaminated surfaces and media. Reliability of analytical data will be accomplished by following a Quality Assurance/Quality Control (QA/QC) program that mandates precision and accuracy of laboratory analyses. The documented QA/QC program currently in place at the WIPP meets the DOE QA requirements.

Final closure of the WIPP facility will prevent the intrusion of fluids into the repository, prevent human intrusion after closure, and minimize future physical and environmental surveillance. Access to the site of the repository's surface footprint will be controlled by the federal government as long as practicable and at least for the 100-year period considered under 40 CFR Part 191. This active control will preclude inadvertent intrusion into the disposed waste by deep drilling or mining natural resources.

The post-closure care period begins after completion of final closure of the facility and continues as long as necessary to satisfy regulatory requirements and to protect human health and the environment. During the post-closure period, the WIPP site will be maintained in a manner that complies with the applicable environmental performance standards. No

post-closure monitoring for detection of releases is proposed since the migration of contaminants is unlikely. However, because there are other regulations that apply to the WIPP long-term performance, several techniques need to be evaluated (e.g., geophysical techniques and subsidence monitoring).

Chapter J, "Corrective Action for Solid Waste Management Units"

The solid waste management units (SWMU) within the 41.4 km2 WIPP site boundary were identified by the EPA Region 6 as requiring further investigation. Corrective actions are required only for SWMUs from which releases of RCRA hazardous wastes or hazardous constituents have occurred.

The definition of a SWMU has not been finalized by the EPA yet. WIPP uses the definition presented in the proposed Subpart S of 40 CFR Part 264. This definition states that SWMUs are "any discernible unit at which solid wastes have been placed at any time irrespective of whether the unit was intended for the management of solid or hazardous waste. Such units also include any area at or around a facility at which solid wastes have been routinely and systematically released." 16 SWMUs requiring further investigation have been identified by the NMED, which conducted a RCRA Facility Assessment for the EPA. Seven others were created by the DOE since the filing of the Revision 3 of the Permit Application for the test program. Units that are similar in physical characteristics or waste type are grouped within an SWMU description for the particular type of discernible unit.

The DOE will institute corrective actions necessary to protect human health and the environment for any release of hazardous waste or hazardous constituents from SWMUs at the WIPP facility. Authority for regulating corrective actions recently passed to the NMED.

Chapter K, "Other Federal Laws"

Other federal laws, Executive Orders (EO), and regulations which deal with environmental protection, or were evaluated for applicability to the operation of the WIPP facility, are listed. These were all assessed because of the nature of the WIPP's proposed activities and the proposed land use.

Chapter L, "No-Migration Variance Petition"

The history of the DOE's submittals of a No-Migration Variance Petition (NMVP) is summarized, as is the subsequent addendum to the EPA to demonstrate that the site characteristics, operational practices, and waste characteristics were sufficient to prevent migration of hazardous constituents beyond the unit boundary during the period planned for the test program.

The EPA issued a Conditional No-Migration Determination for the Test Phase. Subsequently, however, the DOE abandoned plans for testing waste at the WIPP. The DOE submitted a new Draft NMVP for the waste disposal in May 1995. A final NMVP for both operations and post-closure will be submitted in June 1996.

Chapter M, "Certification"

This is a statement of the DOE being the owner and operator of the WIPP facility with the Westinghouse Waste Isolation Division as the co-operator. Managers of both organizations have signed this certification.

RECENT ACTIVITIES

The NMED determined the permit application to be administratively complete and during their subsequent technical review, hosted interactive discussions with the DOE and contractors for both organizations. The

major comments received from the NMED were for much more detail to be supplied.

The DOE set a goal of receiving the Permit in August 1996. The DOE attempted to resolve the NMED's comments with expanded descriptions of all aspects of the facility design and operation, even those associated with the radiological controls. The DOE also attempted to reduce the quantity of comments in the NMED's Notice of Deficiency (NOD) letter via the interactive comment resolution process. All activities under DOE control continue to be aimed at meeting the set goal.

The DOE submitted a final response to the NMED's comments in mid-January 1996, following the early January technical exchange meeting with the NMED. The NMED told the DOE that, based upon the contents of the final response, they would review the responses and revised text, and issue the NOD in mid-February, which they did. The DOE was given 30 days to respond to the NOD, after which the NMED would develop the draft permit.

Upon issuance of the draft permit, a public comment period will be established. Public hearings are anticipated at various cities in New Mexico, in which the DOE will participate. The NMED will consider the testimony from the hearings and all comments received, and craft a final permit (or a denial) with all conditions they believe are necessary to protect the public safety and the environment from harm by the waste disposal operations at the WIPP facility.

CONCLUSION

The WIPP disposal-phase operating permit application was submitted to the NMED by the DOE in May 1995, and is described in some detail. Following a period of technical review, the NMED issued a number of comments, to which the DOE responded with draft revisions to the permit application. After several technical exchange meetings, the NMED issued an NOD in February 1996. A new revision to the application will be completed, along with written responses to each deficiency, and submitted to the NMED within the prescribed response time.

The next step is for the NMED to draft the permit, supplying their conditions for facility operations, and hold public hearings. Upon consideration of the oral and written responses during and after the hearings, the NMED is expected to issue the permit.

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THE WASTE ISOLATION PILOT PLANT NATIONAL ENVIRONMENTAL POLICY ACT DISPOSAL DECISION*

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ABSTRACT

To strengthen the effectiveness of the Department of Energy's (DOE) environmental clean-up efforts, a number of individuals have petitioned

for the accelerated opening of the Waste Isolation Pilot Plant (WIPP). The longstanding mission of WIPP is to demonstrate the safe disposal of transuranic (TRU) waste resulting from United States defense activities. In 1980, to comply with the National Environmental Policy Act (NEPA), the DOE completed its first environmental impact statement addressing TRU waste disposal options which compared the impacts of several alternatives for TRU waste disposal. Based on the 1980 analyses, the DOE decided to proceed with the construction of the WIPP facility in 1981. In a 1990 decision based on the examination of alternatives in a 1990 Supplemental Environmental Impact Statement, the DOE decided to continue development of the WIPP by proceeding with a testing program to further examine WIPP's suitability as a TRU waste repository.

In June 1994, the U.S. Secretary of Energy issued a policy designed to reduce the cost and time required to complete DOE environmental impact statements. Now, as the DOE attempts to complete its regulatory obligations to begin WIPP disposal operations, the DOE is developing the WIPP's second supplemental environmental impact statement (SEIS-II). To complete the SEIS-II in full compliance with NEPA and the stringent requirements of the secretarial policy, the DOE's Carlsbad Area Office will have to successfully rise to a number of challenges. These challenges include: preparing the SEIS-II, while simultaneously finalizing other critical-path compliance documents; controlling SEIS-II costs, while pursuing timely completion and ensuring consistency with compliance documents; keeping stakeholders involved with the NEPA process; and developing contingency plans to address factors that could affect the SEIS-II completion.

INTRODUCTION

To strengthen the effectiveness of the Department of Energy's (DOE) environmental clean-up efforts, a number of individuals have argued in favor of the accelerated opening of the Waste Isolation Pilot Plant (WIPP). In an August 1995 congressional task force report to Speaker Newt Gingrich, entitled The Top 20 Ways to Turbocharge DOE Cleanup (1), task force members made the following statements:

The WIPP is a world-class facility designed specifically for the management of transuranic waste. Unfortunately, it presently sits idle. The facility costs American taxpayers millions of dollars to maintain, while related testing and regulatory hurdles are ongoing. Failure to open WIPP also costs taxpayers millions in related storage, inspections, and monitoring for waste destined for shipment to the WIPP, but which currently remain stored at numerous sites around the country. It's time to open this critical facility.

BACKGROUND

From the late 1970s to date, the DOE and its regulators have scrutinized the WIPP facility to ensure that applicable environmental and safety standards are being met. The DOE has continued to assess the WIPP facility in order to ready the WIPP to demonstrate the safe disposal of transuranic (TRU) waste resulting from United States defense activities. The National Environmental Policy Act (NEPA) (2) requires that federal agencies consider potential environmental impacts of its proposed action and alternatives before deciding to proceed with new projects and activities. NEPA also encourages the DOE consider public comment in its decision making process.

In 1980, to comply with NEPA and further the DOE's decision-making process, the DOE prepared an environmental impact statement (EIS) (3) and

issued a Record of Decision (ROD) (4) in early 1981 to initiate the first phase in development of the WIPP, i.e., to construct the WIPP surface and underground facilities.

In 1990, DOE issued a supplemental EIS (SEIS) (5) to analyze potential environmental impacts resulting from "new information or changed circumstances" (6). These changed circumstances included the elimination of planned experiments with high-level radioactive wastes, and the introduction of phased experiments. These circumstances further defined the proposed actions presented in the 1980 EIS.

The 1990 SEIS ROD (7) stated that the DOE would continue with the WIPP's development by conducting test phase activities in the WIPP underground facility to demonstrate compliance with applicable radioactive disposal regulations. Since that ROD was issued, the DOE announced that it would conduct tests in above-ground laboratories rather than the WIPP's underground facility to save time and costs while obtaining comparable test results(8).

Now, in 1996 the DOE's Carlsbad Area Office (CAO) is facing the final round of regulatory challenges associated with initiating disposal operations at the WIPP. Among these challenges will be the expeditious completion of a second Supplemental Environmental Impact Statement (SEIS-II). The SEIS-II is being prepared, in part, to fulfill a commitment made in the 1990 ROD, which stated that prior to making a decision to proceed into full disposal operations, another supplemental EIS would be prepared. The SEIS-II will examine the impacts of pertinent "new information or changed circumstances" relative to WIPP disposal operations that have become apparent since 1990.

A MORE EFFICIENT NEPA PROCESS; DOING MORE FOR LESS

All government agencies are faced with doing more for less, and the DOE is no exception. In June of 1994, the Secretary of Energy, Hazel R. O'Leary, issued The Secretarial Policy on the National Environmental Policy Act (9). Her proclamation, later formalized within the DOE's NEPA Compliance Order of July 1995 (10), challenged the DOE to minimize costs and the time required for EIS preparation and review. The policy introduced numerous reforms including the requirement that the median preparation time for DOE EISs would be reduced to 15 months, with no decline in quality. It further required that each EIS be assigned a specific DOE document manager to oversee the EIS during all stages of its development. Briefly stated, the policy set goals for the DOE to perform its NEPA business better, faster, smarter, and cheaper.

To complete the SEIS-II in compliance with NEPA and meet the goals set by the Secretarial NEPA Policy, the CAO will have to successfully rise to a number of challenges. The following are samples of these challenges:

- Preparing the SEIS-II, while simultaneously finalizing other critical-path, compliance documents
- Controlling SEIS-II costs, while pursuing timely completion
- Ensuring completion of an objective, thorough analysis
- Ensuring stakeholder involvement in the NEPA process
- Developing contingency plans to address factors that could affect the SEIS-II completion.

SIMULTANEOUS DEVELOPMENT OF COMPLIANCE DOCUMENTS

Between now and October of 1997, the CAO intends to prepare and obtain regulatory approval several compliance documents. Major compliance documents include:

- the No-Migration Variance Petition

the revised Resource Conservation and Recovery Act Part B Application
the 40 CFR 191 Compliance Certification Application.

During this same time frame, the CAO must also demonstrate its readiness to receive and dispose of TRU waste by performing an operational readiness review (ORR). The ORR is a formal process that evaluates all aspects of the WIPP facility's ability to accomplish its mission of the safe disposal of TRU waste in a geologic repository.

Without explicit coordination and planning, concurrent document preparation and completion of the ORR could present stumbling blocks to reaching an expeditious, disposal decision. Coordinating the flow of information is not an easy task. Using the most current information and analyses as they become available will be crucial to make the SEIS-II a quality, decision-making document.

To turn potential stumbling blocks into stepping stones, the SEIS-II team is reviewing data from WIPP's other compliance documents and studies as they are developed to "feed" the SEIS-II analysis. As new data become available, preparation team members glean information, and review it within the context of the NEPA analysis. The SEIS-II team is practicing the teamwork philosophy set forth in the secretarial policy and new DOE Order by working together with the CAO compliance manager, thus assuring that consistency is preserved in the WIPP's regulatory compliance documents.

CONTROLLING THE SEIS-II SCHEDULE AND COSTS

Controlling costs and compressing the schedule for EIS development were clearly priorities in the Secretarial policy. Fulfilling these expectations while developing the SEIS-II will be especially demanding. The SEIS-II will examine environmental impacts throughout the DOE complex and discuss issues that generate national interest. To control costs and keep the analysis on schedule, the CAO is pursuing a number of strategies.

A reference library was created prior to beginning the preparation of the SEIS-II. A full-time librarian began gathering needed reference documents in advance of the SEIS-II Notice of Intent (11) so most of the required reference documents would be available when document preparation began. When the SEIS-II contractor analysts began their work, much of the needed reference materials were already at their finger-tips.

Extensive use of information contained in other available WIPP studies and compliance documents was also planned to avoid rewriting "boiler plate information" or reanalyzing topics already covered within another report. The WIPP's Annual Site Environmental Report (12) is being used to capture data on the existing environment data. Analyses contained in the draft EM Programmatic Environmental Impact Statement (13) are being used to present data on the impacts of waste characterization, packaging, and treatment at the generator sites. The WIPP's Safety Analysis Report (14) is being used to present analysis of potential routine operations and accident scenarios. Existing electronically filed documentation is used whenever possible, rather than retyping text requiring little or no change.

The CAO is relying on a cohesive, DOE management/review team to assist in controlling SEIS-II schedule and costs. The team is comprised of staff-level personnel from the CAO and various branches of the DOE Headquarters who are responsible for reviewing and approving the SEIS-II during all phases of its development. The management/review team is working to assure that a high quality document is prepared and that it is consistent

with DOE's programmatic policies. The management team has concurrently reviewed planning documents such as the SEIS-II annotative outline (15) and the SEIS-II Implementation Plan (16). The team will take the same concurrent approach for reviewing the draft SEIS-II, the final SEIS-II, and its resulting draft ROD. Reaching early agreement on which alternatives should be evaluated in the SEIS-II and the overall approach for that evaluation is reducing the probability of midstream changes in course that would delay the SEIS-II preparation.

For SEIS-II scoping meetings, the CAO chose locations where stakeholders have traditionally demonstrated the most interest in the WIPP. This strategy enabled the CAO to reach the most interested people with the least number of meetings. The process applied a "lesson learned" from the SEIS of 1990. In some of the 1990 scoping meetings, DOE representatives outnumbered the public attendees five to one (17), which did not represent an efficient use of taxpayer dollars.

To reduce travel expenses, the SEIS-II public outreach team chartered its air travel, and used community centers whenever possible for stakeholder meetings. The document preparation team uses phone and video conferences in place of face-to-face meetings whenever feasible. These measures allow participants from Carlsbad and Albuquerque, New Mexico; Richland, Washington; and Washington D.C.; to interface regularly while minimizing travel expenditures.

The CAO is obligated to assure that mitigation-measure benefits are commensurate with their costs. To responsibly manage taxpayer dollars, each mitigation commitment will be carefully reviewed before it is recommended for inclusion in the ROD. This will ensure that necessary precautions are taken for the protection of the workers, the public, and the environment, and that the maximum value is added for mitigation dollars spent.

ENSURING COMPLETION OF AN OBJECTIVE THOROUGH ANALYSIS

The DOE has already prepared two environmental impact statements, and through the NEPA process, has reached decisions relative to proceeding with construction and development of the WIPP. This has led many people to question whether the SEIS-II will really support DOE decision making or merely justify a disposal decision that is the logical result of the phased development process DOE has been pursuing since the 1980 EIS.

The CAO is dedicated to making the SEIS-II more than NEPA lip-service; more than a paper-trail of formality. It is important that the DOE remain open to changing its course of action, even at this late date, if new information or changed circumstances reveal that TRU waste in its current form is unsuitable for disposal at the WIPP. However, even if it is assumed that SEIS-II analysis does not change earlier conclusions, it will still inform the CAO throughout its decision-making process. For example, the WIPP Waste Acceptance Criteria (18) could be further modified as the analysis in the SEIS-II is completed. The DOE has yet to make a final decision whether to backfill the repository, or what other engineered alternatives might be used in addition to shaft seals. These and other engineered barrier alternatives will be explored in the SEIS-II.

ENSURING STAKEHOLDER INVOLVEMENT

Although scoping is not required for supplemental EISs (19), the CAO recognizes the importance of public participation during all phases of its decision making. Stakeholder participation was also emphasized in the

secretarial policy reforms. For the SEIS-II, planning, planning, and more planning have been the keys to maximizing stakeholder input.

Following the issuance of the Notice of Intent in August 1995, the CAO began working with the public outreach team in earnest, preparing the participants to become less threatening, less bureaucratic, and more responsive to public concerns. Training sessions (complete with practical exercises) were conducted for potential public outreach team members, and a format was set for the scoping meetings that would encourage and stimulate public comments.

Two of the topics emphasized during the training (20) were the importance of being an active listener and not a lobbyist. CAO representatives were urged to view stakeholders opposed to WIPP not as adversaries, but rather as individuals who are genuinely concerned about the problems posed by TRU waste disposal. Role playing exercises were conducted and potential public questions were fielded.

The SEIS-II scoping meetings were conducted as simple, information fairs. Members of the public were invited to join with CAO's representatives for round-table discussions, view several information displays, and ask questions. To stimulate discussion, information booths were arranged with challenging questions posted at the top of each. Members of the public were encouraged to ask questions or give comments as they preferred. Comments were recorded by note takers on flip charts and posted on the walls so that everyone could see what comments had already been made. SEIS-II team members asked clarifying questions if comments were unclear. Finally, the DOE extended the official time frame for scoping and receiving public comments (21) and held an additional scoping meeting for stakeholders in the Denver metropolitan area. The additional meeting was held in response to concerns that stakeholders had been faced with schedule conflicts (another scoping meeting was being held during the same time frame for a different DOE EIS).

CONTINGENCY PLANNING

Properly applied, the NEPA compliance process leads to informed decisions - decisions that are made with the full knowledge of their likely environmental impacts and the public's input regarding what the DOE decision should be. However, as the producer and custodian of radioactive wastes and materials perceived by many as extremely dangerous, the DOE's decisions are often scrutinized to a greater degree than those of other government agencies. Years of experience preparing EISs have demonstrated that the NEPA compliance process is frequently a point of attack for those who disagree with the DOE's decisions and wish to delay or stop DOE projects.

With the high potential for litigation in mind, the CAO formed a contingency planning group. This group included members of the SEIS-II preparation team, compliance personnel, and legal counsel. Together, members identified potential risk factors, weighted these risks by priority, and evaluated mitigation actions to reduce or eliminate these risks.

CONCLUSION

The CAO is dedicated to completing the SEIS-II in a manner that is consistent with the secretarial policy, preparing this environmental impact analysis concurrently with other key compliance documents, while controlling schedule and costs. The American taxpayers and WIPP stakeholders have a right to expect that decisions surrounding the operation of the WIPP facility are environmentally sound, and that a cost

effective solution for TRU waste disposal is implemented. The CAO, together with DOE Headquarters representatives, its contractors, and stakeholders will assure that the SEIS-II NEPA disposal decision is sound, timely, and cost effective, and that the public is given the opportunity to be involved in the DOE decision making process.

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RISK ASSESSMENT FOR WASTE EMPLACEMENT AT THE WASTE ISOLATION PILOT PLANT

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ABSTRACT

Administered by the Carlsbad Area Office, the Waste Isolation Pilot Plant (WIPP) is designed to permanently dispose of transuranic waste left from U.S. nuclear weapons research and production. Project facilities, located 26 miles east of Carlsbad, New Mexico, include disposal rooms mined 2,150 feet beneath the earth's surface in a stable, ancient salt formation. The U.S. Department of Energy has established and analyzed the safety basis for handling and emplacement of contact-handled transuranic waste for disposal at the WIPP. The safety basis consists of management, design, construction, operation, and engineering characteristics necessary to protect the public, workers, and environment from the safety and health hazards posed by waste handling and emplacement operations. An assessment of hazards and the associated risk to safety was included in the effort.

The hazard assessment technique used at the WIPP is a creative systematic interaction of a multi-disciplinary team that could be applied to other waste management activities. The qualitative assessment approach, which ranks hazards by likelihood and significance of consequence, would be useful for wide range of risk analysis efforts.

Hazards were systematically identified and assessed to evaluate the potential internal, external, and natural phenomena events that can cause the identified hazards to develop into accidents. The hazard assessment employed at the WIPP identified deviations from the intended design and operation of the waste handling system, analyzed potential accident consequences to the public and workers, estimated likelihood of occurrence, and evaluated associated preventative and mitigative features. It was concluded from the assessment that proposed WIPP waste emplacement operations and design are sufficient to ensure safety of the public, workers, and environment.

FACILITY BACKGROUND AND MISSION

The U.S. Department of Energy (DOE) was authorized by Public Law 96-164 (1) to provide a facility for demonstrating the safe disposal of transuranic (TRU) wastes from national defense activities and programs of the United States exempted from regulation by the U.S. Nuclear Regulatory Commission. The Waste Isolation Pilot Plant (WIPP), located in southeastern New Mexico (as shown in Fig. 1), was constructed to determine the efficacy of an underground repository for safe disposal of TRU wastes.

Fig. 1

In accordance with 1981 and 1990 Records of Decision (2,3), the development of the WIPP was to proceed with a phased approach. Development of the WIPP began with a siting phase, during which several potential sites were evaluated. The present site was selected based on extensive geotechnical research, supplemented by testing. The site and preliminary design validation phase (SPDV) followed the siting phase, during which two shafts were constructed, an underground testing area was excavated, and various geologic, hydrologic, and other geotechnical and geochemical features were investigated. The construction phase followed the SPDV phase during which surface structures for receiving waste were built and underground excavations were completed for commencement of waste emplacement. At the conclusion of the construction phase, the DOE proposed a test phase, to be followed by the disposal phase for waste emplacement operations. The test phase was to involve the use of limited quantities of contact-handled (CH) TRU waste to conduct tests in the WIPP repository to provide data for reducing the uncertainties in the performance assessment required for compliance with the long-term waste isolation regulations of the U.S. Environmental Protection Agency (EPA), Subpart B of 40 CFR Part 191 (4). To enable the receipt of CH TRU waste at the WIPP site for the tests the Congress enacted the WIPP Land Withdrawal Act of 1992 (Public Law 102-579) (5). The law also provides for authorizations of detailed regulatory requirements for the WIPP by the EPA.

As a result of major program redirection in late 1993, the WIPP test phase was modified by substituting the previously planned WIPP underground radioactive tests with laboratory tests. In conjunction, WIPP operations will proceed directly with the disposal phase. CH TRU waste emplacement operations are currently scheduled to start in October 1998, assuming successful demonstration of compliance with applicable federal and state laws and regulations, and successful completion of the WIPP CH TRU operational readiness review. The CH TRU operational readiness review will closely examine the safety bases of the facility and the status of attendant conformance to ensure that the facility is operationally ready and that contact-handled waste emplacement operations will be conducted safely.

The disposal phase is scheduled to last 35 years, will consist of receiving, handling, and emplacing TRU waste in the repository for disposal, and will end when the design capacity of the repository has been reached.

FACILITY DESIGN

The WIPP is located in Eddy County in southeastern New Mexico, as shown in Fig. 1. The amount of land that has been set aside for the WIPP comprises an area of 16 square miles (mi2). The WIPP is located in an area of low population density with less than 30 permanent residents living within a ten-mile radius. The area surrounding the facility is used primarily for grazing and development of potash, oil, and gas resources. Development of these resources results in a transient population (non-permanent) consisting principally of workers at three potash mines that are located within ten miles of the WIPP. The largest population center nearest the WIPP is the city of Carlsbad, 26 miles to the west, with approximately 25,000 inhabitants. Two smaller communities, Loving (population approximately 1300) and Malaga (population approximately 200), are located about 20 miles southwest of the WIPP site. As the result of land use restrictions imposed by the U.S. Bureau

of Land Management, and administrative action by the DOE to purchase lease holdings, no resource development is allowed within the 16 mi2 that have been set aside for the WIPP (with the exception of existing leases). The WIPP is designed to receive and handle a maximum of 500,000 cubic feet per year (ft3/yr) CH TRU waste and 10,000ft3/yr remote handled (RH) TRU waste. The CH TRU waste will be contained in 55-gallon drums and standard waste boxes. The WIPP facility is designed to have a disposal capacity for TRU waste of 6.2x106 ft3. Current design is that RH waste will be packaged in steel canisters and transported to the WIPP facility in shielded road casks. The WIPP facility has sufficient capacity to handle the 250,000 ft3 of RH TRU that was established in the 1981 Record of Decision as a total volume. In addition, the Consultation and Cooperation Agreement (6) between the State of New Mexico and DOE and the Land Withdrawal Act of 1992 limit the total RHTRU activity to 5.1 x 106 curies.

CH TRU* wastes will be disposed of in a 100-acre disposal area located 2,150 feet beneath the surface in a deep bedded salt formation. Waste will be transferred from the surface to the disposal area through a waste shaft using a hoisting arrangement. The disposal phase is currently scheduled to last for 35 years, followed by a 10-year decontamination, decommissioning and closure phase.

The WIPP facility is divided into three basic structural components: surface structures, shafts, and subsurface structures as shown in Fig. 2. The WIPP surface structures accommodate the personnel, equipment, and support services required for the receipt, preparation, and transfer of waste from the surface to the underground. The surface structures are located in an area within a perimeter security fence. The primary surface operations at the WIPP are conducted in the Waste Handling Building. The CH TRU waste handling area includes the entrance air locks, CH Bay, a shielded holding area, an overpack and repair room and CH TRU support facilities.

Fig. 2

FACILITY OPERATIONS

The principal operations of the WIPP facility involve the receipt of TRU waste and TRU waste mixed with hazardous waste, unloading, moving and emplacement of the waste in underground rooms. Transporters carrying TRU waste arrive at the WIPP and are unloaded outside the Waste Handling Building. The shipments are surveyed for external contamination prior to their movement into the Waste Handling Building for unloading.

CH TRU waste will be shipped to the WIPP in Nuclear Regulatory Commission certified shipping containers. After the CH TRU waste shipping container is inspected for contamination, the loaded shipping container is moved into the Waste Handling Building and placed on a handling dock. The container is opened, surveyed for radiation and contamination levels, and the waste containers are removed and placed on a facility pallet. This pallet is then transferred to the conveyance loading car, which is moved into the hoist cage in the Waste Shaft for transfer to the underground disposal horizon. At the disposal horizon, the pallet is removed from the hoist cage, placed on the underground transporter, and moved to a CH TRU waste disposal room. In the disposal room, the containers are removed from the pallet and placed in the waste stack. The empty pallet is returned to the surface for reuse.

The waste received for disposal at the WIPP facility must conform with the WIPP Waste Acceptance Criteria (7). The operational philosophy at the

WIPP facility is to start radiologically clean and stay radiologically clean. Consequently, any containers of waste that are found to be externally contaminated or damaged will be decontaminated or placed in a larger container as required. Also, any local area of contamination will be isolated and/or decontaminated prior to continuation of the waste handling process.

HAZARD EVALUATION

The WIPP CH TRU waste handling process was qualitatively evaluated using a Hazard and Operability Study (HAZOP) (8). This systematic approach to hazard analysis was conducted by a leader knowledgeable in the HAZOP methodology and consisted of personnel from various disciplines familiar with the design and operation of the WIPP (HAZOP Team). The HAZOP Team identified deviations from the intended design and operation of the waste handling system that could: 1) result in process slowdown or shutdown, 2) result in worker injury or fatality, and 3) result in the release of waste container radiological and nonradiological materials. The HAZOP Team assigned a qualitative consequence and likelihood ranking for each deviation as discussed below. A hazard evaluation ranking mechanism utilized the likelihood and the most significant consequences to separate the low risk hazards from high risk hazards that may warrant additional quantitative analysis. Based on that ranking approach a basic set of accidents was chosen for further quantitative assessment in the 1995 WIPP Safety Analysis Report (SAR) (9).

HAZARD AND OPERABILITY STUDY METHODOLOGY

The HAZOP technique, based on a creative systematic interaction of a multi-disciplinary team, evaluated the significance of deviations from the normal waste handling process. The HAZOP Team consisted of experienced personnel from Facility Operations, Maintenance Operations, (including previously certified waste handlers experienced in Transuranic Package Transporter (TRUPACT) and drum handling activities), Industrial and Nuclear Safety, Engineering, and Regulatory Compliance.

The HAZOP Team started by examining the process for receipt of a CH TRU waste transporter at the front gate and ended with the process for emplacement of CH TRU waste in a underground disposal room. HAZOP nodes (process steps) were selected to define the movement of CH TRU waste through the WIPP facility. Deviations of proposed waste handling operations were postulated for each node and once the deviation was confirmed to be plausible, the HAZOP Team determined the possible causes for the deviation. The resulting potential consequences were explored without taking into consideration any mitigating features. An evaluation was made to determine if mitigating safeguards were in place to alleviate the consequences. Some of the potential deviation consequences or concerns identified by the HAZOP Team are:

- Worker injury or fatality;
- Process slowdown or shutdown;
- Internal and external conditions may result in breach/rupture of waste containers resulting in the airborne release of radiological or nonradiological hazardous materials (loss of primary confinement);
- External waste container surface contamination and need for decontamination;
- Worker and public exposure to radiation and airborne radiological and nonradiological hazardous materials; and
- Potential for receipt of damaged waste containers and need for overpack operations.

The HAZOP deviation ranking process used a two-number system, consisting of a qualitative severity classification and a qualitative likelihood classification. The qualitative severity (consequence) classification was ranked without consideration for mitigation. The qualitative likelihood was ranked taking into consideration the probability of failure of identified safeguards and mitigation for that deviation.

SELECTION OF CH POTENTIAL ACCIDENTS

The HAZOP provided a list of deviations that were qualitatively ranked by relative consequence and probability using the 'total rank' consequence criteria of Table I and the probability criteria of Table II. This resulted in the 'total rank' that combines both hazard consequence and probability ranking components (see Fig. 3). As stated in the HAZOP, the consequence ranking of each deviation included both the resultant consequence to the worker and the radiological and nonradiological consequence to the offsite public.

Table I

Table II

Fig. 3

In order to select potential CH TRU waste handling accidents for quantitative accident analysis, the total list of hazards was narrowed to focus on risk posed by radiological and nonradiological hazardous material. This eliminated occupational deviations exclusive of the hazardous materials involved, providing a subset 'hazard rank'.

In order to determine the risk associated with each deviation, the relative probability and hazard consequence ranking were combined. The deviations were then categorized as low, moderate, or high risk based on the Relative Probability and Consequence Ranking Matrix (Fig. 3). Those deviations with a combined 'hazard rank' of less than four were excluded from further quantitative evaluation in the 1995 WIPP SAR, with the exceptions of the waste hoist drop (CH5), earthquake (CH6), and aircraft crash (CH8). The waste hoist drop (CH5) was selected for its significant interest to external organizations, as well as the earthquake (CH6) as a natural event and the aircraft crash (CH8) as an external event. A list of the selected deviations for further consideration in the 1995 WIPP SAR accident analysis is provided below.

Operational Events

CH1 Spontaneous Ignition (Drum) in the Waste Handling Building

CH2 Crane Failure in the Waste Handling Building

CH3 Puncture and Drop of Waste Containers by Forklift in the Waste Handling Building

CH4 Drop of Waste Containers by Forklift in the Waste Handling Building

CH5 Waste Hoist Failure

CH7 Spontaneous Ignition (Drum) in the Underground

CH9 Drop of Waste Containers by Forklift in the Underground

Natural Events

CH6 Seismic Event

CH10 Tornado Event

CH11 Underground Roof Fall

External Events

CH8 Aircraft Crash

PROTECTION OF WORKERS FROM ACCIDENTS

The HAZOP analyses of the CH TRU Waste Handling System identified a number of waste handling process hazards that could potentially lead to events resulting in work injury or fatality, or exposure to radiological

and nonradiological hazardous materials. Consistent with the defense-in-depth philosophy, and the philosophy of Process Safety Management, as published in 29 CFR 1910.119, "Process Safety Management of Highly Hazardous Chemicals," (10) reduction of the risk to workers from accidents is accomplished at the WIPP by identifying controls to prevent the event from happening. Total risk is therefore lowered by reducing the likelihood of the event, as opposed to focusing on post accident consequence mitigation through the performance of quantitative consequence calculations for workers.

The HAZOP Team identified a significant number of existing preventative safeguards that lower the likelihood of occurrence of each deviation, substantially reducing the risk to workers. The HAZOP Team concluded substantial safeguards currently exist at the WIPP to prevent or reduce the likelihood of such deviations from occurring. Identified preventative safeguards generally include the following:

Facility and equipment design, application of appropriate design classification and applicable design codes and standards;

Programs relating to configuration and document control, quality assurance, and preventative maintenance and inspection; and

Administrative controls including the WIPP Waste Acceptance Criteria, waste handling procedures and training, and the WIPP Emergency Plan and associated procedures.

Because of the importance of these preventative features in WIPP defense in depth and worker protection from accidents, Technical Safety Requirement Administrative Controls are assigned in Chapter 6 of the 1995 SAR and required in the WIPP TSR Document.

CONCLUSIONS

The HAZOP Team reviewed the WIPP CH TRU waste handling system to identify deviations from the intended design and operation that could result in hazards that could create adverse consequences to the public and the worker, or result in a process slowdown. Each step of the waste handling process was evaluated, and recommendations were made, where appropriate, to preclude undesirable consequences.

Some of the identified potential hazards or concerns are:

Personnel exposure to radiation and airborne radiological an nonradiological hazardous materials;

Personnel injury or fatality;

Potential that internal and external conditions may result in breach/rupture of waste containers resulting in the airborne release of radiological or nonradiological hazardous materials (loss of confinement);

Potential for external waste container surface contamination and need for decontamination; and

Potential for process slowdown or shutdown.

As stated above, the consequences of each deviation were developed without considering any existing mitigating systems. Identified unmitigated potential airborne release of radiological or nonradiological hazardous materials may result in varying degrees of consequences to workers and the offsite public. However, the HAZOP Team identified:

Substantial safeguards currently exist at the WIPP to prevent or reduce the likelihood of such deviations from occurring. Identified safeguards include procedures, training, preventative maintenance and inspection, and administrative controls including the WIPP Waste Acceptance Criteria.

Substantial mitigation exits to reduce the consequences of any postulated deviation to acceptable levels. Identified mitigation includes the design of the WIPP confinement/ventilation systems and associated HEPA filtration systems.

Postulated credible scenarios that may result in worker injury or fatality, or potential airborne release of radiological or nonradiological hazardous materials and potential exposure to workers and the offsite public, will further be analyzed quantitatively in the WIPP Safety Analysis Report.

However, as qualitatively concluded from the HAZOP analysis, the adequacy of the design of the WIPP CH TRU waste handling system ensures the safety of the public and the worker. The HAZOP Team identified no substantial recommendations for the WIPP management to consider to reduce the severity or likelihood of any of the postulated deviations. Thus, the WIPP facility is ready to commence safe TRU waste handling and emplacement operations.

ACKNOWLEDGMENTS

The Westinghouse Waste Isolation Division team that prepared the WIPP Safety Analysis Report (SAR) (November 1995) included Messrs Doug Gerstner, Fred Bourger, Carl Ortiz, James McCormick, and Dan Standiford. The subject of this report was a small component of the effort that resulted in the 1995 WIPP SAR. I thank them for that job well done and the useful discussions and input that contributed to this report.

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QUALITY ASSURANCE PLAYS A KEY ROLE IN GETTING THE WASTE ISOLATION PILOT PLANT TO OPERATIONAL STATUS

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ABSTRACT

There are many regulatory requirements that must be met before transuranic (TRU) and transuranic mixed wastes are emplaced into the Waste Isolation Pilot Plant (WIPP) for final disposal. These requirements will be satisfied by the Department of Energy (DOE), through the DOE Carlsbad Area Office (CAO) and will be verified through the development and implementation of a performance-based quality assurance program, consisting of numerous quality assurance program plans and procedures. These plans and procedures relate to experimental programs, performance assessment, operations, waste characterization, and waste certification. Effective implementation of quality assurance principles have played and will continue to play a critical role in supporting the WIPP certification and operations processes.

For example, in order for the CAO to receive and dispose of TRU Waste, it must submit a Compliance Certification Application to the U.S. Environmental Protection Agency (EPA), demonstrating that the disposal of TRU Waste at the WIPP complies with the environmental radiation protection standards promulgated by the EPA in the final disposal standards, i.e., the Code of Federal Regulations, Title 40, Part 191 (40 CFR 191), Protection of the Environment-Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Wastes. Likewise, to receive and dispose of mixed TRU Waste, the CAO has to demonstrate to both the EPA (No-Migration Variance Petition) and the New Mexico Environment Department (NMED) (Resource Conservation and Recovery Act Part B Permit) that disposal of mixed TRU Waste does not impose undue risks to present and future generations.

QA REQUIREMENTS

The WIPP project is subject to the requirements of 40 CFR 191 and 10 CFR 830.120, Nuclear Safety Management, Quality Assurance Requirements. 40 CFR 191 requires that the DOE implement a quality assurance (QA) plan. This QA plan is expected to meet the requirements of the American Society of Mechanical Engineers (ASME) NQA-1, Quality Assurance Program Requirements for Nuclear Facilities, NQA-2 (sub-part 2.7), Quality Assurance Requirements of Computer Software for Nuclear Facility Applications, and NQA-3, Quality Assurance Program Requirements for the Collection of Scientific and Technical Information for Site Characterization of High-Level Waste Repositories. Also, DOE contractors are required to meet the QA requirements of 10 CFR 830.120.

CAO QAPD

The CAO Quality Assurance Program Description (QAPD) establishes the QA requirements for WIPP. The quality assurance requirements from the applicable federal, state, and DOE rules, regulations, and orders are identified in Table I. Figure 1 shows the flowdown of QA requirements into the lower level QA plans and procedures. The QAPD is structured in accordance with the requirements of NQA-1 and DOE Order 5700.6C, Quality Assurance. Software QA requirements from NQA-2, Part 2.7 are also included in the QAPD. Since WIPP is also a research and development project, involving scientific investigation, the DOE also implements the applicable portions of NQA-3.

Table I

Fig. 1

PARTICIPANT QA REQUIREMENTS

Each participant is required to prepare, implement, and maintain a QA program that meets the CAO QAPD. Participant QA programs are approved by the next higher organizational level of WIPP participant. Participants use a "graded approach" to implement their QA program requirements. The QA plans are implemented through instructions and procedures at each participant level. The requirements of the QAPD are applied to activities and items important to nuclear safety, waste isolation, regulatory compliance, and the WIPP mission. Participants determine the level of controls that are necessary to meet all the applicable QA requirements; these controls are reviewed and approved by the next higher organizational level. This determination is reviewed by the CAO staff during oversight activities such as audits, surveillances, and program reviews.

CAO QA

In 1995, the CAO decentralized the QA organization and placed QA engineers into each of the technical organization. The CAO QA Manager retains the responsibility and authority for the overall QA program. Each QA engineer now works directly with the CAO technical teams and offices to implement the QAPD, both internally and at the participants.

SNL QA

Most of the WIPP experimental work over the past 20 years has been conducted through the offices of Sandia National Laboratories (SNL) and Los Alamos National Laboratory (LANL). Much of the data was not collected under a QA program that was based on NQA-1. Consequently, CAO and SNL have had to establish methods for qualifying existing data. The methods being used were developed using the guidelines described in NUREG 1298, Qualification of Existing Data for High-Level Nuclear Waste Repositories. Since this process has to eventually receive approval of the Environmental Protection Agency (EPA), the CAO has been pro active in facilitating the oversight and informal acceptance of the process by the EPA Office of Radiation and Indoor Air (ORIA) and the Environmental Evaluation Group (EEG).

Additional experimental programs work has been supported through the auspices of SNL subcontractors. These include the Pacific Northwest Laboratory (PNL), Argonne National Laboratory-West (ANL-W), Lawrence Berkeley Laboratory (LBL), Florida State University (FSU), Stanford University, RE/SPEC, Intera, and several dozen more. SNL performs annual audits of the QA programs at the subcontractor sites. CAO has observed several of these audits over the past two years to assure that SNL is performing comprehensive and effective audits.

SNL is also responsible for implementing the repository performance assessment (PA) activities. PA involves millions of data points collected from hundreds of different data sources, feeding into several dozen major PA computer codes. SNL has recently completed a significant amount of work to upgrade their software documentation and practices to meet the requirements of NQA-2, Part 2.7. CAO has the responsibility to perform oversight of the SNL activities to assure that the requirements have been met.

QA FOR GENERATOR SITE WASTE CHARACTERIZATION AND CERTIFICATION

Another large part of the WIPP QA program activities involves the waste characterization and waste certification activities at the generator sites. CAO oversees QA program activities at the Idaho National

Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Rocky Flats (RF), the Hanford Facility (HF), the Savannah River Site (SRS), Oak Ridge National Laboratory (ORNL), and the Nevada Test Site (NTS). Since WIPP will initially receive waste from INEL, RF, and LANL, CAO has concentrated oversight on these sites. CAO performs generator site audits and surveillances of activities such as waste characterization, transportation, packaging, handling, and waste certification.

WID QA

Westinghouse is the Management and Operating Contractor for the WIPP site. CAO performs an annual audit and periodic surveillances of the Westinghouse QA program. These assessments are performed in order to evaluate site activities and areas subject to the controls of the CAO QAPD. These areas include waste handling, hoisting, environmental monitoring, safety, maintenance, configuration management, design, and training. One of the more important Westinghouse QA program elements is the control of QA records. Westinghouse has been tasked with maintaining the overall QA records storage facility for the WIPP. Controlling QA records is critical to proving that CAO and participants have adequately addressed and effectively implemented QA programs. These records will be reviewed over and over and evaluated by our regulators in the upcoming years.

REGULATOR QA INTERFACE

Interface with our regulators has been and will continue to be crucial to the success of the WIPP project. The EPA has been reviewing and commenting on WIPP QA plans and procedures since 1993. Also, the EPA has participated as observers on numerous QA audits and surveillances since early 1994. These overview activities enable the EPA to gain confidence in the CAO and participant QA programs. The New Mexico Environmental Department (NMED) will also observe audits in the near future. NMED, to this point, has also been reviewing various CAO QA program documents.

SUMMARY

In summary, the QA activities supporting WIPP are being planned, implemented, and verified in a comprehensive and effective manner. This level of implementation must and will continue in the future. It is essential that WIPP participants continue to meet their diverse QA programs throughout the life of the project. The CAO will continue to perform assessments of QA program implementation to demonstrate to DOE, the regulators, and the public that effective controls are in place to assure that the WIPP facility will not allow the release of radiation that may be harmful to the public or the environment.

Session 20 -- FEDERAL FACILITY COMPLIANCE ACT (FFCA) IMPLEMENTATION--HOW DID WE DO?

Co-chairs: Lawrence H. Harmon, MACTEC

20-1

THE FEDERAL FACILITY COMPLIANCE ACT - CONTINUING THE DIALOGUE

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ABSTRACT

The Federal Facility Compliance Act of October 1992 required the Department of Energy to prepare an inventory of all mixed waste and to prepare site treatment plans for mixed waste. Mixed waste is defined as waste containing both hazardous and radioactive constituents. The Act also waived sovereign immunity for storage violations of mixed waste subject to land disposal restrictions, but allowed the Department three years to put treatment plans in place that were subject to approval by the States and the Environmental Protection Agency. The Department effort to inventory and plan for the treatment of mixed waste began with the involvement of the State and EPA regulators early in the process. The success of this effort is due, in large part, to the interactive cooperation between the Department and the regulators that developed from working the issues related to this Act.

This success can be measured in two ways:

By October 6, 1995, over 90% of all mixed waste, 95.9% of all mixed transuranic waste, and 99.9% of all high level waste was covered with a negotiated agreement. The remaining waste was either covered by a unilateral order or an order pending completion by the States with letters in place stating that enforcement action would not occur while the States were completing their orders.

Compliance orders are in place as follows:

- Twenty-five (25) negotiated compliance order and one negotiated unilateral order were in place on October 6, 1995.
- Three unilateral orders were in place October 6, 1995; of which only one order requires appeal by the Department.
- Six other compliance orders were well developed such that no significant potential for fines and penalties existed.

However, there are larger successes that occurred as a result of the activities started under the framework of the Federal Facility Compliance Act. The process begun in the Act has had far-reaching effects on how the Department conducts business with both the regulators and the sites. An integrated budgeting process implemented by the Assistant Secretary in an effort to satisfy conflicting state and local needs with constrained funding is an indication of a partnership established between regulators and DOE during the three year process. The balance of all environmental restoration and waste management activities at a site now reflects the desires of DOE and the stakeholders.

BACKGROUND SUMMARY

The magnitude of the project begins to define the complexity of the issues. Site treatment plans were required for over 2,200 separate waste streams consisting of approximately 471,000 cubic meters of high-level waste, 129,000 cubic meters of mixed low-level waste, and 52,000 cubic meters of mixed transuranic waste located at 41 sites in 22 states.

The Department decided early that the site treatment plans would be drafted in a three-phase approach. The conceptual plans would be delivered to the States/EPA in October 1993, the draft plans in August 1994, and the proposed or final plans in March 1995. This would allow time for feedback from stakeholders as each phase of the plan was completed. It would also allow the States approximately seven months to approve, disapprove, or approve with modifications the site treatment plans. A schematic of this process is presented in Fig. 1.

Fig. 1

The early attitudes of both DOE and the States were not cooperative and clearly provincial. Each state wanted to be independent with no waste being transferred in or out of their state. DOE wanted a more national approach using better economies of scale with fewer treatment facilities, saving more dollars, but transferring more waste out of state.

To involve the States/EPA, the Department initially held a meeting of state/EPA representatives in December 1992 and suggested a national compliance plan, a top-down approach. The States responded negatively to this suggestion, requesting the Department provide a bottom-up approach--each site should decide for itself how to treat its waste and then begin the discussions with its regulators. This resulted in an initiative between the regulators and the Department to work jointly on this project. The Department funded the National Governors' Association to act as the channel for information between the States/EPA and the DOE. This liaison provided the cornerstone for improved communication between DOE and the States/EPA.

The iterative process initiated between the States/EPA and DOE only worked with considerable effort to engage all the parties. The communication effort included the Policy Coordinating Group, the Options Analysis Team, the National Governors' Association, DOE senior management, DOE site management, DOE site representatives, State regulators, EPA regulators, State Governors, Indian Tribal Nations, and public stakeholders. A description of the roles of the ad hoc groups involved in the process is given in Table I.

Table I

ISSUES RAISED AND RESOLVED

Many issues became non-issues when raised with all the parties involved. The most significant examples of these complex issues include duplicate treatment operations among the sites, off-site waste shipments, treatment plans for mixed transuranic waste, and the impact of major funding shortfalls on meeting the objectives of the plans.

After the submittal of the draft site treatment plans, the States/EPA and DOE met to discuss further development. In a marked departure from the initial attitudes, the States/EPA requested DOE to eliminate the duplication of treatment operations, even at the expense of increasing offsite waste shipments. Involvement of the National Governors' Association in the Options Analysis Team meetings as a representative for the States/EPA facilitated a mixed low-level waste configuration that was more cohesive, less costly, and more acceptable to all the parties.

The issue of offsite waste shipments was an area of potential conflict because the States/EPA wanted no waste coming into their particular state or region. DOE wanted to ship waste offsite for treatment because of economies of scale in building one treatment plant rather than several. This issue was resolved by a compromise that satisfied most of the State principles in handling offsite waste shipments and met DOE's objective of fewer new treatment facilities.

The issue of treatment for mixed transuranic waste was raised later in the process. A near-term compromise solution was reached as the deadline for submitting the plans approached. The compromise recognized the lengthy process involved in reaching agreement on a mixed low-level waste configuration and allowed for an interim action. In the interim, the States/EPA agreed to beginning negotiations on the mixed transuranic issue if the scheduled opening of the Waste Isolation Pilot Project is

delayed beyond 1998. Work continues toward defining a mixed transuranic configuration.

The issue of funding shortfalls is perhaps the area of greatest achievement for the process. In late 1994, funding constraints imposed by Congress mandated that DOE work out longer schedules for treatment milestones than originally planned. In order to fully engage the States/EPA in this scheduling process, DOE invited the States to become partners with DOE in setting budget priorities for their states including all aspects of Environmental Management--Environmental Restoration, Waste Management, and Nuclear Materials and Facility Stabilization. This process was to prepare the FY97 budget submission and will continue into the foreseeable future with each site preparing an integrated budget, balancing the needs of that site with available funding. The budget process was successful due to the previous working relationships established in preparing the site treatment plans.

RESULTS OF THE PROCESS

In the last three years, DOE has forged a new way of doing business with the States/EPA. The process implemented under the Act includes open, honest, and frequent communication to better understand and match both the needs of the regulators with the Department. The primary result has been a gradual change in attitudes as seen by the original "Do not send waste to my state" syndrome to one of partnership in addressing waste management and remediation needs.

The effects of this partnership can be measured in terms of a new method of doing business that includes open communication and full sharing of information, successful negotiated compliance orders that cover the majority of mixed waste, no fines or penalties to date, and the continuing priority setting process established for all aspects of Environmental Management at the individual sites.

FEDERAL FACILITIES COMPLIANCE ACT YEAR-3: IMPLEMENTATION, STATUS AND ISSUES STATES PERSPECTIVE: HOW DID WE DO?

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ABSTRACT

The paper presents the status regarding development of Site Treatment Plans and implementation of the Federal Facility Compliance Act in terms of significant issues. Interested persons may find useful the more detailed information which follows.

INTRODUCTION

Over the last several years papers have been submitted concerning the Federal Facility Compliance Agreement (FFCA) from a states perspective. As the process has evolved from the first forum in 1994, we have now reached the critical question. How did we do? The states perspective can be a bit broad knowing that each state has a variety of wants and needs. However, through the efforts of the National Governors Association FFCA task force states were given an arena to pool their needs and wants as a unified body to achieve a much greater degree of success.

The NGA FFCA task force is comprised of a policy representative of the gubernatorial offices and state agencies, as well as regulatory staff from each state. The states included on the task force are those hosting DoE facilities with mixed waste treatment issues. Each state receives input from community and citizen groups, tribal representatives, legal

council and the states business leaders concerned with waste treatment. Each state brings to the table a broad perspective of issues and actions that offer a good building block for dealing with these complex problems. As DoE revamped their decision making process to give greater authority to the regulating agencies and the affected states we have seen a greater willingness on states not to feel their role as antagonist but as partners in addressing the treatment needs of each site and community. As each state has shared information on actions with other states this has allowed the dialogue to focus not only on site specific but DoE complex wide initiatives. As we have seen for the most part, the outcome of decisions from the regulatory authorities (states) in conjunction with DoE and EPA have been better than many could have hoped.

The following is a report completed by states with the NGA and the FFCA process. Ross and Associates out of Seattle help collect much of the data. As you will see much has been accomplished but their is still much to be done.

Report Card

Key requests from states -- What DOE Delivered

- Maximize use of on-site treatment
  - 1.6% of MLLW is targeted out-of-state
- Evaluate use of mobile treatment units to minimize waste movement
  - 10,000 m3 of waste targeted to mobile units
- Use existing facilities wherever possible
  - About 19% of MLLW is targeted to existing facilities
- Evaluate a nearest-site scenario for waste sent out-of-state
  - OAT Team evaluated nearest-site configuration
- Establish satisfactory schedules for waste treatment
  - Consent orders signed or expected in 16 of 20 states
- Address disposal of treated mixed waste
  - Disposal process on-going

The NGA along with participating states formed an Options Analysis Team. This group met with the idea of developing innovative ways to clean the mixed waste at existing sites. Below is a breakdown of their findings.

Options Analysis Team

The Options Analysis Team met in August to determine future configuration changes that might be taken to decrease cost and improve schedules for the treatment of mixed waste.

"...faster or cheaper..."

Potential changes were generally intended for consideration after the approval of STPs, unless earlier consideration was requested by the site's host state (e.g., Ohio).

The OAT Team dealt only with MLLW and Alpha-MLLW. The Carlsbad Area Office is conducting a similar effort to improve the costs and scheduling for mixed transuranic waste.

The OAT team operated under the assumption that treating waste at vendor (commercial or private) facilities would provide cost savings and improved schedules.

Sites identified which treatment options in their plans they expect to propose (after October 1995) to shift to vendor treatment or another DOE facility. The OAT Team also determined which proposed new facilities and wastestreams should be listed as additional "candidates" for vendor or alternate DOE facilities.

Results of OAT meeting: Sites expect to propose shifting about 1,900 m3 to vendor treatment, and identified roughly 10,000 m3 as

"candidate" waste to shift from new DOE facilities to vendor facilities, pending further analysis.

The results of the OAT Team analysis are expected to be proposed to states by the individual DOE sites as orders are reopened for revision. One of the main focuses running not only through DoE but all of the federal government is the move to privatization. This is somewhat a nebulas concept that has not fully been explored by the federal government. With the move to the private sector many of these programs, DoE and many other agencies face a new challenge of defending their existence.

Privatization

STP configuration shows approximately 17% of mixed waste is now targeted to commercial treatment (PSTP configuration was about 6%).

DOE has privatization efforts underway at several sites, including major efforts at Fernald, Hanford, INEL, Oak Ridge.

DOE has established a Commercial Options Analysis Team (COAT)

- COAT's mission is to provide recommendations to DOE on optimization of commercial capability...a national procurement strategy.
- Chaired by Rick Korynta, DOE-Oak Ridge

Conclusion: The STP configuration represents a snapshot in time--the configuration will continue to change, possibly significantly as privatization efforts proceed.

Table I

Mixed Low Level Waste

Most changes affect on-site treatment.

At INEL: A number of treatment units have been consolidated. Some of these treatment units were originally individual treatment steps within larger treatment facilities. The Advanced Mixed Waste Treatment Project (AMWTP) has replaced all of the steps of the previous IWPF and other proposed facilities. All remote-handled and HLW waste will now be treated at one remote handled treatment facility.

Ohio in-state treatment totals decreased 2,647 m3.

At the Portsmouth Site: the site is no longer proposing a Mixed Waste Treatment Facility. A large portion of the waste is now targeted for "commercial offsite incineration," while the remaining waste will be treated on-site by vendors in mobile treatment units.

In-state treatment in Tennessee increased by 5,897 m3. This can be attributed to a large increase in waste awaiting CERCLA RODs at ORNL and Y-12 Sites. This accounts for 5,294 m3 of the increase. Oak Ridge has also added a transportable vitrification system, which will treat 1,870 m3 of waste. Waste with treatment location not specified at Oak Ridge decreased by 2,947 m3. This waste originally had no designated treatment site; it is now bound for on-site treatment and commercial options.

Highlights of changes: PSTP (4/5/95) Configuration to STP (10/4/95)

In-state treatment in Washington decreased by 6,949 m3. Waste targeted to the WRAP IIA facility decreased 2,149 m3, and waste targeted to the commercial thermal treatment facility decreased by 4,799 m3.

The Oakland Area Office (California Sites & Univ. of Missouri Site) re-targeted a total of 32 m3 of waste from WRAP IIA at Hanford to treatment at INEL, due to the uncertainty of the WRAP IIA facility.

Ames Laboratory in Iowa shipped its current inventory to SEG in Tennessee for treatability studies. The site expects to generate waste in the future.

The Nevada Test Site is now targeting 1.8 m3 of waste to the TSCA incinerator in Tennessee.

Many smaller changes occurred in several states. (See State-to-State "mileage charts" for more information.)

Mixed Transuranic Waste

At INEL: alpha-MLLW is now proposed to be treated in conjunction with MTRU waste at the private AMWTP facility.

MLLW FOR WHICH TREATMENT LOCATION IS NOT SPECIFIED

STP Configuration

(23% of 123,455 m3 -- 28,568 m3)

This represents a 9.5% decrease from the amount of waste with no specified treatment location in the April database.

Most of this waste is in TN & OH:

- TN 23,254 m3. 22,214 m3 of this total is targeted for the Oak Ridge Commercial Option. 1,040 m3 is targeted for "commercial disposal".
- OH 3,712 m3. 3,558 m3 of this waste is targeted by the Portsmouth facility for a "Commercial Offsite Incineration Facility."

74% of targeted MLLW is proposed for treatment at new in-state facilities (was 80% in PSTP configuration).

The volume of waste targeted for in-state treatment at new facilities has decreased:

Hanford: 6,949 m3 decrease due to reductions in waste targeted to WRAP IIA and the commercial thermal treatment facility.

Portsmouth: 5,814 m3 reduction due to waste now targeted to "commercial offsite incineration." This waste has been designated as "treatment location not specified."

5,295 m3 of waste at Oak Ridge (Y-12, ORNL) has neither a "new" or "existing" designation; this wastes' treatment is being driven by CERCLA Records of Decision.

Profile of Mobile Treatment

Table II

Volume targeted to mobile (m3)

The amount of waste targeted to mobile treatment units increased by 7,664 m3. The majority of this change can be attributed to the amount of waste targeted for the "Ohio Option - Stabilization Project"(on-site mobile vendor). A total of 5,906 m3 of waste is targeted to this option by Portsmouth (5,487 m3) and Fernald (417 m3).

Six field offices representing six states (ID, NM, OH, SC, TN, TX) are targeting waste for treatment at various mobile facilities.

Five field offices representing six states (ID, NM, OH, SC, TN, TX) will be hosting mobile treatment at their various sites.

4 existing mobile facilities would be used to treat waste in the STP configuration: a lead decontamination trailer from Los Alamos, a Portable Wastewater Treatment Unit at INEL, a sort/survey/char/decon team from Grand Junction, and Thorium Nitrate Tank T-2 at Fernald. 20 new mobile facilities are proposed. Note: DOE-Albuquerque plans to delete several mobile treatment units (other options now believed to be more cost-effective).

Profile of Incineration Facilities

The STP configuration targets waste to the following incineration/thermal treatment facilities:

Three existing DOE facilities (in ID,TN, and SC)

Four existing commercial facilities (in FL, TN(2), and TX)

Two proposed incinerators:

- AMWTP (formerly IWPF) at INEL, Idaho (a proposed private facility)
- Commercial Thermal Treatment Facility, Hanford Site, Washington

Portsmouth is targeting 3,558 m3 of waste to an unspecified commercial offsite incinerator.

Table III

Schedule

The STP database does not provide a coherent look at the schedules for treatment of mixed waste. Schedule information in the STP database is not complete, and the reliability of schedule information that is available is variable. Most schedule information must be surmised from the years that a facility is shown as being funded.

The database does not indicate schedules for specific wastestreams to be treated.

STP documents generally include better schedule information than the database, at least for on-site waste targeted to on-site facilities. However, the STPs do not embody a complete complex-wide integrated schedule for treating MLLW.

MTRU facility schedules appear to more closely match up with WIPP's 25-year operating period (1998-2023) than they did in the PSTPs, although facility operations at Savannah River and Hanford still extend beyond 2023.

Table IV

Costs shown are total life cycle costs (capital plus operating) in constant FY95 dollars for the fiscal years 1995 through 2093, based on verified cost estimates from DOE Operations Offices.

MAJOR CHANGES IN COST ESTIMATES FOR MIXED WASTE TREATMENT

High Level Waste

INEL's New Waste Calcining Facility (NWCF) dropped in cost by $1.2 billion.

INEL's Waste Immobilization Facility (WIF) was deleted (- $4.5 billion) and replaced with the ICPP-Remote Handled Immobilization facility (+ $1.93 billion).

Mixed Transuranic Waste

Oak Ridge's TRU Processing Facility dropped in cost by $0.5 billion.

Hanford's WRAP I facility dropped in cost by $0.6 billion.

Mixed Low Level Waste

Net result of the change from the IWPF facility to AWMTP facility at INEL is a decrease, amount not yet determined.

Virtually no costs are included in the latest database for Hanford's WRAP IIA facility (previously $0.64 billion). Estimated costs of TSCA incinerator increased by $0.59 billion.

Summary of key observations:

Many changes have occurred in the configuration since the 4/5/95 configuration.

The volume of waste targeted to move out-of-state for treatment remained stable at 2,000 m3 (1.6% of MLLW); this represents 24% of the total number of wastestreams (497 wastestreams).

The total number of pairs of states where waste is targeted to be exchanged is now 45 (reduced from 47 in April). Tennessee and Idaho would receive 96.6% of all waste targeted out-of-state. (Increase of 2.6%)

The volume of waste designated as "No treatment location specified" has decreased slightly from 24% to 23% of the MLLW inventory. The "Oak Ridge Commercial Option" (including commercial disposal) accounts for 81% of this category.

The waste volume targeted to mobile treatment increased to about 10,700 m3 from 3,050 m3 in the 4/95 configuration.

According to current plans, nearly all MTRU waste would be treated in-state to meet WIPP Waste Acceptance Criteria, and ultimately shipped to WIPP for disposal. All HLW would be treated on-site for ultimate shipment to the geological repository.

Estimated costs to treat all mixed waste are now estimated to be $45 Billion in constant 1995 dollars. The cost to treat HLW is 88% of this total.

STPs do not embody a complex-wide integrated schedule for treatment of MLLW.

Conclusion: The STP configuration represents a snapshot in time; DOE's desired configuration will continue to change, possibly significantly as privatization efforts proceed. Such changes are now subject to the revision process as specified in each FFCA order.

Finally, the FFCA model has been look at be all the states with different views. However, each sees it as a positive step in the right direction. Some states may view the process as the main problem solving tool for mixed waste. Others may view it as a means to an end on much larger issues of their DoE complexes. In Tennessee we view as both but with a new twist. The following is a focus on Tennessee's perspective in not only environmental issues and cleanup but economic and jobs.

TENNESSEE PERSPECTIVE

Background

The Tennessee Department of Environment and Conservation's Division of DOE Oversight (the Division) was established in July 1991. Due to its level of involvement with the Public, local government, the Department of Energy (DOE), and their Contractors, the Environmental Protection Agency (EPA), other DOE host states and local, federal and state agencies the Division is located in Oak Ridge, Tennessee. The operation of the Oak Ridge Reservation (ORR) for more than 50 years has resulted in the contamination of over 500 sites on the ORR. Surface and groundwater have been contaminated both on and off the ORR. Most contamination is located on the ORR but a significant amount has entered the off-site environment including the East Fork Poplar Creek which runs through Oak Ridge and the sediments in Watts Bar Reservoir. The contamination includes, but is not limited to, radioactive material, mercury, asbestos, PCBs, and other organic chemicals.

DIVISION AGREEMENTS

The Division was established by agreements with DOE. Its purpose is to implement the Tennessee Oversight Agreement (TOA), the Federal Facility Agreement (FFA) and to reduce cost to Tennessee. Since being established, the Division has added Natural Resource Damage Assessment (NRDA) and Federal Facility Compliance Act (FFCAct) activities to its list of responsibilities. The purpose of the TOA is to assure the citizens of Tennessee that their health, safety and environment in and around the ORR are being protected by DOE through a program of independent monitoring and oversight by the state. The FFA is an agreement between the Tennessee Department of Environment and Conservation (TDEC), the Environmental Protection Agency (EPA) and DOE. The general purpose of the FFA is to ensure that the environmental impacts associated with the past activities on the ORR are thoroughly investigated and the appropriate remedial action is taken to protect public health and the environment. The NRDA is a federal program to allow for the recovery of the cost for environmental

resources lost due to pollution. The FFCAct is a federal requirement for the DOE to provide the State with a plan for the treating of its mixed waste inventories. Mixed wastes are those wastes that are both hazardous and radioactive. The state is responsible for reviewing, approving and assuring the implementation of DOE's plan.

DIVISION RESPONSIBILITIES

The Division's responsibilities include assuring DOE is in compliance with all environmental regulations; monitoring surface water and on-site discharges; biomonitoring; public drinking water supply monitoring; fish and wildlife monitoring; groundwater monitoring; air quality monitoring; and radiological surveillance. Monitoring efforts extend both on and off the ORR. A critical aspect of the radiological monitoring is that it allows the State to inspect, review, and comment on DOE's activities in this area without regulatory authority. DOE is "self regulating" in radiological materials handling, treatment, and disposal. Additional responsibilities involve coordination of activities involving DOE with TDEC, other State, Federal, and Local Agencies, including but not limited to the Governor's Office, the Tennessee Emergency Management Agency (TEMA), Tennessee Wildlife Resource Agency (TWRA), the Environmental Protection Agency (EPA), the Tennessee Valley Authority (TVA), the Corp. of Engineers (COE), and the Oak Ridge Reservation Local Oversight Committee (LOC) and participating in community outreach activities.

CONTRACTS

The Division is funded through a grant from the DOE to TDEC. Grant funds are also used to establish contracts with other state and local agencies. The Division acts as a lead in each of these contracts. TDEC has a contract with the Department of Health's Division of Epidemiology for the analysis of historical releases of toxic and radioactive materials from the ORR in an attempt to reconstruct the potential impact on human health received by the public. A Tennessee Wildlife Resources Agency (TWRA) contract performs independent monitoring of toxic materials in fish flesh, mussel tissue, and in sediments from Melton Hill and Watts Bar Reservoirs. A contract with the Oak Ridge Reservation Local Oversight Committee (LOC) includes citizens and representatives of local governments from the city of Oak Ridge and the counties surrounding the ORR and the Watts Bar Reservoir. It is a forum to increase public awareness and understanding of the issues involving the DOE ORR operations and impacts on surrounding communities. The contract with TEMA provides for the development of an Emergency Response Program. The Division's Federal Facilities Compliance Act activities are funded through the TDEC Division of Solid Waste Management's Mixed Waste Treatment Plan Review Fee Rule.

DOE BUDGET AND PROJECT FUNDING PRIORITY

The Division provides lead for Department review of DOE funds for Environmental Management including prioritization based on risk to human health, risk to environment, regulatory compliance, public concerns, and mission. The range of involvement begins at the Oak Ridge Office base program and extends to the National level with other DOE host States and DOE Headquarters to resolve issues of equity, risk debate or compliance agreements. Budget review is a cross cut effort to assure that the most important problems are addressed first. The Division works with DOE and regulators (TDEC, EPA, Defense Nuclear Facility Safety Board (DNFSB)) to seek compliance in a cost effective manner.

STATE EQUITY

The Division is the contact for DOE, DOE host States, EPA, industry and the public to express needs or recommendations concerning treatment, storage or disposal of waste on the Oak Ridge Reservation. The Toxic Substance Control Act (TSCA) Incinerator located at K-25 is used to incinerate mixed waste and PCB's. It is permitted by TDEC (air, water and hazardous waste) and EPA. TSCA is the only permitted mixed waste incinerator in the country and is the preferred treatment facility for numerous out of state DOE facilities. The Oak Ridge Reservation is limited
in disposal options and disposal of most TSCA residuals on-site is not an alternative. The Division is working with DOE and other States to resolve disposal issues.

MEMORANDUM OF UNDERSTANDING (MOU)

The Division is an integrated program with memoranda of understanding establishing the roles and responsibilities with the environmental regulatory Divisions of the Department. The purpose being to reduce state/federal cost, minimize regulatory overlap and facilitate coordination.

The Division serves as the state lead for the Watts Bar Reservoir Interagency Working Group MOU between DOE, TVA, COE, EPA, AND TDEC. The propose being to review and expedite processing of public and private sector applications for dredging and other sediment disturbance activities impacted by DOE release of mercury and radioactive materials into the Clinch River.

The Division serves as lead for the DOE, TDEC and Department of Interior MOU for Oak Ridge Reservation Natural Resource Trustees. The purpose being to establish a Natural Resource Trustee Council to consolidate NRDA efforts, share information, reduce cost and seek uniformity in damage assessment relating to the damage or loss of natural resource to Tennessee. The Commissioner of Environment and Conservation is currently the Natural Resource Trustee for the State.

REUSE AND ECONOMIC DEVELOPMENT

The Division interacts with the Department of Economic and Community Development, DOE, local governments, Chambers of Commerce, East Tennessee Economic Council, Community Reuse Organization of East Tennessee, the Oak Ridge Waste Management Association and private groups or individuals as State focal point for DOE and private sector economic development plans to utilize buildings, surplus materials and/or infrastructure As missions change and the complex is downsized the Division shares with EPA and DOE in the important decisions relating to, site remediation, decontamination activities, environmental regulation, and funding schedules. As the Environmental Remediation industry continues to expand as one of the fastest growth industries how can the state tap into the wealth of jobs and companies involved in this industry? Can partnerships be developed with DoE and EPA for massive privatization of cleanup programs to grow this industry in East Tennessee? Is there a benefit for the Oak Ridge area to be known as the silicon valley of environmental remediation. Note this would not be storage but recycling and returned to the originator.

BENEFIT

The measurement of benefits relates to the primary purpose the agreements which is to ensure compliance with applicable laws and to assure the citizens of Tennessee that their health, safety and environment are being protected through existing programs and new commitments by DOE and through a program of independent monitoring and oversight by the State.

Substantial cost savings well in excess of the State grants have been achieved by DOE through the implementation of recommendations or decisions made by the State. Substantial success has been made by DOE in environmental management through problem identification and cooperative technical assistance in problem solving without enforcement. Substantial reduction in violations and enforcement actions resulting in orders and fines by regulator have resulted from increased oversight. Coordinated review of monitoring activities have identified off-site contamination problems, risk evaluation and at the same time success has been made toward overall reductions in monitoring activities and cost. A notable benefit to the Tennessee taxpayer and the Department has been the DOE workload reduction on central or field office regulatory programs. Reductions in DOE activities allowed for increased regulatory program staff response to industrial, commercial, public and private sectors concerns.

20-3
RISK-BASED RCRA REGULATIONS - FACT OR FANTASY
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ABSTRACT
For almost a decade, the Department of Energy (DOE), like other members of the regulated community, has worked steadfastly to attain compliance with the Resource Conservation and Recovery Act (RCRA). And, like the private sector, DOE largely has been able to manage its hazardous waste in accordance with RCRA. However, DOE waste management activities are confounded and complicated by the presence of radioactivity. In fact, most DOE waste is either radioactive or radioactively contaminated hazardous waste commonly known as "mixed waste". Treatment and disposal capacity for mixed waste has been and continues to be extremely limited because mixed waste must be managed in accordance with the requirements of the Atomic Energy Act (AEA) and RCRA. Although early efforts to demonstrate inconsistencies between RCRA and the AEA failed to identify any, the absence of risk-based requirements or non-regulatory thresholds for both radioactive and hazardous waste forces wastes posing little to no risk or risks dominated by either the radioactive or hazardous waste component of mixed waste to be managed in strict compliance with the full gamut of RCRA and the AEA requirements. The lack of risk-based mixed waste regulations drives up the waste management costs without a concomitant increase in protection of human health and the environment. The Environmental Protection Agency (EPA) intends to issue a final hazardous waste identification rule in late 1996, which for the first time, will establish concentration based exit levels for hazardous waste. This risk-based regulation would allow hazardous wastes below the identified concentrations to be disposed in Subtitle D or other non RCRA regulated facilities. Mixed waste containing hazardous waste below the

exit concentrations would be disposed as radioactive waste. Development of risk based waste management requirements should consider all hazards posed by the waste management requirements necessary to safeguard against those hazards. In the case of mixed waste, waste management requirements should consider the requirements necessary to safeguard the radioactive component as well as the hazardous component in establishing not only exit criteria under HWIR but risk based regulations that enhance protectiveness rather than merely complicate compliance.

Risk based approaches to managing mixed waste can and should be considered in the HWIR rule. Contingent management, dominant hazard, and technology driven regulations are waste management options that would allow mixed waste to be managed consistent with the risk posed by the waste. Efforts to foster consideration these options, their technical basis and justification, and implementation scenarios are being discussed.

For almost a decade, the Department of Energy (DOE), like other members of the regulated community, has worked continually to attain compliance with the Resource Conservation and Recovery Act (RCRA). DOE has largely been able to manage its hazardous waste in accordance with RCRA. However, DOE also manages large quantities of mixed wastea, and the presence of radioactivity complicates the waste management activities. Treatment and disposal capacity for mixed waste has been extremely limited because the mixed waste must be managed in accordance with the requirements of both the Atomic Energy Act (AEA) and RCRA.

Under the authority of the AEA for radioactive waste, DOE has relied on the performance of a system expressed as a radiation exposure limit and has left the decision regulating the appropriate engineering design to the regulated community. DOE also reduces radiation exposure to personnel and workers by applying the "As Low As Reasonably Achievable" (ALARA) principle with radiation protection guidelines. These guidelines generally state that worker exposure to radiological hazards should be maintained at a minimum and assure that exposures are justified by benefits produced by the activity.

In contrast to waste management principles under the AEA, RCRA contains numerous prescriptive design requirements that give the regulated community less flexibility in designing a waste management system on the basis of performance or risk. One example where the RCRA framework does not account for associated risks is listed hazardous waste. Under RCRA, solid wastes are designated as listed waste if they are used in a particular way, such as solvents, or if they come from a particular process, such as sludge from an electroplating process. In order to prevent "dilution as the solution" to managing that particular waste stream, RCRA requires that once a waste is a "listed" hazardous waste, it remains a listed hazardous waste regardless of the hazardous constituent concentration. The RCRA regulatory framework has other provisions including: 1) the "mixture rule" which states if a solid waste is mixed with a listed hazardous waste then the entire mixture must be managed as a listed hazardous, 2) the "derived from" rule which states that wastes generated from the management of listed hazardous waste are themselves listed hazardous wastes, and 3) the "contained in" policy that requires environmental media (i.e., groundwater) contaminated with a listed waste to be managed as a listed waste. Currently, the only option available to the regulated community is to have a waste "delisted," which is a relatively expensive and time-consuming process, especially when applied

to mixed wastes. This prescriptive approach ignores the idea that protection of the radioactive component of mixed waste provides varying degrees of protection for the hazardous component, depending on the nature of the mixed waste.

We recognize, however, that Environmental Protection Agency (EPA) has made attempts at establishing a risk-based regulatory framework. On May 20, 1992, EPA published a Notice of Proposed Rulemaking (NPRM) regarding Identification and Listing of Hazardous waste [57 FR 21450]. In the preamble EPA indicated that it would "begin tailoring the scope of its hazardous waste program to reflect how wastes are actually managed, rather then how they might be managed under a worst-case analysis," after the D.C. Circuit Court remanded EPA's 1980 mixture and derived from rules. The May 1992 proposed rule allowed low concentrations of hazardous constituents to no longer be regulated as hazardous waste and managed as sanitary waste. Mixed waste, with low concentrations of hazardous waste would likewise be managed as low-level radioactive waste. On October 30, 1992 [57 FR 49280], however, EPA withdrew the May 20, 1992 proposed rule due to the opposition it received from the public, Congress and the States.

Since EPA withdrew the proposed rule they have been working with the States and representatives from industry and environmental groups in the development of the Hazardous Waste Identification Rule (HWIR). The HWIR committee disbanded in September 1994 after making its recommendations to EPA. Essentially EPA developed two rules under HWIR, one for process waste which was proposed in November 1995 which EPA intends to finalize in late 1996, and a second for contaminated media which is expected to be proposed in early 1996.

The HWIR rule for process waste, for the first time, will establish concentration based exit levels for hazardous waste. This regulation establishes concentrations for various listed constituents, which will allow hazardous waste to be disposed in a Subtitle D or other nonRCRA regulated facility, based on a risk of 10-6 (for cancer-causing constituents) and hazard quotient (HQ) not exceeding 1 (for noncancer risks). EPA has determined, using a multipathway risk assessment methodology, that hazardous waste containing low hazardous constituent concentrations will not present significant risks to human health and the environment if managed in land disposal facilities not regulated under Subtitle C requirements. EPA analyzed over 400 hazardous constituents for exit levels. EPA's analysis assesses risks to humans and ecological risks from two categories of wastes, wastewaters and nonwastewaters. The analysis also compared risks posed by management of these wastes in five different types of waste units.

Since early 1995, DOE has identified and evaluated several areas of RCRA regulations that if modified, could result in substantial cost savings or cost avoidance, yet pose no adverse impacts to human health and the environment. DOE's focus is on risk-based regulatory changes related to the mixed waste management. DOE has developed the following specific regulatory reform proposals for consideration by EPA:

- Immobilized Mixed Waste Debris
- Vitrified Waste
- Contingent Management

The following provides short discussions on these regulatory reform proposals.

DISCUSSIONS OF DOE'S PROPOSALS

Immobilized Debris Proposal

Debris is defined as "solid material exceeding 60 mm (2.5 inch) particle size that is: (1) a manufactured object; or (2) plant or animal matter; or (3) natural geologic material (e.g., cobbles and boulders), except that any material for which a specific treatment standard is provided in Subpart D, part 268, is not debris" (40 CFR 268.2). Debris are routinely generated from activities such as construction, maintenance, routine operations, closure of facilities, and environmental remediation. Examples of debris include pieces of concrete, metal (e.g., piping), wood, plastics, used personal protective equipment, and other similar wastes. The types of debris normally found at DOE sites do not contain high concentrations of hazardous contaminants.

As part of Phase I Land Disposal Restrictions (LDR) rule, EPA promulgated the Final Rule on Hazardous Debris (Debris Rule) in August 1992. This rule allows hazardous debris treated by extraction or destruction technology to exit RCRA Subtitle C control provided that treated debris does not exhibit a characteristic of hazardous waste (57 FR 37222, August 18, 1992).

At the time the Debris Rule was enacted, EPA chose not to allow debris treated with an immobilization technology to exit Subtitle C control. The rationale for this was that there was not sufficient data available to demonstrate that, absent Subtitle C management, contaminants would not migrate from immobilized debris at levels that could pose a hazard to human health and the environment (57 FR 37240). However, EPA invited the regulated community to submit data on immobilization of debris and requested comments on whether immobilized debris should exit from Subtitle C regulations as part of the proposed Phase II LDR rulemaking (58 FR 48144, September 14, 1993). EPA indicated that the HWIR may be the appropriate rulemaking for this issue. As part of its final Phase II rulemaking, EPA stated that if the technical data were submitted, the Agency would exclude immobilized debris from Subtitle C control (59 FR 48012, September 19, 1994).

In Winter 1995, DOE developed technical data package supporting the proposal that mixed waste debris treated by immobilization, followed by disposal in a low-level waste (LLW) facility is protective of human health and the environment and therefore, should be allowed to exit Subtitle C controls. The basis for this proposal is the combination of the integrity of the encapsulated debris waste form, coupled with the protectiveness of a LLW disposal facility. This technical data package was submitted to EPA in July 1995. On October 20, 1995, DOE supplemented the July 1995 report to EPA with a report entitled "Performance Evaluation for RCRA Toxic Metal Disposal in DOE Low-Level Radioactive Waste Disposal Facilities."

Technologies for immobilization/encapsulation of solids are classified depending on whether the encapsulant is interspersed with the waste (microencapsulation) or surrounds the waste (macroencapsulation). EPA defines macroencapsulation as "the application of surface coating materials such as polymeric organics (e.g., resins and plastics)..." to "...completely encapsulate the debris" (57 FR 37235, August 18, 1992). Furthermore, the encapsulating material must be resistant to (i.e., unreactive with) the debris it encapsulates and other materials it may come in contact with after disposal (i.e., leachate, other wastes and microbes). Microencapsulation is defined as stabilization of debris with Portland cement and lime/pozzolans (fly ash and cement kiln dust) and may

include such additives as iron salts, silicates and clays (57 FR 37235). The resulting encapsulated waste form can be a free-standing monolith, depending on the size of the waste unit, or small pellets (60-mm size limit).

EPA currently recognizes polymeric organic materials as acceptable macroencapsulating agents and only Portland cement and lime/pozzolans as acceptable microencapsulants. Several other encapsulating agents including hydraulic cement, sulfur polymer cement, polyethylene, phosphate ceramics, epoxies, urea formaldehyde polymer and asphalt, have been developed and tested. Because the performance of some of these materials is comparable or superior to that of the accepted microencapsulants, these alternate encapsulants (sulfur polymer cement, polyethylene, phosphate ceramics) have been included in DOE's proposal. For each of the proposed encapsulating materials DOE assembled data on waste form leachability and/or permeability, biodegradation, radiation stability, and long-term environmental stability.

To assure the mixed debris treated by immobilization and placed in a low-level waste disposal facility is sufficiently protective of human health and the environment, DOE proposed that the final waste form meet or exceed an established performance criteria. Because EPA's performance standard for microencapsulated waste is that the "leachability of hazardous contaminants must be reduced" (57 FR 37235) various tests have to be performed on the final waste form to meet that criteria. DOE proposes that testing of immobilized debris be conducted in two tiers. Tier one consists of the following:

Microencapsulated debris - Toxicity Characteristic Leaching Procedure (TCLP) as per EPA Model 1311, or the Synthetic Precipitation Leaching Procedure (SPLP) as per EPA Model 1312.

Macroencapsulated debris - A modified TCLP or SPLP, possibly using an encapsulated coupon of the debris, and waste form integrity testing via a non-destructive test such as real-time radiology, ultrasound, or x-ray. The standard leachability test method is not appropriate for macroencapsulated debris because it would require breaking the protective encapsulant layer and allow the leaching solution to be in contact directly with debris. This is contradictory to the intention of the macroencapsulation process, therefore, the integrity of the final waste form should be verified using non-destructive methods.

Tier two tests could include the following: a compressive strength test, non-destructive test, long-term immersion in water, radiation stability, biodegradation, freeze-thaw cycling, and wet-dry cycling. One or more of the tier two tests would be performed on the waste, based on the tests which are appropriate for a particulate disposal facility location, to demonstrate the integrity of the final encapsulated waste formb. DOE recommended that these tests be done initially as proof of process tests and then periodically repeated as quality assurance checks. DOE sites would work with their respective regulator to decide which tests are appropriate for the treated mixed waste debris after considering the type of encapsulation and the characteristics (e.g., climate, depth to groundwater, etc.) of a disposal site.

Both, RCRA Subtitle C and DOE LLW disposal facility requirements share a common goal of maximizing the protection of human health and the environment from the hazards contained in each type of facility. These two facilities also have several common elements, including limiting the amount of free liquids that may be present, requiring groundwater

monitoring and specifying a period of active institutional control after site closure. The primary difference between the two types of facilities is the RCRA requirement for waste unit double liners and a leachate collection system. This difference is the result of two different philosophical approaches to waste management that have evolved due to fundamental differences in the nature of the hazards associated with RCRA and LLW. The EPA waste management philosophy under RCRA is to treat hazardous constituents to safe levels before land disposal. Radioactivity, on the other hand, cannot be treated to safe levels (excluding transmutation), but can only be eliminated as a result of natural decay. This has led to the waste management philosophy of immobilizing/isolating radioactive waste from the public and sensitive environments until natural decay renders the residuals safe. The extent of immobilization/isolation varies from hundreds to thousands of years, depending on the half-lives of the radionuclides and the curie content of the waste. However, unlike their radioactive counterparts which eventually decay in innocuous levels, the RCRA toxic metals are persistent in the environment. This fact lessens the importance of the effects of a disposal facility's engineered barriers and increases the importance of the site geology.

While EPA has a prescriptive approach to disposal of RCRA hazardous waste, DOE takes a performance-based approach to radioactive waste disposal which allows the disposal site to design and operate the facility as appropriate to achieve the required performance standards. Depending upon characteristics of the disposal site (i.e., geology, meteorology, etc.), designs can range from shallow land burial to containment in above or below-ground concrete enclosures. Facility acceptability is determined via a performance assessment that verifies the ability of the facility as designed and operated to meet the performance objectives prescribed in DOE Order 5820.2A.

In DOE's report, "Performance Evaluation for RCRA Toxic Metal Disposal in DOE Low-Level Radioactive Waste Disposal Facilities", a risk-based analysis evaluated the environmental transport of RCRA toxic metals from six DOE LLW disposal sites. The analysis focuses on the toxic metal component of the mixed waste debris (the principal contaminants in DOE's mixed LLW debris) and the groundwater contaminant pathway, because it is the dominant transport pathway for human exposure from land disposal facilities. The analysis estimates permissible leachate concentrations of toxic metals by using Maximum Contaminant Levels (MCL) concentration values in groundwater at a receptor point along the performance boundary (100 m from the disposal facility boundary), and attenuation factors associated with site-specific conditions. The report concludes that arid DOE LLW sites appear to provide a greater degree of protection of human health and the environment than humid DOE LLW sites based on higher attenuation and longer contaminant travel times. However, even at relatively humid sites most RCRA toxic metals (except for arsenic and selenium) are immobile in the subsurface environment.

In summary, the DOE's proposal is based on the integrity and stability of the immobilized mixed waste debris treated in accordance with 40 CFR 268.45. Protection of human health and the environment is achieved via the combination of this treatment and disposal at LLW disposal facilities regulated under the requirements of the AEA. This proposal allows for a reasonable mixed waste management approach without compromising

protectiveness of human health and the environment, and at the same time would result in substantial cost savings to the regulated community.

Vitrified Waste Form

Vitrification is the process of converting materials into a glass-like substance, typically through a thermal process. Vitrification thermally destroys organic contaminants and stabilizes inorganics and metals by incorporating them into the glass structure. Vitrification has four major advantages over other methods of waste treatment. The primary advantage of vitrification is that it produces durable waste glass. With proper formulation, this waste glass performs exceptionally well in leach tests. The second major advantage of vitrification is the ability of the waste glass to incorporate a wide variety of contaminants and accompanying feed material in its structure, without a significant decrease in quality of the final waste form. The third advantage is that the vitrification process can accommodate both organic and inorganic contaminants of various amounts. Lastly, vitrification typically results in significant volume reductions of waste material. One significant disadvantage of the vitrification technology is the high cost.

Vitrification is a desirable treatment option for long lived radionuclides, despite the cost, because the vitrified waste forms will resist degradation for the thousands of years necessary for radioactive decay to lessen the threat to human health and the environment. During this decay period, the metals and inorganics are chemically bonded in the glass matrix. Due to these features, EPA has declared vitrification to be the specified treatment technology for mixed high-level waste (55 FR 22627, June 1, 1990). DOE will vitrify high-level waste and may propose to vitrify certain other mixed low-level waste where it is economically feasible to do so. The mixed low-level waste candidates for vitrification are predominantly sludges, slurries, and metal/metal oxides.

On October 20, 1995, DOE submitted to EPA a technical data package to support its proposal to exempt vitrified mixed waste forms from RCRA Subtitle C regulations. The proposal would allow vitrified mixed waste forms which have been treated under an environmental permitting process to exit from RCRA hazardous waste regulations based upon inherent destruction and immobilization capabilities of the technology. The technical data package also includes DOE's proposal of an alternative sampling and analysis strategy for certain highly radioactive mixed waste forms. The proposed sampling strategy considers the radiological hazard associated with testing of the final product. For wastes with low radiation hazard, sampling and analysis is performed on the final product. However, sampling and analysis of highly radioactive wastes is performed on surrogate vitrified wastes that are chemically equivalent to the actual waste. DOE believes that this alternate testing strategy provides results with an equal level of confidence as sampling and analysis of the final waste forms, as suggested by 40 CFR 261.36.

DOE proposes performance characteristics for vitrified waste forms with glass compositions that meet or exceed the performance standards of the Environmental Assessment (EA) glass developed as the standard for the immobilization of high-level waste. The EA glass is a borosilicate glass which has been selected as the preferred waste form by the United States and other countries involved with high-level waste management. The EA glass was specifically formulated to retard the release of radioactive components in the waste, but has also been shown to retard the release of non-radioactive components.

Vitrification requires a process control protocol for key operating parameters in order to yield a glass product consistently falling within a pre-defined acceptable performance envelope. The process control is defined by performing treatability studies on either the actual radioactive waste or an appropriate surrogate. The treatability studies provide information on glass formulation process and the balance of operating variables, such as waste loading and viscosity, while ensuring the durability of the final waste form. Once the parameter values, which produce a durable glass are determined, they are used to define the Process Control Program. Therefore, DOE's vitrification proposal is contingent on the required use of the Process Control Program that ensures desired durability of the end product. The Process Control Program not only defines the process and ensures final product performance, but is required to demonstrate compliance with existing federal and state environmental regulations associated with the operation of a mixed waste treatment facility. Once this Process Control Program is defined and accepted as part of an environmental permitting process, the inherent properties of the final glass waste form ensure protection of human health and the environment.

The Process Control Program requirements also include detail description of the sampling and analysis program. To ensure the glass' durability, DOE proposes to monitor the leach rates of several of the most leachable glass components. Two forms of leach tests, the Product Consistency Test (PCT) (ASTM-C1285-94) and TCLP have been proposed. The PCT test was developed for evaluating the performance of high-level waste glass and its durability as it relates to the release of radioactive components. Testing requirements for organic constituents identified in the vitrified waste stream are eliminated because organic wastes are thermally destroyed by the vitrification process, which typically operate in excess of 1000oC.

In summary, the DOE's proposal is based on the inherent characteristics of the vitrification process and stability of the waste form. Upon meeting specific Process Control Program requirements, the glass product would be exempt from RCRA Subtitle C control. Because the mixed waste vitrification process will be regulated under 40 CFR 264 Subpart X, the regulatory permitting agencies will be involved in the approval of the Process Control Program. This proposal provides a means to facilitate the use of a superior technology for responsible waste management while reducing costs, and maintaining full regulatory authority until an acceptable vitrified waste form is produced.

Contingent Management

While the proposals discussed above deal with specific waste forms, we believe they constitute examples of a broader concept which EPA has been considering. Both the 1992 PRM and the HWIR proposal discuss options whereby wastes that would otherwise be regulated under RCRA could exit the control of Subtitle C contingent upon subsequent management that would protect human health and the environment. In the most recent proposal, EPA detailed potential approaches to allow wastes to exit the Subtitle C system. These approaches fall into the following three broad categories:

1) Conditional exemptions based on unit type without additional management controls. Under this approach EPA discusses two possible options: a) constituent specific exit levels could be based on the second riskiest management scenario if there were a mechanism to ensure that the

riskiest management scenario would not occur; or b) set separate exit levels for each management scenario.
2) Setting conditional exemption levels for unit types with additional controls. Under this approach EPA would consider establishing less conservative exit levels at units applying specific design or operating controls to further minimize risk.
3) Conditional exit levels in states with qualified industrial non-hazardous waste programs. EPA discusses several options which would exempt some waste form Subtitle C and rely on state imposed requirements to ensure protectiveness.

DOE is currently investigating a variety of potential applications of the contingent management approach. As detailed earlier, these proposals may center on a specific waste stream or final waste form coupled with disposal. Proper management could also be demonstrated by evaluating specific waste disposal options alone. This could include a generic process, whereby Subtitle C exit levels are established contingent upon waste disposal at an AEA compliant site rather than an uncontrolled site; or a site-specific process whereby Subtitle C exit levels are established by considering detailed site-specific information and a multi-pathway exposure model. Regardless of the method, the validity of any contingent management scenario would be based on the ability to meet the goals of 10-6 and HQ=1.

Benefits Associated with DOE's Proposals

The following provides a discussion of the waste volumes that could potentially be impacted and the associated potential cost savings resulting from each of the three proposals. The volume estimates were obtained from DOE's Mixed Waste Inventory Report required by the Federal Facility Compliance Act of 1992.

Approximately 27,000 m3 of DOE's current inventory of mixed waste is debris and is a candidate for immobilization. This waste volume includes both heterogeneous and inorganic debris and accounts for roughly 4 percent of DOE's current mixed waste inventory. If the EPA, and subsequently the states, allow immobilized mixed waste debris to be disposed as low-level mixed waste, DOE could potentially save several million dollars, and possibly up to $1 billion, on disposal costs. This cost savings is based on comparing unit costs of disposing in a low-level waste landfill to a mixed waste landfill.

Approximately 360,000 m3 of DOE's current mixed waste inventory is high-level waste that can be vitrified. This accounts for roughly 60 percent of DOE's current mixed waste inventory. If EPA and the States except and implement the vitrification proposal, DOE could save over $1.5 to 1.6 billion on high-level waste disposal. This estimate is based on the assumption that a RCRA permit would not be required for the geologic repository receiving the vitrified high-level waste. DOE also has approximately 35,000 m3 of mixed low level waste that could be a candidate for vitrification. These waste streams include glass debris, wastewater treatment sludges, soil, acidic wastewaters, inorganic particulates, organic debris, ash, inorganic heterogeneous materials, and predominately inorganic materials. If EPA and the states accept and implement the vitrification proposal, DOE could save between $50 million to $100 million on disposal costs. This cost savings is based on comparing unit costs of disposing in a low-level waste landfill to a mixed waste landfill.

Because some of the individual waste streams may be covered by more than one proposal, such as debris waste, the totaling of the proposals may show a doubled cost savings. Therefore, the waste volumes and cost estimates for these proposals stand alone and cannot be combined to determine a total cost savings.

Future Plans and Proposals

DOE plans to continue the dialogue with EPA and provide information to support these and future proposals. DOE is optimistic that the basic tenets of the proposals will be accepted by the EPA and the States because they provide reasonable regulatory alternatives and substantial cost savings to the DOE without compromising protection of human health and the environment. If other viable alternatives for the regulatory scheme related to management of mixed wastes are identified and the technical data packages can be assembled, DOE will provide proposals to EPA for consideration.

An opportunity for continuation of the regulatory discussions with EPA will surface again in early 1996 when EPA proposes the second part of the HWIR rule for contaminated media generated from remediation of hazardous waste sites. The rule will exempt certain lower risk contaminated media from the traditional "prevention oriented" RCRA regulations and will set treatment standards for higher risk media that reflect the inherent differences between contaminated media (e.g., soils, groundwater) and newly generated wastes. EPA will set a "bright line" above which the wastes must meet full RCRA Subtitle C standards, and below which the requirements will be site-specific.

20-4

STREAMLINING THE MIXED WASTE TREATMENT CONFIGURATION - RESULTS OF THE DOE OPTIONS ANALYSIS TEAM

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ABSTRACT

The Federal Facility Compliance Act required that U.S. Department of Energy sites that generate and store mixed waste develop Site Treatment Plans describing how they intend to develop treatment capacities and technologies to treat mixed waste. Site Treatment Plans were developed by individual sites and eventually consolidated. Once combined, the resulting national configuration of treatment capabilities showed many redundancies and inefficiencies. The DOE Options Analysis Team (OAT) was chartered to analyze the national configuration and develop a "wise configuration" of treatment facilities. The OAT was able to optimize the use of existing and new facilities, minimize transportation, and minimize overall cost and schedule for completion of treatment, while addressing state equity and risk concerns. This paper presents the OAT approach, how configuration analyses were performed, how states' input was incorporated, how budget constraints were resolved, and the overall outcome of the OAT process.

INTRODUCTION

The Federal Facility Compliance Act (FFCAct) of 1992 added Section 3021(b) to the Resource Conservation and Recovery Act (RCRA) and required federal facilities which generate mixed waste (radioactive and hazardous waste) to develop Site Treatment Plans (STP) for treating currently

stored and future generated mixed waste. The FFCAct provided an unprecedented opportunity for the U.S. Department of Energy (DOE) to work with its regulators to resolve a long-standing issue - how to treat large amounts of mixed waste now being stored or generated at DOE sites. DOE sites completed these plans in three phases: conceptual, draft, and proposed. 49 site Draft STPs were completed in September 1994 which identified preferred treatment options and the selection processes used. Proposed STPs were completed in April 1995 and identified schedules for treatment options selected. Most sites entered into consent orders with their respective states by October 6, 1995, the date marking the end of sovereign immunity of the federal government from fines and penalties associated with violation of the storage prohibitions of RCRA. The individual site consent orders specify implementation requirements for the STPs.

Because the Draft STPs were prepared by the sites using a "bottoms up" approach, the resulting national treatment configuration contained many redundancies and inefficiencies. Thomas P. Grumbly, DOE Assistant Secretary for Environmental Management, requested that the STP Task Force develop a "wise configuration" by minimizing redundancy, reducing costs, yet fulfilling the needs of the complex by providing timely and efficient waste management. In developing the Proposed STPs, an assessment was performed to determine what accommodations were necessary to blend the "bottoms-up" Draft STPs into a more sensible national configuration of treatment systems. To facilitate this assessment, DOE established the Options Analysis Team (OAT), comprised of site representatives and members of the DOE Headquarters' FFCAct Task Force. The OAT coordinated their efforts with the states through the National Governors' Association (NGA) to ensure the national mixed waste configuration reflects both the states' and DOE's concerns. As part of this evaluation, the impacts of implementing the emerging DSTP configuration, as well as alternative configurations, were evaluated. The resulting national configuration is DOE's best attempt to balance competing DOE and stakeholder interests and decreasing government resources.

OAT CHARTER AND MEMBERSHIP

The OAT was formed to provide a diverse yet representative group that could overhaul the Draft STP configuration while reflecting site capabilities, needs, and regulator interests. This team was formed in June 1994 and was comprised of DOE site representatives and members of the FFCAct Task Force. Technical support was provided by site technical experts and technology development representatives. The site representatives offered the sites' perspectives and brought individual state and site issues to light, while the Headquarter's representatives offered insight from their dealings with policy issues and the "big picture." The technology development Mixed Waste Focus Area supported the OAT, and their goals were to identify technology development needs, assess benefits of using emerging technologies, and scope technology development programs that are responsive to sites' compliance requirements.

The OAT worked closely with the states through the NGA, which opened up communication at a key time in the development of individual STPs. The OAT and NGA met at least quarterly through development of the STPs, OAT process, and consent orders. State representatives were given the opportunity to interact with DOE as a whole -- both individual site representatives and policy makers from DOE Headquarters. It was also

valuable for the states to confer with one another about DOE site issues and approaches.

The OAT began analyzing the Draft STP configurations in August 1994 and completed the majority of the work by January 1995. The OAT, however, continued to work with the sites to refine treatment options through the completion of STPs (October 1996). The OAT dealt with the treatment configuration for mixed low-level waste (MLLW) only. Treatment for high-level waste and mixed transuranic waste are handled through other efforts. The results of the initial OAT analysis was shared with each of the sites and the state regulators, as well as DOE management. The following months were spent responding to state requests for additional analysis, incorporating ongoing site analysis, and responding to comments. Open communication and involvement by the states, as well as their influencing the outcome of the OAT process, were fundamental to the success of the overall STP process. This success is demonstrated by the achievement of signed consent orders for most sites by the October 6, 1995 target date.

TREATMENT CONFIGURATION EVALUATION

The integration of the 49 Draft STP configurations formed the baseline for the OAT analyses. A database was developed with key waste stream and treatment facility data and was the key tool in analyzing the STP configuration. Waste streams were divided by the types of treatment needed to show the baseline configuration and allow comparisons. Several different configurations were then analyzed based on criteria requested by the states, and the OAT selected their preferred configuration to present to the states and DOE management.

OAT Database

The OAT database tracks key information about STP facilities and waste streams. Facilities were tracked by unique identifying numbers which included a site code prefix and an ID number. Waste streams were also tracked by unique identification numbers used in the Mixed Waste Inventory Report, a database of mixed waste information used to develop the STPs. Waste streams were categorized by waste type (mixed low-level waste, mixed transuranic waste) and waste matrix (a code identifying physical and chemical characteristics of the waste). The resulting database was used to sort the data on various types of information and allowed scrutiny of the configuration against evaluation criteria defined by the states and DOE. Both mixed low-level waste and mixed transuranic waste were included in the database; however, only the mixed low-level waste configuration was analyzed. Mixed transuranic waste treatment options are being reviewed under a separate effort.

Treatment Types

Initial sorts of waste streams were performed using the treatment type required and the treatment option (specific facility and location). These sorts allowed analysis of the Draft STP configuration by showing which facilities were being under-utilized, over-utilized, or were duplicated. The breakdown of treatment types is as follows:

- Organic Destruction
- Deactivation
- Non-Aqueous Neutralization
- Stabilization
- Mercury Amalgamation
- Inorganic Debris Treatment
- Mercury Separation

Soil Washing
Alkali Metals Treatment
Waste Water Treatment

These treatment types generally describe the mixed waste treatment needs across the DOE complex.

Treatment Configurations Analyzed

With input from the NGA, the OAT evaluated the following MLLW treatment configurations for each treatment type. In all configurations, the OAT attempted to use existing treatment systems to the maximum extent possible.

DSTP Configuration - site-preferred treatment options presented to the States in the Draft STPs. This configuration was used as the baseline for developing the other configuration scenarios. The Draft STP is a "bottoms-up" configuration reflecting the States' preference for on-site treatment.

Quick-Start Configuration - a mixed waste treatment configuration that emphasizes use of existing treatment systems, including DOE and commercial facilities, and mobile treatment units.

Centralized Treatment Type Configuration - a regionalized configuration based on treatment type and waste volume. This configuration emphasizes consolidation of a large number of treatment systems located in geographically strategic areas, based on waste volume and sites generating the wastes.

The Quick Start and Centralized Treatment Type configurations were created to emphasize different key State and DOE concerns (relating primarily to schedule and cost) and as such, to serve as building blocks for creating the OAT Proposed configuration. To meet this objective, OAT members decided it was important to have flexibility in creating these two configurations for the individual treatment types. Because of this, the consideration of existing facilities in these two configurations may seem inconsistent among sites.

OAT Proposed Configuration - the mixed waste treatment configuration that attempts to take into account the advantages and disadvantages of the three original scenarios (DSTP, Quick Start, and Centralized). This configuration incorporated the comments from the sites and was presented to the States and DOE management as a basis for further discussion.

Modified Centralized Treatment Configuration - a revised version of the Centralized Treatment Type configuration (above). This configuration extends the logic of the Centralized Treatment Type configuration in order to provide a least-cost reference point. This configuration was developed after completion of the OAT Proposed configuration, and analysis of this configuration supported ongoing cost evaluations described later.

In addition to the above configurations, three other scenarios were added later and evaluated:

Nearest-Site Scenario - a revised version of the DSTP configuration in which wastes proposed for off-site treatment are redirected to the nearest off-site treatment system. This configuration was created at the States' request to evaluate the impact of a "minimal shipment" option on interstate transportation of mixed waste.

Compact Scenario - a configuration that would restrict waste shipments to established low-level waste compact regions. In most cases, OAT members recognized that this configuration was not feasible. Therefore, the OAT did not pursue further evaluation of this scenario.

Alpha-Contaminated Waste Scenario - the treatment of alpha-contaminated waste (10-100 nanocuries per gram transuranic alpha) was originally intended to be evaluated as a separate scenario. However, it proved very difficult to completely separate it from the rest of the MLLW and as such, was considered along with MLLW in all configurations evaluated.

Site OAT members were each assigned treatment types for configuration development and evaluation. Members used the baseline database extract reports showing all new and existing facilities which could treat each respective treatment type and all waste streams proposed to be sent to each facility. The site representatives then developed each of the above configurations by proposing which facilities could most likely be deleted or consolidated with others, and which waste streams could be treated commercially.

Six specific evaluation areas influenced the configuration development process. The evaluation areas are described below. Considering these evaluation areas in developing the configurations provided consistency and enhanced comparison and evaluation.

Evaluation Areas

Mixed waste treatment configurations were evaluated based on their relative ranking (high, medium, and low) with respect to the following six evaluation areas:

Use of existing treatment facilities: evaluation based on number of existing treatment systems to be used under a given configuration and volume of waste proposed for treatment in them.

Number of new treatment facilities: evaluation based on number of new treatment systems proposed under a given configuration. This evaluation area, in addition to addressing cost, also addresses construction risk. It was assumed that new treatment facilities would be designed or re-scoped to handle any waste stream changes proposed by the OAT.

Volume of waste transported across state lines: evaluation based on volume of waste proposed for shipment to another state under a given configuration. This evaluation area addresses equity issues, as well as transportation risk.

Cost: evaluation based on best available cost estimates and results from a cost estimating model under development at the time. This model provided a life cycle cost estimate based on key waste management activities, such as characterization, treatment, storage, disposal, shipping, etc. The cost model had not been validated by DOE at the time of this analysis; therefore, the results were only used for preliminary relative cost comparisons.

Time required to complete treatment: evaluation based on OAT members' experience in scheduling new facility design, construction, permitting, etc. and on assumption that commercial and mobile treatments can be implemented faster.

Use of alternative treatment technologies: evaluation based on the extent to which the use of future alternatives is feasible under a given configuration. Since the OAT did not select specific technologies for new facilities, all configurations were deemed equal in this regard. The maximum opportunity for using alternative technologies will be in the construction of new treatment facilities. All configurations analyzed assumed some new treatment facilities would be built.

For each configuration, representatives rated each evaluation area as high, medium, or low relative to the other configurations. The OAT met in October 1994 to review the OAT members' treatment type analysis and

further refine this process. The goal of this meeting was to develop the national OAT Proposed configuration. To do this, the team members discussed the results of their analyses, and it became clear that some alternatives were better than others. It was decided that the OAT Proposed configuration should combine the successful attributes of the different scenarios. The team reviewed each treatment type and which facilities the site representative recommended be eliminated or increased in scope. The team examined each treatment system, focusing on treatment type and volume of waste targeted to the system, and decided which would remain in the OAT Proposed configuration. Both site programmatic and technical representatives were present to discuss with the group why different systems were or were not justified. The team discussed the relative importance of the various evaluation areas and used the database to quantify results as the Proposed configuration was further refined. Once this was completed, the team reviewed the remaining treatment systems based on the evaluation areas to determine if the list was acceptable from a national perspective. The result of this process was an OAT Proposed configuration which was sent to the sites for review and comment. The team then met in November 1994 with additional site representatives to resolve site comments. In addition, the Modified Centralized treatment configuration was developed to provide a least-cost reference point. The results were provided to NGA and the states, and their issues and concerns were identified.

As costs and schedules were further refined at the site level, the OAT continued to meet and work towards an OAT Proposed configuration which would be the basis for the Proposed STPs. The costs derived from the cost model were merely used for comparison of configurations. Actual facility estimates generated by the sites, using consistent Headquarters guidance, became the basis for cost-

dependent decisions. The final OAT Proposed configuration was approved by DOE management in February 1995. This configuration became the basis for the site Proposed STPs and was thus the baseline for the STP process. Any changes resulting from sites' further review of both commercial and available treatment options had to be approved by the OAT team leader and FFCAct Task Force.

Assumptions

The proposed OAT configuration could only be valid if certain key assumptions were considered. These assumptions were as follows:

- States will agree to accept some treatment of out-of-state waste
- DOE will be able to operate existing incinerators
- DOE will be able to use off-site existing commercial treatment facilities
- Existing commercial and mobile treatment will be less expensive and will treat waste faster than construction of a new facility

The OAT, NGA, and DOE management agreed with application of these assumptions in the OAT process.

Mobile Treatment

In preparing the Draft STPs, the states asked DOE to evaluate on-site waste treatment to the extent possible. Accordingly, the states were very interested in evaluating the use of mobile treatment across the DOE complex. Mobile units offer a compromise between building expensive permanent treatment units at smaller sites and shipping waste off-site for treatment.

The Albuquerque Operations Office relied heavily on mobile treatment options when preparing their Draft STPs due to the mid-sized mixed waste volumes present at most of their sites. The rest of the sites were encouraged to evaluate use of mobile treatment in preparing their Draft STPs. The initial OAT Proposed configuration doubled the use of mobile treatment originally proposed in the Draft STPs. A more concentrated effort was then focused on maximizing the capabilities and use of the mobile units. This effort further increased the proposed use of DOE complex mobile units.

EVALUATION RESULTS

The results of the OAT evaluations of the initial three configurations (Draft STP Baseline, Quick Start, and Centralized) provided a starting point for developing the OAT Proposed configuration. Table I, below, shows how each of the initial five configurations evaluated by the OAT rank relative to one another in five evaluation areas. As noted earlier, the sixth evaluation area--"Use of Alternative Treatment Technologies"--was judged equivalent for all treatment configurations.

The high, medium, and low ratings are defined as follows:

HIGH Indicates the configuration was rated more favorably in this area than the other configurations.

MEDIUM Indicates the configuration was rated no better or worse in this area than other configurations.

LOW Indicates the configuration was rated less favorably in this area than the other configurations.

Table I

Comparing the ratings of the OAT Proposed configuration demonstrates how this alternative optimized the positive attributes of the other configurations. While the OAT attempted to maximize the use of existing facilities, no configuration rated "high" on Table I in this area because it appears there is still excess capacity available. The excess is due to a lack of waste currently available to support the existing types of treatment. The excess capacity is planned to be used to treat future environmental restoration and decontamination/ decommissioning mixed wastes.

In creating the OAT Proposed configuration, certain Draft STP treatment systems were eliminated. The main reasons for elimination of a treatment system are the following:

- There is excess capacity at existing DOE or commercial treatment systems
- mall volumes of waste were targeted for a different facility where other options were available
- More favorable treatment schedules were available at other existing or proposed treatment systems
- There is the potential for cost savings through use of another option
- Interstate transportation of large waste volumes could be minimized

For example, during development of the Proposed STPs, DOE Oak Ridge determined that developing their own mixed waste treatment capabilities was prohibitively expensive. They proposed to meet their mixed waste treatment needs through using both established and developing commercial processes using DOE mobile units. Later, after the Proposed STPs were submitted to the states, DOE Idaho also chose to replace their Idaho Waste Processing Facility (IWPF), consisting of 10 treatment trains, with a commercial option. These decisions significantly affected the overall cost and schedule of the final configuration.

Use of DOE mobile units was significantly increased through the OAT process. The Oak Ridge Mixed Waste Treatment Facility reactive metals treatment train was replaced with the Los Alamos National Laboratory Reactive Metals Skid mobile treatment unit. This change diverted treatment of 46 cubic meters of reactive solids to a mobile unit and avoided construction of this treatment system. A similar case was the replacement of the IWPF debris decontamination train and the Idaho WEDF macroencapsulation train with the Los Alamos Lead Decontamination Trailer and the Pantex Mobile Macroencapsulation Process, respectively. These two replacements diverted 400 cubic meters to mobile units that could be moved to Idaho and avoided construction of two new treatment trains and transportation of the waste off site.

These results from the OAT analyses, in addition to discussions with the sites, States, Environmental Protection Agency, and DOE management, led to the final STP configuration. Funding constraints led to further streamlining, which was mainly accomplished through privatization of some treatment options as previously described. Table II shows quantitatively how the final STP configuration (as of 10/95) measures up against the DSTP Baseline configuration in some key areas.

Table II

The overall "report card" for the STP process, as developed by the NGA from information requested by the states, shows the following.

States requested that DOE:

- Maximize use of on-site treatment
  - ...1.6% of MLLW is targeted out-of-state
- Evaluate use of mobile treatment units to minimize waste movement
  - ...10,725 m3 of waste is targeted to mobile units (9% of MLLW)
- Use existing facilities wherever possible
  - ...About 19% of MLLW is targeted to existing facilities
- Evaluate a nearest-site scenario for waste sent out-of-state
  - ...OAT evaluated nearest-site configuration
- Establish satisfactory schedules for waste treatment
  - ...Consent orders for treatment schedules signed for 29 of 35 STPs
- Address disposal of treated mixed waste

Disposal process is on-going

From DOE's standpoint, this report card shows a very successful effort on both the states' and DOE's part. DOE's cost saving initiatives, coupled with the equity and risk minimizing initiatives injected by the states, resulted in the best possible approach for mixed waste at the federal, state and stakeholder levels.

FUTURE EFFORTS OF THE OAT

Because many well-established technologies already exist in the hazardous waste industry, and are constantly improving, DOE would like to benefit from this expertise by working with commercial organizations to adapt or apply these technologies to mixed waste. In order to fully utilize this option and focus this effort, the OAT has evolved into the Commercial Options Analysis Team (COAT). The goal of this team is to continually improve our MLLW treatment alternatives, since many new facilities are not scheduled for several years and are not fully scoped. The goal of this team is to find ways to do things better, faster, and cheaper. The COAT is working with the technology development Mixed Waste Focus Area to identify existing partnerships with industry, demonstrations, and promising new technologies that can be applied to this MLLW effort. The COAT will continue to coordinate with the states and NGA, and sites will

notify the states of changes to STP treatment options according to the requirements of their FFCAct implementing orders.

20-5

DEVELOPING A NATIONAL TREATMENT CONFIGURATION FOR TRANSURANIC MIXED WASTE

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ABSTRACT

The Department of Energy (DOE) manages its transuranic (TRU) waste in a manner which protects human health and the environment and meets regulatory requirements. Waste managers continue to work with federal and state regulators to implement the Federal Facility Compliance Act (FFCA), since much of DOE's current inventory of TRU waste is suspected to contain hazardous constituents and is managed as mixed waste. Treatment of this waste is addressed in individual Site Treatment Plans (STPs) and consent or unilateral orders developed through the FFCA process.

Site Treatment Plans (STPs) were developed consistent with DOE's TRU waste management policy, which prescribes waste processing/treatment to meet waste acceptance criteria (WAC) for disposal at the Waste Isolation Pilot Plant (WIPP). The STPs describe each site's mixed waste treatment plans with associated cost and schedule, which in the case of TRU waste is generally waste characterization and any needed repackaging.

As the FFCA process proceeded, DOE determined a need to develop a national waste treatment configuration analysis for TRU waste and certain mixed low-level wastes (MLLW) excluded from prior configuration deliberations. The analysis, conducted with participation of effected states, will be completed to support future discussions with the regulators and possible revision to STPs. This analysis, when complete in fiscal year 1996, will formulate a national TRU waste treatment configuration that addresses options such as: aligning treatment schedules to WIPP operating schedules, using mobile waste characterization equipment in existing facilities as opposed to constructing new facilities; and using more robust treatment for commingled TRU waste/MLLW when found to be cost effective for the total waste management system.

The configuration analysis is part of the National TRU Waste Management Plan being developed by the DOE Carlsbad Area Office. A systems model developed for the TRU waste management system is one tool that is being used in the analysis. Acting on behalf of affected states, the National Governors' Association is peer reviewing the process, assumptions, and scenario input to ensure that the results will be valid.

INTRODUCTION

The Department of Energy and its predecessor agencies have generated waste with transuranic (TRU) radionuclides since the early 1940s primarily by weapons manufacturing, plutonium recovery operations, and research and development activities. More recently and in the future these wastes will result from environmental restoration, facility stabilization, and weapons dismantlement activities.

Transuranic Waste

Transuranic (TRU) waste is defined as, without regard to source or form, waste contaminated with transuranium radionuclides (atomic number greater than 92) with half lives greater than 20 years and concentrations greater than 100 nanocuries/gram at the time of assay.

A TRU waste definition has existed since 1970, although such wastes have been generated since the 1940's (Manhattan Project). The initial TRU waste definition had concentrations greater than 10 nanocuries/gram. The current definition, adopted in 1984, increased the level to 100 nanocuries per gram. Wastes generated since 1970 and classified as transuranic waste have been stored in a retrievable manner pending shipment to a deep geologic repository when disposal is available.

This category of waste is peculiar to the Department of Energy (DOE). Transuranic elements include isotopes of plutonium, neptunium, americium, curium, and californium. The primary contaminant being plutonium, due to the nature of waste generated from recovery of plutonium during research and development and weapons production activities by DOE and its predecessors.

The TRU waste management strategy has evolved over the years to reflect increased regulatory requirements and emphasis on protecting the environment and the safety and health of workers and the public. The strategy continues to evolve as stakeholders increase their participation in the process to set waste management priorities with limited resources available. The strategy for managing defense-related TRU waste is to retrievably store waste in regulatory compliant facilities for the near term prior to permanent disposal in a deep geologic repository, the Waste Isolation Pilot Plant (WIPP). Prior to transport and disposal at WIPP, TRU waste must be characterized, treated/processed (as necessary), and certified to meet the WIPP waste acceptance criteria (WAC).

Approximately 70,000 cubic meters of contact-handled transuranic (CHTRU) waste and 1,600 cubic meters of remote-handled transuranic (RHTRU) waste are retrievably stored at 10 major sites and numerous small quantity sites throughout the nation. Over 97 percent of the stored CHTRU waste by volume is located at 5 sites: Hanford Site, Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Rocky Flats Environmental Technology Site, and Savannah River Site. Much of this waste is stored on pads and covered with earth pending retrieval and characterization prior to shipment to WIPP for disposal. About 87 percent of the stored RHTRU is at the Oak Ridge National Laboratory.

As much as two-thirds of existing TRU waste in storage is managedas mixed waste, meaning that in addition to the radionuclides it contains hazardous wastes regulated in accordance with the Resource Conservation and Recovery Act (RCRA) and subject to provisions in the Federal Facility Compliance Act (FFCA).

TREATMENT - A TRU WASTE PERSPECTIVE

Treatment is defined as any method, technique, or process designed to change the physical or chemical character of the waste to render it less hazardous, safer to transport, store, or dispose. Treatment can range from repackaging or overpacking waste, to volume reduction, to destruction of hazardous constituents and solidification of residues. Numerous federal, state, and local regulations, DOE Orders, and agreements establish the legal and regulatory framework under which the Department must manage TRU wastes. Authority to manage TRU wastes is provided to the Department through the Atomic Energy Act of 1954 (AEA), as amended, for radioactive constituents. The Department's authority

under the AEA is formalized in DOE Order 5820.2A, "Radioactive Waste Management."

DOE Order 5820.2A Transuranic Waste Treatment Requirements

TRU waste shall be certified in compliance with Waste Isolation Pilot Plant Waste Acceptance Criteria (WIPP WAC)

Mixed TRU waste shall be treated, where feasible and practical, to destroy the hazardous waste component

Classified waste shall be treated to remove or destroy classified characteristics

As part of waste minimization efforts, waste volume reduction techniques, such as incineration, compaction, extrication, and shredding, shall be implemented, wherever cost effective and practical

Treatment facilities must be permitted by appropriate regulatory authority.

The Order clearly requires the waste generator to provide processing/treatment to minimize volumes, remove classified characteristics, and segregate TRU from other waste types. Waste characterization (and possibly some treatment) in order to obtain the WIPP WAC certification are necessary for WIPP disposal. The operative words on treatment requirements with regard to mixed waste are "where feasible and practical." The Order allows, but does not prevent or require, TRU waste from being treated to levels beyond WIPP WAC requirements. The Order is currently being revised and is expected to be issued later this year.

Regulatory authority for any hazardous constituents comes under the auspices of the U.S. Environmental Protection Agency (EPA) and states primarily through RCRA. Agreements have been established at many sites with local and state regulators that define specific waste management requirements.

The FFCA was enacted into law on October 6, 1992, to amend the Solid Waste Disposal Act, which was previously amended by RCRA. The FFCA required DOE to develop site treatment plans (STPs) which describe mixed waste treatment capacity and technology development, including schedules, for bringing each DOE mixed waste generator or treatment site into full compliance with RCRA. EPA or the effected state then negotiated legally enforceable compliance agreements with the Department to allow continued waste storage subject to RCRA.

The Department's policy is to dispose of defense-related mixed TRU waste in WIPP using a No-Migration Determination (NMD) as established in RCRA regulations, thus the FFCA process produced STPs that focus on mixed TRU waste treatment capability to meet requirements for WIPP disposal and not treatment to meet Land Disposal Restrictions (LDR) under RCRA. Disposal of mixed TRU waste at WIPP is regulated by both the EPA, for waste disposal regulations and NMD, and the State of New Mexico, for RCRA Part B permit.

Resource Conservation and Recovery Act (RCRA) of 1976

Title 40, Code of Federal Regulations, Parts 260-270

Applies to the treatment, storage, an disposal of mixed TRU waste (and waste that has the possibility of being mixed)

Requires all waste generation and storage sites to store any RCRA-controlled hazardous waste only for limited periods prior to shipping them to a regulated treatment or disposal facility

Regulates treatment, storage, and disposal facilities through permits issued by EPA or states with authority granted by EPA (Part 264)

Restricts what materials may be disposed through land disposal restrictions (LDR), thus necessitating treatment to remove or make innocuous hazardous constituents prior to disposal

Provides for a No Migration Determination for the disposal facility to exempt waste from treatment to LDR requirements (Part 268)

Waste Disposal Regulations

The WIPP Land Withdrawal Act, enacted in 1992, established a regulatory framework for WIPP

ERA regulations for governing the management of transuranic waste are promulgated in 40 CFR 191

Criteria for WIPP to meet the regulatory standard in 40 CFR 191 are promulgated in 40 CFR 194

The radioactive portion of waste disposed at WIPP is regulated by EPA under Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Wastes (40 CFR 191).

As the Department finalizes the No Migration Variance Petition and resolves issues associated with the Part B Permit, it may be necessary for the Department to add engineered barriers to the repository or alter the waste package or waste form, such as waste treatment, to comply with the disposal regulations.

Formal decisions on waste treatment will be made based on the disposal regulatory compliance requirements and within the National Environmental Policy Act (NEPA) process. A Draft Waste Management Programmatic Environmental Impact Statement (WMPEIS) has been prepared to support anticipated waste management system configuration decisions. The WMPEIS is examining a range of TRU waste treatment configuration alternatives ranging from the current (decentralized) approach to regional and centralized alternatives. Impacts of the level of waste treatment performed, e.g., WIPP WAC, are also being addressed with respect to impact on configuration.

In addition to the WMPEIS, the second supplement to the WIPP Environmental Impact Statement (SEIS-II) for disposal operations is currently being prepared. A record of decision regarding TRU waste treatment within the NEPA process will use the results of the WMPEIS and WIPP SEIS-II. The record of decision is scheduled for fiscal year 1997.

TRU WASTE CONSIDERATIONS IN THE FFCA PROCESS

During the development of the STPs, concerns were identified regarding system-wide decisions for all waste types. In particular for mixed-TRU waste, the plans were developed based on the start of WIPP disposal operations in 1998 under an NMD (and certification of compliance) from EPA. Within this strategy each site developed its STP assuming TRU waste would be characterized and treated only as necessary for shipment and disposal at WIPP. As the STPs matured, there was a need to ensure that the national TRU waste treatment and disposal strategy was comprehensive and consistent. In many instances resource constraints forced scheduling of TRU waste retrieval and characterization activities well into the future and in some instances beyond the projected end of WIPP disposal operations. There was a need to integrate complex-wide characterization and treatment schedules with the disposal schedule. The final STPs attempted to correct some of the schedule inconsistencies as certain sites re-evaluated their plans and schedules with a new forecast WIPP disposal operations schedule. The TRU waste treatment schedules in the STPs vary from site to site, but have operational schedules which now are

within the current WIPP disposal operations window. The current forecast for WIPP operations is to begin CHTRU waste receipts in 1998 on a limited basis taking several years to ramp up to full operations and begin RHTRU waste receipts in 2002 under the current forecast. Disposal at WIPP will continue through 2032, for a total operating period of 35 years.

One other TRU waste activity had a significant impact on the FFCA process. A decision was made by the Department to further study and begin procurement on the Advanced Mixed Waste Treatment Facility (AMWTF) at INEL for treatment of commingled mixed low-level waste (MLLW) and TRU waste.

Advanced Mixed Waste Treatment Facility

In January 1994, DOE's Idaho Operations Office issued a Request For Proposal for feasibility studies to obtain industry's best thinking for an approach to cost effectively treat large quantities of stored commingled TRU and mixed low-level waste (MLLW). The results of the feasibility studies indicated that at INEL it may be cost effective to treat their commingled MLLW and TRU waste to requirements of Land Disposal Restrictions (LDR) established in RCRA regulations.

This conclusion was a result of processing requirements for the large volumes of stored mixed TRU waste commingled with MLLW. The MLLW is generally the same physical and chemical form as the TRU waste at INEL, but it contains concentrations of a alpha-emitting radionuclides under the threshold in the TRU waste definition (100 nanocuries per gram). The MLLW must be treated to comply with LDR requirements prior to disposal in a permitted disposal facility. The proposed strategy treats the commingled waste as a single waste source, resulting in a concentrated waste product that meets the definition of TRU waste which can be disposed at WIPP.

Cost savings to the waste management system at INEL have been identified as a result of eliminated costs needed to characterize and segregate the MLLW from the TRU waste, reduced storage and transportation costs due to treated TRU waste volume reduction, and the use of a privatized facility. It is unknown at this time whether the unique waste management strategy being considered by the Idaho Operations Office is applicable to other sites.

The Department considered, in this case, more robust treatment of TRU waste than required for WIPP disposal, since initial feasibility studies indicated it to be cost effective from a total waste management systems perspective. Since the facility, when operational, may be able to treat similar waste from other sites, there is a need to systematically evaluate TRU waste treatment facility configuration for sites with similar commingled MLLW and TRU waste.

During STP discussions with certain states concerns were raised that WIPP may not open on schedule or that the NMD may not be issued. The Orders issued by the states under the FFCA, for the most part, reflect the Department's TRU treatment and disposal strategy in the STPs, but some went further to provide specific conditions to address these concerns. New Mexico issued a unilateral order which requires the Department to submit plans and schedules for treatment of TRU wastes at LANL and Sandia National Laboratories - New Mexico by 1999. The Department can request discussions with the State regarding treatment requirements if WIPP receives its NMD and opens.

CONFIGURATION ANALYSIS

As discussed above the FFCA process resulted in STPs with a configuration for TRU waste facilities that, in general, is consistent with the current WIPP disposal operations window. There was no attempt to optimize the configuration in the STPs, to address potential privatization, or look at the tradeoffs of new permanent facilities, or use of mobile capability for characterization and limited treatment. There was no attempt to fully analyze the proposed AMWTF at INEL with respect to the remainder of the TRU waste system. Since the STPs are revised annually, the opportunity exists for the States and the Department to modify plans in the future. The National Governors' Association (NGA) was briefed on a proposed configuration analysis as a first step.

The Department's National Transuranic Waste Program (NTP) team is currently developing a National TRU Waste Management Plan to detail a strategy that aligns the complex-wide TRU waste management activities with the WIPP disposal schedule. The NTP team is formulating a strategy that will optimize the amount of TRU waste available to support the disposal schedule. This Plan will consider existing site facilities, proposed new facilities, and mobile units to prepare, characterize, and certify waste for disposal at WIPP. The Plan will consider use of mobile units at sites where additional or expanded facilities are needed to characterize, re-package, or otherwise treat TRU waste to meet WIPP waste acceptance criteria. The use of mobile units or transportation to larger sites will be an important consideration for small quantity sites, where construction of permanent facilities is not practical. The National TRU Waste Management Plan will be complete in late fiscal year 1996.

The National Transuranic Waste Management Plan analysis examines the current configuration, the configuration proposed in the STPs, and various alternatives. The analysis will assess where commingled MLLW and TRU waste inventories exist, consider expanded use of the AMWTF, and/or consider other regional treatment centers for TRU waste. It was determined that the systems analysis and results could not be completed in time to support the initial STPs, but the studies could be completed and results available for future discussions with the states on annual updates of the STPs. These studies will also be linked to similar future activities planned in the MLLW area.

The NTP team is developing a model that simulates waste processing (characterization, treatment, certification) at the generator sites in a systems context. The model tracks in detail waste movements in each scenario analyzed and uses cost modules from the Systems Cost Model developed by the Idaho National Engineering Laboratory, which were utilized for the WMPEIS and the Baseline Environmental Management Report. The model results will allow multiple scenarios to be analyzed and help screen scenarios for further consideration.

National TRU Waste Management Plan

Input:

Waste characterization studies (mobile systems for NDA/NDE, active and passive neutron detection, headspace gas sampling, sample coring, etc.)

Waste certification quality assurance program plans

Waste information (TRU Waste Baseline Inventory Report)

INEL treatment studies (AMWTF)

Current site capacity for characterization and treatment with costs and schedules

Output:

Waste schedule for disposal integrated with generater site plans

More optimized system configuration for waste characterization and treatment

Prioritization of projects and activities

Waste removal plan for small quantity sites

Integration of commingled MLLW in TRU waste planning

The proposed plan was presented to the NGA and feedback has been positive. NGA and state representatives have visited WIPP and met with the NTP team to review the National TRU Waste Management Plan process. NGA and state participation in the process will continue throughout this year with regard to reviewing the process, assumptions and data, selecting scenarios to be analyzed, and peer reviewing tools and models. Through this participation, the results will be comprehensive, meeting the states' and DOE's needs.

NEXT STEPS

The results of the configuration analysis will be discussed with the NGA and states. Active involvement by NGA and states in the process will ensure success. The results will provide input on suggested changes to Site Treatment Plans that will provide an integrated and coordinated approach to TRU waste management. The Department will be able to utilize the information in its budget formulation process to impact decisions regarding its request for fiscal year 1998 and beyond. The cooperative FFCA process, successful to date, continues as the TRU waste treatment system configuration matures.

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ACCELERATED ROCKY FLATS MIXED WASTE TREATMENT FOR FEDERAL FACILITIES COMPLIANCE ACT IMPLEMENTATION

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ABSTRACT

Rocky Flats Environmental Technology Site (RFETS) received a Compliance Order from the State of Colorado on October 3, 1995, approving with modifications the radioactively contaminated hazardous (mixed) waste strategies outlined in the RFETS Proposed Site Treatment Plan (STP) (1) and committing the Department of Energy (DOE) to actions that will achieve compliance with the land disposal restriction (LDR) regulations

covering the hazardous portion of mixed wastes presently stored at the site. The STP was submitted to the State of Colorado per the mandates of the Federal Facility Compliance Act of 1992 (FFC Act) and was the result of an iterative process between the DOE, the lead regulatory agency from the State of Colorado, and local stakeholders. The STP presents a baseline implementation approach for compliance with LDR standards given the strict federal budget constraints forecasted for future years. Accelerated treatment of the RFETS mixed waste inventory to meet LDR limits is required to satisfy several purposes. In order to support a proposed Accelerated Site Action Project (ASAP) that has the objective of accelerated deactivation and cleanup, treatment and disposal of the mixed waste inventory is necessary within the next ten year period (2). Accelerated treatment has other merits for its implementation including outyear storage cost avoidance, as portions of the mixed waste inventory are stabilized and disposed in such a manner that can facilitate future retreivability and offsite disposal, if economically feasible. This paper will discuss how the magnitude of the storage costs for maintaining the mixed waste inventory at RFETS has become an important economic driver for accelerated mixed waste treatment in light of continued limited resources available to the site. Options for accelerated waste treatment, with an emphasis of more accomplished with less resources, are being explored and include the use of regional treatment facilities, outsourcing of selected waste streams to commercially available treatment systems or accelerated construction and operation of treatment facilities through private sector initiatives. Accelerated mixed waste treatment has captured widespread interest within the DOE complex and the commercial industry as both institutions grapple with similar issues regarding conventional mixed waste management strategies. It is the intent of this paper to communicate the experiences gained at RFETS in evaluating and exploring accelerated treatment strategies, the implementation for these strategies at the site, and foster the active participation of all DOE or commercial interests engaged in the development of mixed waste treatment solutions as an answer to compliance under the FFC Act.

INTRODUCTION

The Rocky Flats Environmental Technology Site (RFETS) is a government-owned, contractor-operated facility which is a part of the nationwide Department of Energy (DOE) nuclear weapons production complex. Prior to 1989, the primary mission of site was the continual production of components for nuclear weapons. Production activities included metalworking, fabrication and component assembly, plutonium recovery and purification, and associated quality control functions ensuring the technical performance of the weapons' components. The plant was built in 1951 and began operations in 1952. In 1989, as a result of a changing international political climate, the decision was made by the United States government to discontinue production of components for nuclear weapons at Rocky Flats. Rocky Flats has undergone a transition from a weapons production facility to an environmental restoration and waste management site. The current mission of the site is to manage waste and material, clean up and convert RFETS to beneficial use in a manner that is safe, environmentally and socially responsible, physically secure, and cost-effective.

As a result of these activities of over forty years in production, RFETS has accumulated a significant quantity of mixed wastes subject to the regulation under the land disposal restriction (LDR) standards of the

Resource Conservation and Recovery Act (RCRA) (3). These waste streams include low level mixed waste and transuranic mixed waste, both subject to regulation primarily due to solvent and heavy metal contamination arising from RFETS heavily industrialized past. The LDR compliance issues are not unique to RFETS but are relevant to all DOE sites that manage mixed wastes given the recognized lack of existing treatment capacity and technology applications within the DOE complex for the treatment of LDR mixed wastes.

ISSUE DEFINITION/RESOLUTION

As required under the FFC Act, the DOE submitted the RFETS Proposed Site Treatment Plan (STP) in early April and received on October 3, 1995, a Compliance Order from the State of Colorado Department of Public Health and Environment (CDPHE). The Compliance Order approved with modifications the mixed waste management strategies outlined as the baseline implementation path in the STP. Five mixed waste management strategies comprise the framework for achieving compliance with the Order. Three of these strategies are focused on compliance for mixed low level (MLL) wastes with the remaining two strategies are oriented towards compliance with mixed transuranic (MTRU) wastes. For MLL wastes, the framework for compliance includes the use of a graded characterization program, evaluating the physical and chemical properties of the waste, to determine the specific applicability of LDR standards. The second MLL strategy in the STP baseline has identified treatability group matches for treatment offsite at existing or planned DOE or commercial facilities. The third MLL strategy proposes treatment of the mixed waste backlog through the implementation of three capital projects at the Site. The capital projects focus upon two primary treatment methods; separation of solvent contaminants from the physical waste matrices and immobilization of heavy metal constituents. The STP baseline represents an enforceable treatment-based effort spanning twenty five years and costing over $900 million dollars. Over $600 million dollars or an average of $24 million dollars per year, of this implementation cost is associated with the continual storage of backlog and newly generated MLL wastes while treatment technologies and capacities are developed and operated.

The dilemma of increasing costs to implement the cleanup of legacy wastes at RFETS is shared among other sites in the DOE Complex. In response to this challenge, DOE has formed task force working groups to identify privatization initiatives and change the fundamental way that DOE has conducted business in the past. DOE's goals for privatization include improving the quality of contractor performance and saving taxpayer dollars with no sacrifices made in DOE's commitment to protection of the environment, public health, or safety. Preliminary data gleaned from initial DOE ventures into privatization have generally demonstrated increased cost effectiveness, schedule efficiencies, and the development of new contractual relationships between DOE and vendor participants.

A strong economic incentive exists at RFETS for identification of alternatives to the STP baseline which can accelerate treatment schedules and generate significant cost savings in the storage and management of backlog wastes. A recent development at RFETS is the preparation of a draft conceptual vision by the new integrating management contractor team which took over day-to-day operations at the site in July 1995. The Accelerated Site Action Plan or ASAP focuses on accelerated cleanup, plutonium consolidation, physical plant conversion, and end land use. A

major premise underlying ASAP is that all stored plutonium and other special nuclear materials will have been removed from the site by the target date of 2015. In addition, no DOE-operated buildings will remain at the site. From this perspective, a new approach for compliance under the FFC Act, while integrating with the ASAP focus on faster and more cost-effective cleanup, is what is required. The issue facing RFETS is how to challenge conventional methodologies for onsite waste management, accelerate mixed waste treatment at the Site, and capture the cost savings that accompany avoiding the large operating budget outlays in storage costs and capital investment for on site treatment systems as projected by implementing the enforceable STP baseline.

Accelerated treatment of LDR wastes within the next ten years can be accomplished through initiatives with the private sector especially if certain barriers common to private sector contracting are resolved effectively. Such initiatives can be successful due to the expedited schedules possible by exiting the DOE funding cycle constraints and using private sector financing of treatment facilities and operation. Capital cost for development and construction of the treatment facility can be recovered through the use of fixed price or fixed price per unit output contracts. The private firm furnishes goods and services using their own equipment and gains the operating efficiencies developed in other commercial applications. During or after the contract performance, the private firm is able to seek new markets for the equipment outside of the existing contract, thereby further lowering unit price. The new integrating management contract implemented at RFETS is an example of innovative contracting with the objectives of accelerated schedules with greater cost efficiency and productivity.

Commercial enterprises exist that have the proven track record in hazardous and radioactive wastes and are willing to apply significant resources to provide mixed waste treatment capability. These enterprises have developed in support of industries and markets such as the commercial nuclear power industry, both domestic and foreign, the defense industry, and various commercial generators of hazardous waste. A survey of mixed waste treatment capabilities was performed to determine the availability of these systems for RFETS LDR waste by either transporting wastes off-site or bringing or constructing treatment systems on-site. Some important criteria necessary for system selection were that the systems are currently developed, available and ready for implementation in radioactively hot systems with minimal modification. The survey assessed matches between RFETS waste forms and vendor treatment capabilities. With continued research and development to treat RFETS LLM waste, existing, planned, or proposed vendor treatment option technologies will be capable of treating RFETS waste forms. Vendor treatment capabilities include:

- Stabilization/immobilization via the application of cementation, vitrification, or emerging polymer solidification
- Separation or decontamination technology applications through the use of thermal desorption, supercritical extraction, or steam stripping
- Organic destruction through the application of ultraviolet oxidation, non-thermal plasma, incineration, and emerging catalytic chemical oxidation.

Use of on-site high temperature thermal treatment systems has been previously discounted due to high cost and anticipated negative public reaction. Careful consideration of off-site treatment to comply with the

receiving facility waste acceptance criteria is needed to assure that the waste can be economically shipped and treated. Work is progressing on contracting with these facilities to treat some of the higher risk and difficult to treat LDR waste at the commercial facilities.

In September 1995, Rocky Mountain Remediation Services, LLC, (RMRS) prepared a feasibility study (4) that assessed a spectrum of on- and off-site privatization opportunities. Anticipating accelerated and cost-effective treatment of RFETS waste via privatization, the Study purposefully compared the cost, schedule, and risk of all feasible privatization options to the RFETS STP baseline. The Study selected four options for evaluation:

Option 1, the RFETS STP baseline

Option 2, treatment at the Idaho Alpha Low Level Mixed Waste (ALLMW) treatment facility

Option 3, a DOE-provided facility for location of vendor equipment

Option 4, a privately financed, designed, and constructed facility

Option 2 consisted of the Idaho Waste Processing Facility (IWPF)/Alpha Low Level Mixed Waste (ALLMW) Treatment Facility. This facility is a privatized facility proposed to treat TRU and alpha wastes which are stored either in boxes or drums at the Idaho National Engineering Laboratory (INEL). Option 3 conceptualized an on-site DOE-provided facility as the location for skid-mounted, specialized, or niche treatment units provided by commercial vendors. Commercial vendors with niche treatment experience would be selected using a systems approach to campaign RFETS waste forms appropriate to anticipated budget constraints. Option 4 developed the concept of an on-site, newly-constructed, privately financed, and operated facility to treat RFETS waste. This option provides an integrated treatment systems approach that could also make use of commercial vendor skid-mounted, modular technologies. Existing or planned off-site commercial treatment facilities were assessed and found to be of limited value in conforming to the RFETS ASAP timeline.

OPTIONS COMPARISON

Option 1, the RFETS STP Baseline, meets the compliance milestones in the STP Order through the use of resource intensive technology and treatment system development schedules. Financial risk to the DOE is extremely high from both the total life-cycle cost involved with extended storage and treatment schedules, and line-item funding process perspectives. Regulatory compliance, permitting, and transportation risk is perceived as relatively low, however.

Option 2, off-site treatment at the IWPF/ALLMWF, meets STP Order compliance milestones but does not integrate with the objective in ASAP of dispositioning all RFETS mixed wastes within the next ten years. Financial risk to the DOE appears to be relatively low due to private sector financing. The thermal treatment technology selected is robust and suitable for RFETS TRU and LLM waste but may have elevated technological, regulatory compliance, and permitting risk. Further, the need to pretreat wastes prior to shipment add to the costs of this option. Off-site transportation of RFETS waste to INEL may have greater relative risk as perceived by Denver metro area stakeholders (5). This option offers a potential cost and schedule savings, compared to Option 1, of approximately $450 million dollars and 8 years, respectively.

Option 3, on-site treatment in a DOE-provided facility, meets STP Order compliance milestones and supports the ASAP objectives for the Site. This

option's systems design approach offers the flexibility to campaign higher risk wastes in a more cost effective manner. Thus, RFETS will only pay for the treatment capacity required for the Site's needs. Lower technological risk is anticipated from the use of experienced commercial vendors. Minimal capital investment is required of the DOE for retrofitting an existing facility. Operational treatment system costs are expected to be lower due to vendor distribution of fixed investment costs over multiple customers via multiple site operations of skid-mounted systems. This option offers a potential cost and schedule savings, compared to Option 1, of approximately $660 million dollars and 11 years, respectively.

Option 4, on-site treatment in a newly-constructed facility, also meets STP Order compliance milestones and near-term waste management timelines within ASAP. This option requires full capital investment by a commercial vendor to supply a turn-key treatment and storage facility. Fixed capital investment must be fully recovered by processing only RFETS waste, resulting in greater per unit costs than with Option 3. Schedule implementation and operational readiness is anticipated to take longer than Option 3 due to inclusion of an entire entrained process utilizing multiple technologies. This option offers a potential cost and schedule savings, compared to Option 1, of approximately $620 million dollars and 11 years, respectively.

OPTIONS ANALYSIS CONCLUSIONS

The DOE Line Item funding process and constraints result in Option 1's extended schedule and higher life-cycle costs. Off-site treatment technologies and treatment systems (Option 2) offer little near-term treatment capabilities for RFETS waste. Off-site treatment and disposal carries technological, permitting, transportation, and stakeholder risk that does not integrate with the ASAP concept.

Privatization of facility construction (for both Options 3 and 4) and operation result in accelerated schedules, storage cost avoidance, and reduced capital expenditures and operating costs ultimately reducing DOE risk/liability. When risk is factored into the decision-making process, as well as cost and schedule elements, Options 3 and 4 are more viable than either Option 1 (higher cost, longer schedule, greater financial risk to DOE) or Option 2 (longer schedule, transportation/permitting risk).

The Privatization Feasibility Study recommends implementation of Option 3 due to achievable permitting, design, construction, and operational cost and schedule improvements beyond Option 4's capability. Option 3 leverages a vendor's treatment system design, development, and fabrication costs, and eliminates many infrastructure issues associated with siting a new facility at RFETS as presented with Option 4.

RECENT DEVELOPMENTS

An innovative solution for the treatment of LDR waste that allows storage in the waste management unit retrievable storage cell on-site at RFETS is the immobilization system being investigated as part of the Alternative Water Treatment System (AWTS). The AWTS is proposed to replace the aging process water treatment facility at RFETS and avoid the approximate $60-100 million upgrade of the facility for continued use. Treatment of LDR sludge can be initiated by 1998 on tank stored sludge as compared to 15 years later under the baseline site treatment plan. Due to the accelerated schedule and avoidance of capital upgrades, the AWTS life

cycle costs are projected to be approximately half of that required under the baseline approach.

CONCLUSIONS

Through the utilization of life cycle cost analysis as a basis for decision-making, it has become evident that privatization and commercial type contracting of RFETS waste treatment can result in accelerated schedules, significant storage cost avoidance, and reduced capital expenditures and operating costs. RMRS was able to complete a risk based and best management practice treatment prioritization of the stored inventory utilizing commercial expertise of personnel and numerous commercial vendors. From this approach, a solution for RFETS ultimately reducing DOE risk and liability is privatization. When cost, schedule, and risk elements are factored into the decision-making process, Options 3 and 4 are more viable than either Options 1 or 2. Option 3, however, offers the flexibility necessary to campaign waste by waste form, including those of "higher risk", such as liquids, oils, and semi-solids. Such waste treatment flexibility is paramount during DOE budgetary uncertainties.

In order to more effectively bring private sector resources to bear on managing LDR waste problems at RFETS and other DOE facilities, barriers must be resolved. Commercial type contracting reform is needed to allow DOE to enter long-term contracts without the Federal budget uncertainties that make private sector contracting with DOE a high risk prospect. Contract reform efforts with DOE's prime contractors such as the integrating management contract at RFETS should be considered as part of DOE's privatization initiatives. Labor unions throughout the DOE complex view privatization as a potential loss of jobs and wages. Worker and community transition is as important as the actual work and a team approach must be implemented by all affected parties.

To enhance the probability of successful privatization of waste treatment, commercial vendors typically require definition of 1) input waste feed material or waste stream and a guarantee of minimum quantities of feed material to allow fixed price bid proposals, 2) specification of all final product or service requirements including owner/operator penalties for failure to comply with those requirements, 3) investment framework resulting in a balanced risk-reward equation including clear division of responsibility and risk, and 4) regulatory framework clarifying oversight requirements and minimizing regulatory uncertainties (6).

Recent events point towards changing business conditions within the DOE complex that will impact future mixed waste management strategies. Accelerated treatment opportunities can incorporate the changing business climate into future realities by proposing innovative and affordable mixed waste treatment systems within the constrained DOE operating budget. Accelerated treatment at Rocky Flats will seek to become the driving mechanism for expediting compliance with the STP Compliance Order through cost advantages of commercial treatment systems, while balancing the environmental and safety objectives governing the site.

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20-7

DISPOSAL OF MIXED WASTE: TECHNICAL, INSTITUTIONAL, AND POLICY FACTORS*

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ABSTRACT

In conjunction with the affected States as part of their interactions required by the Federal Facilities Compliance Act, the Department of Energy has been developing a process for a disposal configuration for its mixed low-level waste (MLLW). This effort, spanning more than two years, has reduced the potential disposal sites from 49 to 15. The remaining 15 sites have been subjected to a performance evaluation to determine their strengths and weaknesses for disposal of MLLW. The process has included institutional and policy factors as well as strictly technical factors, with each highly dependent on the other: policy decisions must be supported by technical analyses, and technical analyses must be performed within a framework which includes some institutional considerations, with the institutional considerations selected for inclusion largely a matter of policy. While the disposal configuration process is yet to be completed, the experience to date offers a viable approach for solving some of these issues. Additionally, several factors remain to be addressed before an MLLW disposal configuration can be developed.

INTRODUCTION

The Federal Facility Compliance Act (FFCAct) of 1992 (1) requires the U.S. Department of Energy (DOE) to work with its regulators and with members of the public to establish plans for the treatment of DOE's mixed low-level waste (MLLW). Although the FFCAct does not specifically address disposal of treated MLLW, both DOE and the States recognize that disposal issues are an integral part of treatment discussions. The DOE established the FFCAct Disposal Workgroup (DWG) in June 1993 to work with the States in defining and developing a process for evaluating disposal options for treated MLLW. This joint DOE-State process has currently narrowed the DOE sites for further evaluation from 49 to 15.

Several technical, institutional, and policy factors have been used throughout this project. The overriding policy factor was to enable decisions that are supported by technical analyses, that incorporate

appropriate institutional factors, and that give the best value for the tax dollars spent. In this context, the technical analyses are following a phased approach in which 1) technically unacceptable candidate sites were screened from further consideration, 2) the remaining sites were subjected to a scoping analysis to identify their strengths and weaknesses for disposal of MLLW, and 3) results from the scoping analysis will be compared to actual waste streams to determine the ability of the sites to dispose of actual DOE MLLW. These and other technical analyses will provide the support to enable further policy decisions pertaining to the final disposal configuration for MLLW. The final configuration will also include consideration of institutional factors such as the existing disposal infrastructure, other on-going complex-wide assessments, and input from interested stakeholders.

SCREENING ANALYSES

The overriding policy of enabling decisions that are supported by technical analyses, incorporating appropriate institutional factors, and giving the best value for the tax dollars spent was first met by a screening analysis. This analysis eliminated obviously technically unacceptable candidate sites from further consideration so that resources could be focused on the more viable sites. Forty-nine sites that were identified in the first draft of the Mixed Waste Inventory Report (MWIR) (2) comprised the initial universe of potential candidates for MLLW disposal. After consultation with the States, the DWG initiated and implemented a tiered screening process to narrow the field of potential candidate sites from 49 to 15 in two phases (3). The results of this screening process, which were reviewed and agreed to by the affected states, are illustrated in Fig. 1.

Fig. 1

After combining five sites based on geographic proximity, the initial screening eliminated 18 of the most obviously poor candidate sites based on three objective criteria with regulatory or operational basis. The site

- must not be located within a 100-year flood plain (5),
- must not be located within 61 meters of an active fault (5), and
- must have sufficient area to accommodate a 100-meter buffer zone (6).

The second phase of the screening process was based on a more refined evaluation of the remaining 26 sites (7) using several criteria grouped into three board categories:

- technical considerations (e.g., hydrology, geology, topography, and volcanic and tectonic potential),
- potential receptor considerations (e.g., populations significant groundwater resources, and sensitive environments), and
- practical considerations (e.g., ownership, mission, MLLW storage and generation, and regulatory considerations).

Each category was evaluated for each site and assigned a ranking as either a major problem, moderate problem, or a minor problem (3). Baaed on this analysis, the States agreed to eliminate an additional 5 sites from further consideration and to assign a lower priority to another 6 sites. The lower priority sites were to continue to be considered for on-site disposal and would be considered for disposal of wastes from off-site only if a disposal configuration could not be defined with the remaining 15 sites.

SCOPING ANALYSES

A more technically detailed performance evaluation (PE) was conducted on the remaining 15 sites (Fig. 1) to estimate their strengths and weaknesses for disposal of MLLW (4). The PE evaluated the water and atmospheric transport pathways and inadvertent intruder scenarios for 58 radionuclides expected to be in DOE MLLW for trench and tumulus disposal facility types. The permissible radionuclide concentrations in grouted waste were estimated based on site-specific data and on performance objectives determined from DOE Order 5820.2A (8). These "permissible waste concentrations" (i.e., the radionuclide concentrations in waste in a disposal facility that do not exceed the performance criteria specified at the performance boundary) were estimated for each pathway and for each radionuclide. The smallest of these values represents the limiting concentration for each radionuclide at each site. The methodology and results of the PE were reviewed by both internal and external review panels as well as DOE Headquarters, the affected sites, and the States.
A summary of the results of the analysis are shown in Table I. This table presents the radionuclides that were limited by the water or atmospheric pathway for each of the 15 sites. Blank cells indicate that the radionuclide was limited by a human intrusion scenario at the site. Fourteen radionuclides were limited by intrusion at all sites, and an additional 27 radionuclides were limited by intrusion at 13 or 14 of the 15 sites. The results of the PE demonstrated that the intrusion scenarios selected for evaluation, which were based on performance assessments (PAs) of DOE low-level waste (LLW) disposal facilities (9, 10), provided the most limiting permissible waste concentrations for most radionuclides at most sites.

Table I

The water pathway limited several of the more environmentally mobile radionuclides at sites located primarily in the more humid region of the country (Table II). With the exception of C-14, each of these radionuclides is long-lived relative to the 10,000-year period of performance. All the radionuclides have high or medium environmental mobility (4), indicating that they would migrate to the 100-meter performance boundary within the 10,000-year period and, therefore, would not decay appreciably. The number of sites limited by the water pathway is an indication of the relative mobility and persistence of the radionuclides, with Tc-99 and I-129 being the most mobile and persistent of the radionuclides evaluated.

Table II

The atmospheric pathway was evaluated only for the volatile radionuclides tritium (as tritiated water) and carbon-14 (as carbon dioxide gas carrying the C-14 isotope). This analysis indicated that tritium would not be limited by the atmospheric pathway at any site and that C-14 would be limited by the water pathway at about half of the sites.
Although the purpose of the PE was not to eliminate sites from further consideration, the analysis indicated that several radionuclides can be disposed of at the more arid sites at higher permissible concentrations than at the more humid sites. The PE analysis revealed that engineered barriers offer no long-term advantages for the disposal of wastes containing long-lived radionuclides; their benefits are for containing shorter-lived radionuclides while they decay to insignificant levels. The analysis also identified key parameters characterizing both the sites and the wastes and identified several indicator radionuclides which can be used to represent the behavior of broad classes of radionuclides.

The permissible radio nuclide concentrations in waste estimated by the PE will be compared with estimates of radionuclide concentrations in treated MLLW streams to determine the ability of the 15 sites to dispose of actual DOE MLLW. The radionuclide concentrations in treated MLLW will be estimated by using existing waste stream and treatment train databases and process knowledge to estimate the concentration changes due to the various treatment processes. Other information that will be provided by this analysis includes the estimated volume of MLLW after treatment and the usefulness of the existing DOE MLLW databases.

Upon completion of this analysis, the technical capability of the 15 sites for disposal of DOE's MLLW will be presented to the States. Before further progress can be made in refining the MLLW disposal configuration, the incorporation of institutional and policy factors will be required.

Institutional and Policy Factors

The major institutional and policy factors that remain to be addressed before a final disposal configuration for MLLW can be proposed can be grouped into three categories: integration of MLLW into the larger picture of low-level waste disposal; the disposal facility infrastructure; and the evolving regulatory landscape. Each will require one or more supporting technical analyses.

Perhaps the most pressing complex-wide initiative for integration of MLLW disposal issues is DOE's response to the Defense Nuclear Facilities Safety Board's (DNFSB) Recommendation 94-2 (11) pertaining to the DOE's management of low-level waste. One of the most important recommendations of the DNFSB is that the effects of nearby source terms and existing contamination be considered when analyzing the impacts of operations of planned LLW (including MLLW) disposal facilities. Prior to this recommendation, each disposal facility was evaluated according to specific performance objectives independently of nearby facilities or contamination.

Incorporation of existing disposed waste and contamination is problematic in terms of technical issues as well as institutional issues. One of the main technical problems to be addressed is that the inventories of past disposal activities or accidental releases are generally poorly known. This lack of knowledge introduces uncertainty into the analysis that may overshadow the deleterious effects of the wastes to be disposed of. Until characterization of the existing in situ waste and contamination can be characterized, they must be treated in a conservative fashion that may tend to limit the capability of planned disposal facilities. Scoping-level technical analyses can aid formulation of a coherent policy that will in turn result in clear direction for more detailed technical analyses.

One of the important institutional problems this new approach introduces is the distinction between the PA methodology for LLW and MLLW and the risk assessment methodology specified under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) (12). The approaches to evaluating risk and performance, the timeframes of the analyses, and the endpoints for analyses are different for PAs and CERCLA risk assessments. These issues must be resolved before significant progress can be made. Again, scoping-level technical analyses, such as the process described above, can be used to help formulate policy for addressing these issues.

Another complex-wide initiative for integration of MLLW disposal activities is DOE's Programmatic Environmental Impact Statement (PEIS)

(13). A PEIS is required to provide National Environmental Policy Act (NEPA) coverage for major federal programmatic actions and includes a Record of Decision which is issued to support the selected alternative. As currently envisioned, the analyses used to support the planning process for MLLW disposal will be incorporated into the PEIS documentation so that NEPA coverage will be provided for the recommended configuration decision.

Integrating the MLLW disposal configuration process into the existing LLW disposal facility infrastructure is another important institutional factor and is interrelated with the 94-2 and PEIS considerations. Because the nearby existing source terms and contamination will now be considered in the performance assessments of disposal facilities, the future LLW and MLLW disposal volumes must compete for a finite disposal capacity at some sites. The estimates of treated MLLW volume are an important piece of technical data that will factor into the analyses; however, longer-term projections of expected waste volumes are difficult to justify with any accuracy, especially when considering highly uncertain environmental restoration waste volumes.

There are still several institutional issues the must be addressed. Some of the more important questions are the following: How will commercial disposal factor into the analysis? How will the input from the States and stakeholders influence the decision-making process? What changes in operating practices will be required for existing disposal facilities? What factors are the most important in selecting new disposal sites? Are the current disposal sites the most appropriate ones for continued disposal? Technical analyses will provide the basis for answering many of these questions, but some policy decision will also be required to provide direction to those technical analyses.

In addition to the resolution of the internal DOE policy issues mentioned above, at least one external regulatory change is expected to occur that will influence MLLW disposal: the Environmental Protection Agency's Hazardous Waste Identification Rule (HWIR) modifications to the Resource Conservation and Recovery Act (RCRA) (14). The HWIR is expected to establish "exit levels" for listed RCRA wastes that pass a test for leachability. The mixed wastes that pass this test are not required to be disposed of in a RCRA Subtitle C-type disposal facility, and therefore, these wastes can be disposed of in a LLW DOE disposal facility. Presently, listed RCRA wastes remain classified as hazardous even after they have been treated. The HWIR may have the effect of reclassifying some MLLW as strictly LLW in terms of disposal, but it will not reduce the total combined volume of MLLW and strictly LLW to be disposed of.

SUMMARY

Significant progress has been made toward developing a process for determining the disposal configuration for DOE's MLLW. The number of DOE sites being considered has been reduced from 49 to 15, performance evaluation of these 15 sites has been completed, and additional technical analyses are being conducted to determine the technical capabilities of the 15 sites for disposal of treated DOE MLLW. However, several institutional and policy factors must still be addressed. Some of these factors include 1) integrating MLLW disposal with other complex-wide assessments of LLW disposal practices, 2) MLLW and LLW competing for a finite disposal capacity, 3) evaluating the impact of potential changes in the regulations affecting MLLW disposal, and 4) developing a fair and equitable process for determining the MLLW disposal configuration. Each

of these factors has components that are technical in nature and that additional technical analyses can help solve. However, there will always be an intimate relationship between the policy guidance necessary for establishing appropriate technical analyses and the technical analyses that will be used to support policy.

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Session 21 -- INDIGENOUS PEOPLES' PERSPECTIVES REGARDING RADIOACTIVE WASTE MANAGEMENT AND ENVIRONMENTAL RESTORATION

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AN INDIAN TRIBAL VIEW OF THE BACK END OF THE NUCLEAR FUEL CYCLE: HISTORICAL AND CULTURAL LESSONS

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ABSTRACT

Indian tribes of the western United States, including the Nez Perce Tribe, the Confederated Tribes of the Umatilla India Reservation, and the Yakama Indian Nation, have entered into cooperative agreements with the U.S. Department of Energy to oversee the cleanup of the Hanford Reservation, in Washington state. These and other tribes considering involvement in nuclear waste management programs have been subjected to severe criticism from some Indians and non-Indians, accusing them of aiding and abetting the violation of Mother Earth by acquiescing in the contamination of lands by radioactive wastes. The authors suggest that this view of the Indian relationship to nature and the environment is too narrow. While the purpose of this article is not to suggest that Indian beliefs support the location of waste management facilities on Indian land, the authors describe aspects of Indian religion that support tribal involvement in radioactive waste management and environmental restoration, and participation in radioactive waste management decision making.

INTRODUCTION

Indian tribes of the western United States, including the Nez Perce Tribe, the Confederated Tribes of the Umatilla Indian Reservation, and the Yakama Indian Nation, have entered into cooperative agreements with the U.S. Department of Energy to oversee the cleanup of the Hanford Reservation, in Washington state. These and other tribes considering involvement in nuclear waste management programs have been subjected to severe criticism from some Indians and non-Indians, accusing them of aiding and abetting the violation of Mother Earth by acquiescing in the contamination of lands by radioactive wastes. The authors suggest that this view of the Indian relationship to nature and the environment is too narrow. While the purpose of this article is not to suggest that Indian beliefs support the location of waste management facilities on Indian lands, the authors will describe aspects of Indian religion and culture that support tribal involvement in radioactive waste management and environmental restoration, and participation in radioactive waste management decision making.

THE RELATIONSHIP OF THE INDIAN TO NATURE AND THE ENVIRONMENT

When Europeans arrived in the Western Hemisphere, they did not find an empty continent, a virgin land, an untouched landscape. North America was already home to millions of Native Americans. When whites first saw the eastern woodlands or the California coast, they found numerous parklands created by Indian burning techniques. These were not forests primeval, but habitats arranged by humans for the benefit of plants and animals necessary for human subsistence. In climates that permitted horticulture, whites discovered vast fields of crops of corn, beans, squash, and many other food products. Indians developed different species of corn and engineered and constructed huge, sophisticated irrigation projects, particularly in the southwestern United States. From the earliest times,

Indians used fire, stone tools, and other implements to modify their environment in order to survive and to enjoy the world.

American Indians loved nature, and nature granted them the food, fuel, fiber, and materials they needed to sustain life. Their environmental religions promoted both dimensions of human-nature relations. Their religions supported, apologized for, and justified subsistence activities. Their religions asserted both a separation from nature (because nature was the Indian's economic source of life) and a participation in nature (because nature was the Indian's spiritual source of life). Their religions recognized the lack of full human harmony with nature and sought to overcome it.

The buffalo story told by many tribes and the Nez Perce name-giving story are illustrative of this paradoxical cultural view of nature. As discussed below with regard to implications for the back end of the nuclear fuel cycle, these stories capture the essence of living in harmony with nature, balanced with the need for the responsible and respectful use of natural resources. Thus, as human use may alter the natural balance, responsible management actions are required to strike a new, albeit man-made balance.

THE BUFFALO STORY

The buffalo story illustrates the American Indian belief that nature is there to be used by humans, but with consideration and appreciation of the changes that our use effects:

Once, a long time ago, the humans were accustomed to killing and eating members of the Buffalo Nation. They did this all the time and thought nothing of it. So the Buffalo Nation held a big council, and after much debate and discussion the Buffalo Chief announced that the Buffalo Nation would no longer allow themselves to be used by the human beings for food, clothing, and all the things they require to live as humans because they did not appreciate the Buffalo Nation's generosity. The people were soon starving and their moccasins were worn, their clothes were wearing out, and they had no robes to keep them warm in the winter. The Buffalo Chief complained that for years the Buffalo Nation shared their bodies with the People for meat, robes, and homes, and never once did the People mourn the death of the Buffalo or smoke or pray for their return.

The Chief of the People recognized the wrong that the People had committed against the Buffalo Nation and agreed that they would sing and dance after killing the Buffalo so that the Buffalo could return and live again. They did so, and for a long time there was plenty for all.

THE NEZ PERCE NAME-GIVING STORY

The Nez Perce name-giving story carries forward the time-honored theme of respect for the complexity of nature while highlighting the ability and need to adapt to environmental perturbations. This maintains a balance in nature:

One day when Coyote was camped among his animal friends, he was called to the mountain by the Creator. "Coyote come here," the voice called. So Coyote went to the mountain to hear what the Creator had to say. "I want you to call all of the animals together tomorrow morning. The two-legged animal is soon to be here and I want to give the animals names or rename them so that they can be recognized."

All of this was happening when all that existed on this earth were the animals, birds, and fish. They had lived together since the great Creator put together Mother Earth. So Coyote went to all animal camps and told them to be at the mountain at sunup so that they might receive the names

of their choice. Coyote was excited, excited because perhaps he would be able to change his name to something more desirable. Coyote was a name recognized in the animal world as one being sly, crafty, and sometimes dishonorable. He wanted to be something else, more respected, more powerful and strong, like the grizzly bear, the eagle, or the salmon.
So Coyote decided that he would stay up all night so that he could be first in line to choose the name that he wanted. He went back to his tent and sat by his fire, preparing to stay up all night. But near dawn Coyote fell asleep. At dawn, all of the animals were lined up to receive their names while Coyote slept. When Coyote finally awoke, he went to the mountain to receive his chosen name. As he approached the Creator, he noticed that none of the other animals were around, and thinking that he was first, he was very happy. He greeted the Creator and said that he wanted the name of Grizzly Bear, the most powerful to rule the forests.
"Too late!" answered the Creator. "That name has already been taken."
"Then I want the name of Eagle, to rule the skies."
Again, "Too late!" cried the Creator.
"Then give me the name of Salmon," he said in a low and meek tone.
The Creator then told him that all of the names have been taken except for one and that is the name of Coyote. "No one wanted the name of Coyote, and that shall be your name."
Coyote, very dejected, went back down the mountain almost crying. The Creator, feeling sorry for Coyote, called him back. Coyote ran back up the mountain. "Yes, yes, what is it?"
The Creator reminded Coyote that the two-legged animal was coming soon, which is why he was doing the name-giving. "Someone has to prepare all of the animals for the coming of the two-legged animals. That will be your job. You will teach them to be alert, cunning, and evasive. The animals will provide for their subsistence, their food, clothing, and tools. But the two-legged ones must learn to respect the animals that provide this."

THE ROLE OF RELIGION

Indian environmental religions sought to overcome the tension between reverence for and exploitation of the world of nature. Note, however, that the method of overcoming, as in the buffalo story, was apologetic in a double sense. Indians apologized to the animals they killed; they begged their pardon. At the same time, Indians manipulated animals; they thanked them in order that they might kill them in the future. It was said of an Ojibwa cultural hero that "he called the birds, beasts, trees, flowers, and all he saw around him his relatives and friends, yet he was always trying to outwit them and use them as servants, and would maneuver to get them to do his bidding." In the same way, Indians both apologized to nature and created a body of apologist oral literature regarding their use of natural resources. A large corpus of Indian beliefs attempt to justify Indian use of the natural world, including the killing of animals. And Indians could cultivate and use corn because the corn spirit had given the food as a gift.

MUTUAL RESPONSIBILITIES

The relationship between hunter and prey parallels the relationship between any exploiter of natural resources and that resource. What are the mutual and reciprocal responsibilities between hunter and prey? The relationship is of an essentially spiritual nature. The interdependence of life finds expression in Chief Seattle's wisdom: "When you spit on the earth, you spit on yourself. What one does to the web of life, one does to oneself." The hunter seeks his prey. The prey offers itself to the

hunter. The hunter's first responsibility is to sing, dance, and pray to become a worthy recipient of the sacrifice of the hunted. His second responsibility is to sing, dance, and pray the willing sacrifice back to life, fulfilling the spiritual obligation. The third responsibility is to take as little as is required for physical and spiritual survival and to avoid wasting any part of the sacrifice. Wasted sacrifice is sacrilege, the transformation of the sacred to the profane. Therefore the failure of the hunt was not caused by mere misfortune, inferior stalking skills, or by poor marksmanship, but was viewed as a sign of inadequate spiritual preparation by the hunter. An animal would give itself up only to one who was spiritually prepared. Thus the great care with which so many of the Indians utilized every portion of the carcass of a hunted animal was an expression not of economic thrift but of courtesy and respect and an aspect of the religious relationship that exists between the hunter and the hunted.

The hunted's responsibility is to offer himself as a willing sacrifice to sustain the life of the hunter and his kind. In so doing, one validates his true relationship with the other and so maintains the balance and harmony of creation. The common good of the entire community of creation has primacy over individual self-interest.

The buffalo story reveals the responsibility of the hunter to his prey and the corresponding reciprocal responsibility of the prey to his hunter. That story and the Nez Perce name-giving story reveal the Indian social truth of reciprocal responsibility that characterizes Indian social interactionthat one cannot exist without the other. This mutuality of interests whereby one's very existence is dependent on respect, honor, and compassion for self and others (whether they be human or nonhuman, or animate or inanimate) produces a very different model for human social imagination from which to fashion social structures and practices. If a hunter and his prey are unified rather than polarized, then creation itself represents a universal community. The individual responsibility is to assure the continuity of life and, by implication, the well-being of the people. This does not violate the right of any individual, but instead places individual rights in balance with the responsibility for continuity of community life.

THE IMPLICATIONS FOR THE BACK END OF THE FUEL CYCLE

Historically, Indians did not exist in unending harmony with nature, the source of their life. Instead, they cared about and recognized the tensions in their relation with nature, and they tried to reckon with them, make them livable, without doing away with the question of an ethical response to the world. Indians were uneasy about the apparent paradox of being part of the world and also using its resources. Their religions revealed that uneasiness and sought to resolve its tensions, not by denying either the "economic" or participatory dimensions, but rather by affirming them simultaneously.

The experience of Indian tribes with the front end of the fuel cycle, primarily mining and milling, illustrates that paradox. Uranium mining had been at once profitable and devastating to Indian tribes. The lands of the Navajo Nation and the Spokane Tribe are dotted with abandoned uranium mines that pose risks to human health and the environment. Many of the Navajo and other miners who toiled in unventilated underground uranium mines are dead or dying of cancer. The experience of these tribes and the current inability of American society to deal with radioactive waste disposal and environmental issues has led the Council of Energy

Resource Tribes (CERT) to promulgate a policy calling for the phase-out of nuclear power in the United States. At this time, CERT sees a need for further research into reactor and disposal technologies that would lessen the environmental impacts of using nuclear power. The CERT policy also acknowledges the mutual and reciprocal responsibilities of tribes to participate in the safe and environmentally sound management of radioactive wastes.

Ultimately, Indian beliefs do not provide the answers regarding the back end of the fuel cycle. However, they do provide a framework for expanding the scope of the dialogue. For example, we ask whether reprocessing should be viewed as a way of using all of the resource, as a way of avoiding further injury to the Earth, as a way of avoiding further uranium mining and the consequential destruction of the lands of indigenous peoples. Does reprocessing prevent the sacrilegious waste of the resource? Is reprocessing a way of honoring the efforts of those workers whose health was destroyed by mining and milling uranium and fabricating the fuel? Conversely, does the once-through cycle dishonor their sacrifices?

The Indian view of the Earth as Mother also leads to the question of whether the construction of deep repositories is wise. Is the disposal of spent fuel in deep underground repositories with no provision for retrieval tantamount to the entombment of something alive and useful? More importantly, we ask why these decisions should be made without the participation and the point of view of those who have paid some of the highest costs for the United States' entry into the nuclear era, namely, the Indian tribes.

It may be that the involvement of Indian tribes in waste management and environmental restoration is entirely consistent with their religious tenets. Certainly, the Nez Perce Tribe would not undertake any activity or endorse any action that would be inconsistent with its core beliefs. If tribes such as the Mescalero Apaches are willing to assume the responsibility to assure the safety and security of spent nuclear fuel, it may be not only for the monetary benefits but also to help protect the well-being of the people and of future generations. Even if tribal involvement in radioactive waste management is construed by others to be contrary to their teachings and detrimental to their environment, ultimately it is for the individual tribe to decide, based on their teachings, whether these actions comport with their views of their responsibility to Mother Earth. For the Nez Perce Tribe, like the many of the Indians, the world does not consist of inanimate materials to be used and of animals to be butchered and eaten. It is alive, and everything in it is alive and could help or harm people. Such was a basic premise of Indian existence: the world and everything in it is alive and powerful and personally significant. The debate over the back end of the fuel cycle would do well to include these views.

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A LIMITED SAMPLE OF CONCERNS OF THE CONFEDERATED TRIBES OF THE UMATILLA INDIAN RESERVATION COMMUNITY ON USING THE APPROPRIATELY DEFINED RISK ASSESSMENT MODEL

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ABSTRACT
Risk assessment methodologies for traditional American Indian lifestyles are inadequate, and have gone unnoticed in classical risk assessments that typically focus on suburban lifestyles. A potential risk from nuclear or hazardous waste that affects one member of the Confederated Tribes of the Umatilla Indian Reservation (CTUIR) may affect all of the tribal members and may have lasting impacts throughout the entire indigenous community. In order to encompass the wide range of factors directly tied to the culture of the traditional American Indians of the CTUIR, current human health risk assessments must implement a re-structured risk assessment process. These processes must address the overwhelming problems including but not limited to: 1) lack of breadth of coverage, 2) lack of integration, 3) deficiencies related to ignoring or inadequately addressing the CTUIR traditional American Indians' quality of life, 4) the interrelated and inseparable eco-culture, and 5) the unique exposure parameters and pathways. The need for understanding the pathways and associated exposures that directly involve the CTUIR traditional American Indian cannot be understated. The traditional CTUIR American Indian ties to the environment are overwhelmingly more significant and complex than is currently understood by contemporary risk assessors. These ties if holistically accounted for, will play a very important role in determining how future risk assessment methodology is produced. The effectiveness of risk management will be dependent on identifying these cultural unknowns. The issues of environmental racism, environmental justice, and the right to a healthy environment, further highlight a need to formally incorporate affected tribal input.
INTRODUCTION
The Umatilla Indian Reservation located near Pendleton, Oregon is occupied by descendants of three Columbia Plateau Tribes, the Cayuse, the Walla Walla, and the Umatilla (Tribes). The Tribal Government is referred to as the Confederated Tribes of the Umatilla Indian Reservation (CTUIR). As a full service government, the CTUIR Board of Trustees (BOT), makes the decisions on providing detailed information regarding culturally sensitive information.
Under these Tribes' Treaty of 1855 [12 Stat. 945], the Tribes ceded lands to the United States. The lands comprising the eastern portion of the U. S. Department of Energy's (DOE) Hanford Site is among the lands ceded by the Tribes. Under the treaty the Tribes retained rights to perform many activities on those lands, including but not limited to fishing, hunting, gathering roots, berries, and pasturing livestock.
Long standing U.S. Supreme Court precedent holds that the federal government (including its executive agencies) has a trust responsibility to Indian Tribes. This means that the U.S. has a fiduciary responsibility to protect the rights of Indian tribes, including tribes' property and treaty rights. Additionally, a succession of U.S. Presidents beginning with President Nixon, have affirmed a federal policy of upholding tribal sovereignty and dealing with tribal governments on a "government to government" basis. Furthermore, there are federal laws to protect tribes' cultural, religious, and archeological sites, access to, and exclusive use, of those sites, and of traditions, activities, and practices associated with those sites as well as Hanford as a whole. Finally, environmental laws also confer rights upon the tribes. For example, the

CTUIR is a Trustee for Natural Resources under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA).

CTUIR - AN INTERDEPENDENT CULTURE AND ENVIRONMENT

The CTUIR is a sovereign government, that has legal interest in the natural resources upon which the CTUIR's Treaty rights are based, including lands of the Hanford Site. Effective exercise of these treaty rights depends on the health of the natural resources. The CTUIR does not want the people exercising their treaty rights to be placed at risk.

A risk from nuclear or hazardous waste that potentially affects one person of the CTUIR community may have lasting impacts throughout all of the community. In other words, a wave of risk can ripple outwards affecting all of the individuals in our culture, just like a wave generated and propagated in a tapestry. The unique CTUIR culture can be irrevocably changed or extinguished if enough of the environment and the natural resources on which the CTUIR treaty rights are based are irreparably harmed. Without the natural resources, the cultural values of critical significance to the traditional CTUIR American Indian, and her/his community would be lost. If a culture dies, the only remnant is the material culture. In the event of the unthinkable happening, a continuously sustainable natural resource based material culture, such as the CTUIR would rapidly disperse into the natural environment leaving no trace of the living CTUIR culture.

The people of the CTUIR are a unique culture, that has long been complexly intertwined with the environment through their cultural, familial ties, (e.g., marriage, gender, extended families), and relationships with other tribes. The CTUIR people have enjoyed since time immemorial, many types of native foods and artistically constructed items of material culture (e.g., cookware, clothing, etc.). Individual members are an inextricable part of the environment. These members, their community and the environment are essentially one in the same.

The CTUIR culture, which has co-evolved with nature and through thousands of years of ecological education, has provided its' people with their unique and valid version of holistic environmental management. The traditional CTUIR American Indian is aware from cultural teachings that the appropriate behavior leads to continuous sustainable success in gathering food and material. Traditional education regarding food or raw material gathering practices are passed on from one generation to the next, and is done to ensure food for the next season or generation. The knowledge of the many gathering seasons and areas the traditional CTUIR American Indians get to utilize during the year has been handed down from generation to generation. Some CTUIR families teach cultural knowledge in complete secrecy on the maternal or paternal side of the family/tribal unit in order to protect tribal cultural/spiritual knowledge from exploitation from the non-American Indian societies and governments. Within the traditional lifestyle or culture, it simply is not enough to know that there are supposed to be salmon runs at certain times of the year. To sustain the tribes during the remaining interim periods when salmon are not returning to spawn and other foods are available, there has to be knowledge about other interrelated food chain cycles, gathering techniques, preparation, and cultural/spiritual relationships about what is needed for sustenance. This interdependency of the collective knowledge about the seasonal foods not only affects traditional individuals, but affects the whole tribe as a culture. One person can not be expected to know all things. In practical terms, if a tribe depended

on one critical individual, the loss of that one "all knowing" person would effectively end or severely disrupt subsistence existence for the rest of the cultural unit. The same is true of oral tribal history, songs, heritable religious practices and numerous other cultural practices Continuity may depend on specialized knowledge in each generation.

The natural world in the Northern temperate zone operates on a seasonal clock. Traditional American Indians of the CTUIR are influenced by this clock, and expectantly look forward to the next cyclic event. These events include not only birth and death but change in general. Throughout the year, when the CTUIR traditional American Indian participates in activities, (e.g. hunting and gathering for foods, medicines, ceremonial, and/or subsistence), the associated activities are as important as the end product. In the Judeo-Christian tradition, an analogy would be "kosher" dietary practices. In the exercise of these activities, the traditional CTUIR American Indian may cover hundreds of square miles, thousands of feet of relative elevation, and cross numerous types of physiographic provinces. All of the country crossed in the search for food has special meaning to the traditional American Indian and each area demands special effort and behavior. This traditional activity is a key to the hunting of, and gathering of, traditional American Indian foods and culturally significant materials.

All the foods and implements gathered and manufactured by the traditional American Indian are interconnected in at least one, but more often in many ways. For example, trade made up for what could not be physically gathered by one person in one time period. Salmon caught on the Columbia River are often traded for roots, other produce, or material culture. This trade creates a web of interaction and interdependence cutting across families, bands, and tribes. These objects of life are as important to the traditional American Indian as the materials that comprise them.

The people of the CTUIR community follow cultural teachings or lessons brought down through history from the elders. The goal of these teachings is to foster community cohesion and interdependence. Emphasis is placed upon cooperation and helping others in the community, cultivating close community interactions. This is an ancient oral tradition of cultural norms. The material or fabric of this tradition is unique, and is woven into a single tapestry that extends from the past into the future.

RISK ASSESSMENT PATHWAYS

The methodologies used in classical risk assessments are being critically considered by the CTUIR. The classical risk assessment has many deficiencies, including a limited breadth of coverage and lack of integration. Through a pseudo-scientific methodology, the classic risk assessment: 1) ignores time, 2) extrapolates from the lab into the field, 3) contains biotoxicological effects that are not fully understood, 4) ignores multiple pathways and complex contaminants, 5) contains enormous uncertainties, 6) ignores long term impacts, effects to health, environment, workers and society, 7) prejudices future options, 8) loses the big picture by ignoring cumulative effects related to assessing only one chemical/one path/one site assessment at a time, 9)ignores eco-cultural sustainability, and 10)is based on a suburban lifestyle. The holistic environmental management strategies outlined in the Blacksburg Forum (1) or Toward the 21st Century: Planning for the Protection of California's Environment (2) highlight these major problems.

In order to encompass the wide range of factors directly tied to the traditional American Indians of the CTUIR, a risk assessment has to be scaled appropriately. In effect, a re-structuring of the risk assessment process must occur in order to address the overwhelming problems including but not limited to, lack of breadth of coverage, lack of integration and deficiencies related to not addressing the CTUIR traditional American Indians' quality of life, the interrelated eco-culture and their unique exposure parameters and pathways. Other deficiencies include the failure to address the role of time to adequately assess risks to future generations of CTUIR members. The process of American Indian Tribes supplying cultural conversion metrics for risk assessments is, at best, subject to the legislative processes of the various sovereign Tribal governments. Unfortunately for the risk assessor there are few traditional American Indians willing and able to supply the appropriate pathway information, and to say they can speak for any one but themselves. A risk assessor in search of identifying American Indian data gaps has to identify the affected tribe(s) and approach the subject of lifestyles tentatively identified with a potential risk through the proper protocol of the individual tribal government. Until that information is obtained, the results of the classic risk assessment in no way suggest the potential pathways or exposure routes that fall within the breadth, depth, and richness of the CTUIRs' culture. Unfortunately, the processes, the approach and even the necessity to account for traditional American Indian lifestyles have gone unnoticed in classical risk assessments that typically focus on suburban lifestyles. The potential exposure pathways specifically oriented towards the traditional American Indian lifestyles need further identification to ensure protection of the CTUIR and the resources on which CTUIR culture is based. This must be done to provide risk assessors with the most accurate information possible. The principal concerns that affect the CTUIR traditional American Indian relate to a lack of identification of the critical pathways. In addition some risk assessments identify these pathways, "consider" them, and then ignore them, or label them as "insignificant." These multiple potential pathways to exposure are not included in typical suburban exposure pathway model, which has a seriously deficient relationship to the lifestyle of the traditional CTUIR American Indian. Each path stems from unique and multiple uses of the resources for food, ceremonial, cultural, or religious practices. Just as important to the people of the CTUIR are the more intangible considerations such as: aesthetics;physical, economic, community, future well-being, and equity; peace of mind; and sustainability.

A risk assessment covering only mechanistic exposure routes linking a single toxicological component to simple one celled organisms, to mega fauna, then to humans, without accounting for the time involved, does little to express the complexity of the interrelationships between the traditional American Indian, their lifestyles, their relationship with the earth and the natural resources. Anyone attempting to derive and plot on a chart the life cycles of all the native plants, animals, as well as the methods of storage, preparation, and all the unique interrelationships that stem from the area of concern, in order to deduce the complete functional pathways for exposure, will find that the process is probably beyond our capabilities and is expensive. Charting whole ecosystems is certainty not in the realm of this paper, moreover, the thought of placing a value on each and every organism for the purposes of

producing a number, does not convey what is a traditional American Indian entity. Even if a number could be produced, this does not take into account the traditional American Indian values, let alone uptake rates, absorption rates, mutation rates, bioaccumulation rates, and other food chain data needed to make a decision on what is important and what may affect the CTUIR traditional American Indian.

There are some common food plants such as the common cattail, the tule, the willow, and the nettle, that serve dual or more purposes. These could be considered by risk assessors, if nothing less than to point out the enormous data gaps involved. The traditional tribal communities often constitute critical segments of populations whose cultural lifestyles result in disproportionately greater than average exposure potential. Gathering, cleaning, eating, and using these plants may potentially expose many traditional American Indians multiple times, and may subject critical CTUIR population groups to unneeded exposure. The life of the cultural items made from potentially contaminated plants may last years; exposure may occur daily or more, over multiple generations.

Traditional American Indians of the CTUIR have to bear a disproportionate amount of risk in relation to the longevity of radionuclide contaminated groundwater. Take, for example, the common cattail: in the spring the shoots are eaten, the roots are consumed, and the fibrous stalks and leaves are split, woven or twisted. Later in the year the pollen is used in breads, and the stalks are used. The woven products may include food storage bags, food storage baskets, cook hole layers, cooking baskets, mats for the floor, mats for the sweat lodge, or mats for the funerary. Each of these activities necessitates a behavior pattern that encompasses: traveling to the plants, selection, gathering, sorting, cleaning, stripping, peeling, splitting, chewing, and forming of the plant materials. This is just for one type of plant among the hundreds of plants and animals that are used by traditional CTUIR American Indians.

CRITICAL SUB-POPULATIONS OF THE CTUIR

Even during the quest for some food, a typical CTUIR member may potentially be exposed through a variety of pathways. The riverbank walk towards the spring where the plant of interest grows may contain discreet particles of radioactive material, such as Co60. This affects certain subgroups within the CTUIR population more than expected, such as the women and the children. The classic risk assessment focuses on a healthy suburban male of average mass. In comparison the women and children as a result of their smaller mass and shorter stature will receive a higher dose (3). The mud surrounding some Hanford springs may potentially contain Cr [+6], Sr90, or H3.

During the assessment of the quality of the plants (i.e., which ones to select for gathering), a process that demands time standing in spring water, or in spring water saturated mud, could result in absorption of H3 through the skin (4). The women and children, due to their physical characteristics and their culture, may receive greater exposure. Children in particular may be at much higher risk of radionuclide contamination of the environment than adults. Children have a much shorter stature and less body mass than adults, meaning that they have less natural shielding and are closer to source materials.

The gathering process involves not only continued immersion in the spring water, but immersing the hands and compacting mud under and around the fingernails as well. Sorting the plants afterwards, either at the site or elsewhere involves more handling and washing. The bulbs or root of the

food plant may have special cleaning needs. Roots may not be uniformly smooth as carrots or potatoes but undulated, having places where the earth can not be washed out, and if eaten, creates an ingestion pathway for potential exposure. The skin of the root may need to be peeled. Peeling roots is a difficult and time consuming chore involving not only the hands but in many cases a knife and the teeth. Splitting the leaves involves a lot of handling and the experience comes with cuts and abrasions, and more soil accumulation under the nails. If the food is to be eaten and not stored, another potential pathway for contamination is revealed through traditional cooking methods. Local rocks are gathered and heated with local wood. A hole is dug. The heated rocks are dumped in the hole. The rocks are covered with the cattail leaves. The cleaned, peeled, roots are placed on the leaves, and covered with more leaves. This is covered with soil, and a fire is built over the covered cook pit. The result is tasty, but in certain places this type of unique cultural activity could increase exposure. Thus, traditional CTUIR American Indians can be exposed to radionuclides through digging, breathing smoke, breathing dust, breathing steam, eating dust and soil, storing vegetables underground, and eating steamed vegetables.

This risk scenario is but one of many that can be played out for one food, at one site, during one time of the year. The complexities involved with hunting and gathering foods are extremely time consuming and involve at a very primary level many traditional American Indians and the environment. Other significant factors include higher intake rates per body mass for children than adults, the fact that primary gathers are likely to be women of childbearing age, variations in metabolic parameters, and increased risk to CTUIR elders with age-dependent decreased physiological resistance or underlying health problems. Because the CTUIR is unique, risk assessors must realize and accept that the threat to the whole living CTUIR culture begins with two reasons for increased risk: increased exposure and increased sensitivity

The Columbia River continues to be very important to the traditional American Indians that live around it. The river provides a link to the past and a path [for] the future of their children. Understanding the ecosystem and how the traditional American Indian is associated with it is critical for these people and their survival. The health of the river is dependent on the health of the groundwater; the peoples' health is dependent on the river and all that comes from it (5).

The need for understanding the pathways that directly involve the traditional American Indian cannot be understated. The ties to the environment are much more fixed than is currently understood. These ties will play a very important role in determining how risk assessment methodology is produced and how effective risk management will be. The issues of environmental racism, environmental justice, and the right to a healthy environment, highlight a need to formally incorporate affected tribal input.

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UNDERSTANDING THE FEDERAL/TRIBAL RELATIONSHIP AND INCORPORATING TRIBAL VALUES INTO ENVIRONMENTAL POLICY DECISIONS

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ABSTRACT

Federal agency personnel and contractors who find themselves working on projects and in programs that require consultation with American Indian tribal governments and their representatives often find these interactions less satisfactory and useful than they could be. Federal personnel and contractors often find themselves wishing for:

a more practical understanding of the political concepts that have determined the unique relationships that exist between the United States government and tribal governments,

more information about appropriate approaches and methods for interactions with tribes, and

ways to avoid the all-too-frequent political and institutional barriers that impede communication and hinder the building of trust.

This paper explains the key political concepts and federal policies that have defined the federal/tribal relationship. It outlines a style of approach and methods for effective, culturally appropriate interactions with tribal governments and their representatives. The paper also examines obstacles to trusting, effective communication between the federal government and tribal governments; these include conflicting values within tribes as well as political and institutional barriers within agencies. the paper makes these points in part by examining the case of a recent environmental risk evaluation in which JK Research Associates was responsible for coordinating tribal involvement. The paper concludes with recommendations for changes in institutional practices and policy-making that will facilitate greater inclusion of tribes in policy decisions.

INTRODUCTION

Federal agency personnel and contractors and American Indian tribal government representatives who find themselves working on projects and in programs that require consultation, often find these interactions less satisfactory and useful than they could be. Federal personnel and contractors often find themselves wishing for a more practical understanding of the political concepts, treaties, and policies that have determined the unique relationship that exists between the United States government and tribal governments and how the federal/tribal relationship

should be fulfilled within their work. Tribal government representatives and tribal personnel often find themselves wishing that federal agencies would take seriously commitments to consult with tribal governments, consider tribal rights, needs and values, and incorporate tribal information when making policy decisions. Both federal and tribal personnel have found themselves discouraged and wondering how it is possible to work through the all-too-frequent political, institutional, and cultural barriers that impede communication and collaboration and hinder the building of trust between tribes and federal agencies.

This paper explains the political concepts and federal policies that have defined the federal/tribal relationship. The paper examines barriers that hinder the building of a cooperative relationship. These include conflicting and/or misunderstood values, underdeveloped communication strategies, and the absence of guidance to direct federal personnel in how to fulfill the requirements of the federal/tribal relationship within their program responsibilities. The paper examines such barriers by discussing the case of a recent environmental risk evaluation in which the author was involved in coordinating tribal involvement. The paper also discusses two other cases in which the U.S. Department of Energy (DOE) and the U.S. Environmental Protection Agency (EPA) made policy and/or regulatory decisions that would impact tribal lands and tribal health. However, the tribes affected by those decisions viewed tribal input to those decisions as non-existent or not very meaningful. The paper concludes with a discussion of strategies that tribes and federal agencies might consider in order to facilitate meaningful tribal involvement in program and policy decisions that affect tribes and their interests.

THE FEDERAL/TRIBAL RELATIONSHIP

DOE and EPA, as arms of the federal government, are directed to uphold federal commitments within the federal/tribal relationship. In order to fulfill this relationship, federal personnel should understand federal responsibilities to tribes and how they apply as federal personnel carry out their program activities and implement policies that affect tribes. Federal responsibilities to tribes were determined by political concepts, policies, treaties and other laws. These concepts and policies include tribal sovereignty, tribal self-determination, the government-to-government relationship and the federal/tribal trust relationship (1,2,3,4). In addition, some of the tribes near DOE installations have treaty-protected rights to access cultural and natural resources within portions of these installations (5).

Tribal Sovereignty

Sovereignty is a concept that is difficult to define and understand. However, the powers held by sovereign nations illustrate more clearly, the concept of sovereignty. They include lawmaking and enforcement, defining territory, determining citizenship, regulating trade and property, and forming alliances with other nations through treaties, and other agreements. The Indian nations of North America exercised the powers of sovereign national and recognized the sovereignty of one another by making treaties and trade agreements and by forming political and military alliances with other tribal nations. The colonial governments and later the United States recognized the sovereignty of Indian nations by entering into over 800 treaties with tribes. In Worcester v. Georgia (1832), the U.S. Supreme Court said:

. . .the very fact of repeated treaties with [tribes] recognizes [the Indians' right to self-government] and the
. . .doctrine of the law of nations is that a weaker power does not surrender its independence--it's right to self-government by associating with a stronger and taking its protection.

The U.S. Constitution also recognizes tribal sovereignty by directing the federal government, rather than the states, to conduct official relations with Indian nations (2).

Tribal Self-determination and the Federal/Tribal Government-to-government Relationship

In 1934 Congress passed the Indian Reorganization Act (IRA). The IRA recognized tribal governments, reaffirmed federal recognition of the sovereignty of tribal nations, restricted the power of the federal government over Indians, and established economic development programs to assist tribes. IRA has contributed substantially to the present definition of self-government and self-determination that tribes use as a basis for seeking to be involved nfederal policy decisions. Since the 1960s, beginning with the Johnson administration, administrative policy has also reaffirmed the federal government's recognition of tribal self-determination. Such recognition of tribal sovereignty was formalized with the passage of the Indian Self-Determination and Education Assistance Act in 1973 and is illustrated by the government-to-government relationship held between the federal government and tribal governments. Within this relationship, the federal government pledges to work directly with tribes as separate governments and not as political subdivisions of states or other governmental units (1,3).

Since Johnson, succeeding presidents have also expressed support for tribal self-determination and the government-to-government relationship. President Johnson covered tribal self-determination in his 1968 congressional message. President Nixon followed in 1970. In 1983 the Reagan Administration expressed support for tribal self-determination and a government-to-government relationship. In April 1994, President Clinton issued a memorandum to the heads of executive departments and agencies that outlined the principles that define the federal government's responsibility to operate within a government-to-government relationship with federally recognized tribal governments (4,6).

The Federal/Tribal Trust Relationship

Characteristic of the unique federal/tribal relationship is the concept of "federal trust" responsibility to tribes. The U.S. Supreme Court ruled in Cherokee Nation v. Georgia (1831) and in Worcester v. Georgia (1832) that the federal government's trust responsibility is to ensure "the continued survival of Indian tribes as self-governing peoples." From this ruling, the original definition of federal trust responsibility emerged:

The United States Trust responsibility toward American Indians is the unique legal and moral duty of the United States to assist Indians in the protection of their property and rights.

Other definitions of the federal trust responsibility have been used. The Department of Interior defines the federal responsibility as the legal obligation of the U.S. government to protect "valuable Indian lands, water, minerals, and other natural resources." However, this definition is incomplete as it mentions only physical properties. In 1977 the American Indian Policy Review Commission defined the trust responsibility as "an established legal obligation which requires the United States to protect and enhance Indian trust resources and tribal

self-government and to provide economic and social programs necessary to raise the standard of living and social well-being of the Indian people to a level comparable to the non-Indian society." Despite differing interpretations of the federal trust obligation to tribes, the original purpose of federal-tribal trust has appeared in treaties and agreements, court decisions, tribal statements, and Congressional acts. This definition is most often cited despite many examples of federal actions which are inconsistent with intent of the Supreme Court (3,7).

Tribal Treaty Rights

One of the most misunderstood areas of federal Indian law are the treaties that have been negotiated between the United States and Tribal nations. Under treaties, tribes ceded certain lands to the United States, but reserved other lands and retained perpetual rights to perform certain activities on certain lands ceded to the U.S. In order to fully understand the rights of tribal governments and Indian people, one must understand how treaties have recognized the distinct identity of tribal governments. Many Americans, unfamiliar with Indian history and Indian law, oppose the treaty rights of tribes. Many people mistakenly believe that a violation of the treaty by the United States has nullified those treaties. However, Congress must repeal a treaty legislatively in order to invalidate the treaty (1,2).

There are several tribes that have rights to access natural and other cultural resources found within several DOE nuclear installations. Tribal rights to hunt, gather, fish, and engage in other traditional activities are protected by treaties that the tribes signed with the federal government. In these treaties, tribes retained access to resources within land areas that now host several DOE nuclear installations. For example, in the Treaty of 1855, the Walla Walla, Cayuse and Umatilla Tribes (now the Confederated Tribes of the Umatilla Indian Reservation or CTUIR) ceded lands to the U.S. that included what is now the eastern third of the Hanford Nuclear Reservation. As a result, CTUIR's rights regarding lands and resources within the Hanford reservation were retained under the 1855 treaty. Under the treaty, the Tribes retained the exclusive right to take fish from the streams running through and bordering the Tribes' reservation and retained the privileges of hunting, gathering roots and berries, and pasturing their stock on unclaimed lands that include the portion of Hanford that falls within the Tribes' traditional lands (5,8,9). The Yakama Indian Nation and the Nez Perce Tribe also have treaty-reserved rights to access resources within portions of the Hanford installation. In 1855, the Yakama Indian Nation ceded 10.8 million acres to the U.S. government in exchange for their present reservation and retained rights to hunt, fish, gather, and pasture stock on portions of their ceded lands, some of which are within the Hanford site. In 1855, the Nez Perce also signed a treaty with the U.S. in which they retained similar rights of access to portions of their ceded lands (5,10).

The Shoshone-Bannock Tribes of Fort Hall, Idaho are yet another example of a tribe that possesses treaty-protected rights to use resources within a DOE installation. The Idaho National Engineering Laboratory (INEL) lies on the aboriginal territory of the Shoshone and Bannock people. Before Whites settled in the area the Tribes used the lands and waters within the area that now hosts the INEL for fishing, hunting, gathering plants for food and medicine, and for other cultural purposes. In 1868 the Tribes signed the Treaty of Fort Bridger with the U.S. government which protected tribal rights to perform traditional activities on unoccupied

lands of the federal government, some of which include the area that the INEL occupies today (11,12,13,14).

Finally, for tribes to safely and effectively exercise their treaty rights to hunt, fish, and gather on these lands, the natural resources upon which these rights are based must be safe toconsume. However, activities undertaken by DOE, other governmental entities, and private parties, have harmed natural and cultural resources that are important to many tribes across the country.

The DOE and EPA Indian Policies

The U.S. Department of Energy and the U.S. Environmental Protection Agency, as well as other agencies and departments, have developed Indian policies that outline their responsibilities, as they interpret them, to American Indian Tribal governments. In 1984 the "EPA Policy For the Administration of Environmental Programs on Indian Reservations" was signed. EPA's Indian Policy recognizes tribal self-government and the federal/tribal government-to-government relationship. The EPA policy outlines nine general principles to guide the agency in managing and regulating environmental problems on tribal lands. EPA has stated that it will involve tribes in decision-making and program management that affect tribal lands and that tribes should be included as partners in environmental management in much the same way as states are involved in policy decisions that affect state lands. Five of the nine EPA Indian Policy principles address tribal government involvement in EPA decision-making. Two principles stress that state, local, and the federal government and its agencies should work cooperatively with tribal governments to aid tribes in assuming environmental protection responsibilities for reservations. The EPA policy is primarily emphasizes EPA's recognition and respect of the rights of tribal governments to protect the reservation environment and the reservation populace. However, the EPA policy is without implementation guidance that would explain to agency personnel how the policy principles apply to their work (15). Like the EPA Indian Policy, the "DOE American Indian Tribal Government Policy," released in December 1991, recognizes tribal sovereignty, the government-to-government relationship, tribal self-determination, the federal/tribal trust relationship, federal responsibilities to tribes determined by tribal political status, treaties, and federal law (16).

THREE CASES

Following, are brief descriptions of three processes in which DOE or EPA were implementing programs and determining policy and regulations. In all three processes, tribes would be affected by the decisions made by DOE and EPA, and their contractors. In all three processes there was a breakdown in communication and understanding between federal personnel and tribal representatives. In addition, in all three cases there was cultural information that was either not accessible to federal personnel or was not fully considered in federal decision-making.

DOE Consortium for Environmental Risk Evaluation

The Tulane/Xavier Consortium for Environmental Risk Evaluation (CERE) consisted of academic institutions and corporations with risk analysis and public involvement expertise. The U.S. Department of Energy (DOE) retained CERE under a cooperative agreement toperform "an independent qualitative evaluation of risks to the public and tribal health, to worker health and to the environment" arising from DOE's environmental management activities (17).

Conventional risk assessment methods do not generally evaluate exposure resulting from tribal lifestyles. For example, assumptions regarding food consumption are based on average consumption practices which may differ significantly from tribal practices. Conventional risk assessment also does not evaluate risks that are difficult to quantify, such as risks to cultural resources. Partly because of these limitations and because existing risk documents had obviously not assessed some of the crucial risks to tribes, the CERE team was not equipped to evaluate risks to tribal cultural resources. However, in the tribes' view, a legitimate evaluation of risks to tribal health was not possible without evaluating the risks to natural and other cultural resources, particularly from the transportation of radioactive and hazardous materials across tribal lands.

The evaluation of culturally-specific risks is a still emerging practice in risk assessment. Unfortunately, none of the members of the CERE risk evaluation team had experience with incorporating tribal values into a traditional risk evaluation framework. CERE focused primarily on documenting direct risks to human health. In addition, CERE was charged with performing a qualitative, rather than a quantitative evaluation, that was based on existing risk information, which did not include information on risks to cultural resources. During discussions held at CERE workshops and in a separate report prepared by Confederated Tribes of the Umatilla Indian Reservation (CTUIR), the tribes involved in the program made it clear that they considered the absence of an evaluation of culturally-specific risks to seriously limit the value of CERE's report (9).

Tribal representatives involved in the CERE program were also dissatisfied with how CERE communicated with tribes. The tribes do not believe that interactions with tribes were a priority for the CERE risk team, which they also believe had little understanding about the nature of the federal/tribal relationship. In addition to staff level communications which did not occur regularly (thereby hindering the gathering of existing tribal risk information), formal communications between CERE and the tribal governments did not occur (11,9). The CERE public involvement team did distribute a paper to its team members outlining the key political concepts and federal/tribal policies that have determined the nature of the federal/tribal relationship. This effort was recognized by the tribes as a positive step by CERE to educate its team members, but did not fulfill the need for CERE decision-makers to act as official points of contact to open up direct lines of communication between tribal government leaders and contractor decision-makers. However, the CERE risk evaluation team should not be held entirely at fault for not understanding the nature of the federal/tribal relationship and the proper protocol for dealing with tribal governments. DOE and other federal agency personnel also do not seem to understand how federal responsibilities to tribes apply to their work and the decisions they must make.

The Penobscot Nation of Maine and EPA Regulation of Dioxin Discharge

The Penobscot Nation Reservation comprises approximately 200 islands in Maine's Penobscot River, including Indian Island, the main island of residence for the tribe. Historically, and still today, tribal members use the river for subsistence fishing, hunting, trapping, and to gather fiddlehead ferns which are either consumed or sold to grocery stores. Because the reservation includes waters of the Penobscot River, and

because the river is vital to maintaining tribal traditions, the preservation and enhancement of the river and its resources are of the utmost concern to the Nation. Unfortunately, the river has suffered degradation from sawmills, textile mills, leather tanneries, and pulp and paper mills. Though water quality has improved since the late 1960's with the passage of environmental protection laws, the water quality of the river is still not sufficient for tribal members to safely consume fish at subsistence levels from its waters (18,19).

Dioxin that is present in the river and in fish tissue is one of the greatest issues of environmental concern to the tribe. Dioxin, a likely human carcinogen, is produced as a byproduct of the chlorine bleach process used to produce high quality white paper (New England Journal of Medicine, 1991). The Penobscot Nation has consistently stated that their right to fish at subsistence levels, which is guaranteed in treaties, has been impinged upon by paper mills that discharge dioxin laden effluent into the river and by the State of Maine, which allows such practices. In 1992, Lincoln Pulp and Paper, located 35 miles upstream from Indian Island, applied to EPA Region I for renewal of a five-year permit that allows the mill to discharge dioxin laden effluent into the Penobscot River at a specific concentration. Because the State of Maine did not have permitting authority under the Clean Water Act, EPA was responsible for issuing the permit using EPA's recommended dioxin standard. The State of Maine, as a "primary affected party," received special consideration as a primary commenter on the permit and was notified of Lincoln Pulp and Paper's application for renewal of the discharge permit. However, the Penobscot Nation was not viewed as a primary affected party, nor was it given the same early opportunity to comment on the permit. EPA Water Quality staff, in an interview, admitted that this oversight was due to the lack of awareness by staff of the location of the reservation, and of the traditional fishing practices of tribal members (21). Given the reservation location and tribal fishing and gathering practices, tribal members are more directly affected by dioxin contamination in the river. The fishing rights of the Penobscot Nation have been acknowledged by treaties and reaffirmed by the Maine Indian Claims Settlement Act of 1980. The Penobscot Nation should have received special consideration as a primary party and commenter on the permit. In comments submitted to EPA, Region I and to EPA Administrator, Carol Browner, the Penobscot Nation pushed EPA to uphold its trust responsibilities to the Penobscot Nation by protecting the Penobscot River from environmental degradation.

DOE's Development of Site Treatment Plans for Mixed Waste

Under the 1992 Federal Facility Compliance Act (FFCA), which amends the Resource Conservation and Recovery Act (RCRA), the U.S. DOE was required to develop andimplement Site Treatment Plans (STPs) for the treatment, transportation, and storage of mixed waste at 38 DOE sites (NGA, 1994). Several of the DOE sites occupy land areas that were ceded by tribes in treaties with the U.S. government. These tribes still have treaty-guaranteed rights to access natural and cultural resources within these sites. In meeting the requirements of the FFCA, DOE will transport wastes between sites for treatment and will transport waste treatment residuals back to their sites of origin. Many of these waste shipments will cross reservation lands. Tribes near DOE installations are concerned that DOE waste treatment activities will pose risks to tribal health and resources and tribal sovereignty (23,24).

From October 6, 1992, when the FFCA was passed, until October 6, 1995, when DOE was required to gain approval of STPs from the regulating states and EPA, DOE worked with the states and EPA to develop the plans (22). Although the FFCA did not require state involvement, DOE recognized that states should be involved because ultimately, they would approve the completed plans. The states recommended that the National Governors' Association (NGA) coordinate the states' role in STP development. DOE agreed and negotiated a cooperative agreement with the NGA. During the three years after FFCA was enacted, the NGA coordinated state involvement in the development of STPs by providing forums primarily for state, EPA, and DOE representatives to discuss technical, policy, and implementation issues related to the STPs.

However, because tribes did not possess regulatory authority under RCRA for hazardous waste, the FFCA did not require tribal government approval of STPs as it required approval by regulating states. Therefore, DOE did not have the impetus to prioritize tribal consultation in the same way as it prioritized state consultation. Consequently, tribes have expressed dissatisfaction with how they were involved in developing STPs. The tribes state that they did not have sufficient opportunity to engage in government-to-government discussions with DOE and to provide information about risks to tribal communities to be incorporated into the transportation, treatment and storage decisions reflected in the plans. During STP development, tribes were invited to attend meetings that primarily reflected the agendas and concerns of state regulators including a national FFCA meeting and NGA meetings (25). While some tribal representatives attended some of these meetings, those tribal representatives note that such meetings did not fulfill DOE's responsibility to interact with tribes on a government-to-government level. The NGA exists as a forum to bring together representatives of states. Its primary goal in the STP development process must be to facilitate effective discussion between states and DOE and to promote state influence in the STP process. While inviting tribes to participate in the NGA meetings does not in itself contradict DOE's government-to-government relationship with tribes, tribal representatives point out that in order to fulfill the responsibilities of the relationship, similar meeting should have been held between DOE's tribal governments throughout the STP development process. Such meetings would have provided tribes with opportunities to talk at length with DOE about mixed waste activities and their potential impacts on tribal communities and resources in order to influence STP decisions about how to treat, transport, and store mixed waste. Because such meetings did not occur, the tribes have view their involvement in STP development as minimal (25).

INSTITUTIONAL AND CULTURAL BARRIERS THAT HINDER COMMUNICATION BETWEEN FEDERAL AGENCIES AND TRIBAL GOVERNMENTS

Relations between tribes and DOE and EPA have improved, becoming more consistent as tribal environmental programs have emerged to work with DOE and EPA on a daily basis. However, tribal representatives continue to express frustration with federal agency actions that seem to be inconsistent with federal trust and treaty responsibilities. Tribal representatives remark that program and policy decisions are still being made and implemented without consideration of federal commitments to tribes and without incorporating tribal cultural information into these decisions that affect reservation lands.

In examining the three cases, two types of barriers emerge that hinder communication and cooperation between tribal governments and the federal government. These are institutional barriers such as an organizational infrastructure that has not developed decision criteria to guide agency staff (and contractors) in incorporating tribal values and information into technical and policy decisions. Such criteria should be based on the federal/tribal relationship and tribal treaty rights, but should also consider the needs and restrictions of tribal governments and federal agency programs. Barriers to cultural understanding also hinder communication and collaboration. These include an incomplete understanding by federal agency personnel and tribal representatives about the resource restrictions, knowledge base, and cultural tenets that determine the priorities and capabilities of each party.

Tribal representatives noted the CERE team did not communicate regularly with tribal representatives, did not communicate on a staff level, and did not communicate formally with tribal government leaders. The tribes determined that CERE did not communicate well with tribes because the team did not truly understand how federal responsibilities to tribes applied to their evaluation of risks. The CERE team was largely untrained in tribal rights, values, and government-to-government protocol. Consequently, an unfortunate, yet common pattern emerged in the CERE teams' relationship with the involved tribes. Because the team did not seem to understand how to fulfill its responsibilities to tribes, communication with tribal representatives was, for the most part, "ghettoized." This means that tribal "involvement" responsibilities were thought to rest on the shoulders of the public involvement team that was working with the risk evaluation team. However, meaningful tribal consultation on a technical level could only have been carried out by the risk evaluators. Rather, it was left to the public involvement team to write a brief section within the risk evaluation report about the risks to tribal cultural resources. This section was based on non-technical information the public involvement team had gathered from the tribes. The risk assessment community has not yet figured out how to quantify risks to tribal members resulting from exposure through traditional activities. It is likely that these risks cannot be quantitatively evaluated. Therefore, the CERE team members were handicapped in their evaluation of the risks to tribes by more then the fact that they did not understand tribal rights and values. The team also did not consist of any members with experience in considering the risks to tribes from exposure through traditional lifestyles. In addition, the team was not provided with training in the government-to-government relationship, the treaty rights of the tribes they would be working with, and they were not equipped to evaluate risks from culturally unique activities.

EPA, Region I, in issuing a dioxin permit to Lincoln Pulp and Paper, faced similar institutional barriers. EPA Water Quality staff responsible for issuing the permit, did not understand how the federal/tribal relationship determined their responsibilities to tribes. They also were not aware of basic information about the Penobscot Nation, such as the reservation location or the tribe's treaty rights regarding traditional usage of Penobscot River resources. Had EPA Water Quality staff understood the federal/tribal relationship and Penobscot Nations treaty rights, they would have designated the tribe as a primary commenter on the permit. Once they were informed of the tribal situation EPA Water Quality staff seemed very willing to use information regarding tribal

member exposure to calculate the dioxin standard for the final permit. However, the Penobscot Nation had received several grants for their activities related to monitoring the river. Therefore, it follows that pertinent tribal information was already available within the Water Quality section. However, it was obviously not used to train staff in tribal issues relevant to their work.

During the development of the Site Treatment Plans for mixed waste, DOE and the affected tribes encountered additional barriers to working collaboratively. While DOE staff may have had a general understanding of the federal/tribal relationship, they did not seem to understand how this relationship was affected by the FFCA. Although the FFCA did not mention the role of tribes (because tribes were not regulators under RCRA), this did not release DOE from its government-to-government relationship or its trust responsibilities to tribes. Although tribal governments would not officially approve the plans, tribal consultation should have remained a high priority during STP development because of the government-to-government relationship and because many waste shipments will cross reservation lands. In addition, tribes have certain regulatory rights under the Hazardous Materials Transportation Uniform Safety Act.

CONCLUSION

In all three cases, federal personnel and tribal representatives encountered barriers that resulted from differing priorities, and from their inability to communicate with each other and to share their expertise. EPA and DOE almost always have priorities that differ from the priorities of tribes. Federal personnel and tribal representatives operate within very different cultures. They are subject to the demands of constituencies that have often conflicting interests, values, and needs. Although federal agencies and departments are bound to fulfillthe federal/tribal relationship, their priorities and budgets are significantly determined by a Congress and Administration that are heavily influenced by the priorities of states and other interests that are able to exercise greater power than tribes are able to exercise.

In addition, EPA and DOE program staff and their contractors often have different areas of expertise than tribal staff and tribal government leaders. While DOE and EPA each have a cadre of staff with technical expertise in specific areas, tribes have much more limited resources and thus often hire one person to operate and entire tribal environmental program. Sometimes tribes hire several staff that possess technical expertise in specific areas. However, tribes still rely on the wealth of technical expertise held within EPA and DOE to supplement their capabilities. On the other hand, tribal staff, tribal leaders, and tribal members possess a wealth of knowledge about tribal cultural resources. Tribal staff and members know where to find these resources and how the resources are used. Therefore, they know how tribal members might be exposed to contamination. Tribal staff and tribal members know why these resources are so important for maintaining traditions and they understand the implications for the tribe if such resources were to be damaged or completely lost. DOE and EPA must rely on this expertise to supplement their understanding of the risks to tribes and tribal resources from federal activities.

While tribes have had little input to federal agency priorities, this situation does not have to continue. Tribes are working diligently to develop environmental program capacity. In doing so, they are developing technical and policy expertise and are gaining regulatory and political

experience that will enable tribal environmental protection specialists and tribal policy experts to wield greater influence over policy and regulatory decisions so that tribal interests are protected.

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DEVELOPING NEZ PERCE TECHNICAL EXPERTISE IN WASTE MANAGEMENT AND ENVIRONMENTAL RESTORATION ACTIVITIES

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ABSTRACT

Since the inception of the Nez Perce relationship with the U.S. Department of Energy, and its regulating counterparts, the U.S. Environmental Protection Agency and the Washington State Department of Ecology (the signatories to the Tri-Party Agreement), the Nez Perce Department of Environmental Restoration and Waste Management (ERWM) has been building its technical expertise. The ERWM Department experienced the immediate need to develop this expertise when confronted with massive technical studies of impacts, and plans for remediation and restoration, of those impacts to areas considered critical to Nez Perce Treaty rights (Treaty of 1855). The Nez Perce Tribe has recognized, however, that maintaining this level of technical effort depends upon a sustained level of political support. The brief history of the Nez Perce/Federal relationship (and the Federal/Tribal relationship in general) leads the Nez Perce leadership to seek to strengthen political support, highlight economic justification, and market its technical excellence. The Nez Perce Tribe is aggressively pursuing contracts in areas critical to protection of Nez Perce Treaty rights at the Hanford site. Within the two years that the Nez Perce ERWM Department has been funded under the cooperative agreement with the Department of Energy, it has built a technically credible team, is seeking contracting opportunities in the global arena, has just graduated its first engineering student/intern,

and is seeking ways to improve environmental management decision-making based on the Nez Perce traditional worldview and practices.

INTRODUCTION

The Nez Perce Tribe is involved with the U.S. Department of Energy and its environmental management activities at the Hanford Site in south-central Washington through the legal doctrine of trust responsibility by virtue of the Treaty of 1855, signed by the Nez Perce Tribe and the federal government. Under a cooperative agreement between the U.S. Department of Energy and the Nez Perce Tribe, the Tribe is actively participating in issues regarding the clean up of radioactive and chemical contaminants that were a result of nuclear weapons production since the 1940's.

What follows is a brief history of Nez Perce involvement at Hanford,on the technical front. Where this paper leaves off is the beginning for our second paper on translating the traditional Nez Perce worldview into environmental management strategies.

The Nez Perce Tribe specifically retained customary usage rights to the Columbia River when the Tribe ceded over 11 million acres to the Federal Government in the Treaty of 1855. By entering into The Treaty of 1855 and subsequent treaties, the Federal Government has acknowledged that our Tribal Rights preceded the U.S. Constitution and are retained with regard to the Columbia River. The disposition of more than 100 million curies of radioactivity into the Columbia River, since Hanford operations began, has diminished the value of our retained rights without consultation or compensation.

TECHNICAL INVOLVEMENT

The Nez Perce Tribe's Department of Environmental Restoration and Waste Management (ERWM), formed in 1992, is involved in reviewing regulatory documents and suggesting implementation strategies in a number of areas under various regulatory authorities and Departmental orders. This involvement in Hanford cleanup activities is a way for the Tribe to keep tribal members informed and to build technical expertise in remediation and restoration procedures. Nez Perce involvement at Hanford is also the way by which treaty rights and cultural and natural resources are protected. The Tribe's primary focus is stopping further contamination to the groundwater and the Columbia River. The Tribe is also concerned about protecting native shrub-steppe habitat.

There are over 1,500 contaminated sites at Hanford that have been grouped into 78 operable units according to contaminants and location. Cleanup procedures are guided by three regulatory programs: National Environmental Protection Act (NEPA), Comprehensive Environmental Response Compensation and Liability Act (CERCLA), and the Resource Conservation and Recovery Act (RCRA). The guidelines indicated by NEPA, CERCLA, and RCRA set the framework for clean-up procedures and include similar stages such as data gathering for risk characterization, remedial action and design plans, and monitoring and surveillance activities. DOE's federal trust responsibilities require that the Nez Perce Tribe is included in all aspects of this work before comments about remedial actions are accepted from the public.

The Tribe's monitoring of and participation in clean-up activities centers on six primary areas: 1) Waste Management and Environmental Restoration, 2) Cultural Resources, 3) Human Resource Development, 4) Technology Development, 5) Public Education and Information, and 6) Health and Emergency Response. The ERWM has a technical staff with

expertise in policy analysis, geology, hydrogeology, biology, agricultural engineering, cultural and natural resources, communications, health, and pollution prevention. The technical staff are active participants on several DOE and Hanford related advisory committees including the Hanford Health Information Network, Hanford Advisory Board, Columbia River Impact Assessment Workshops, Community Leaders Workshop Forum, and Natural Resource Trustee Council.

In addition to the technical staff, the Nez Perce Tribe has an active internship program which provides Nez Perce students with educational opportunities and access to job training. Student interns have helped conduct research on DOE policies, proposal development, health issues, botanical collecting and revegetation, and cultural resources issues.

What follows is a select list of projects the Nez Perce ERWM Department is currently involved in. The select list was chosen to reflect a combination of priorities, demonstrable expertise, and practical solutions. To this end, the Tribe reviews Hanford Operations which directly and indirectly affect ground water and the Columbia River.

The U.S. Government operated eight "single-pass" (open-coolant) reactors along the Columbia River from 1944 to 1971. During this time period, the Hanford Site released more than 100 million curies of radioactivity into the Columbia River and large quantities of chemicals into the soil column surrounding its reactors and spent nuclear fuel reprocessing plants. Additionally, fuel slug ruptures, chemicals added to the cooling water, and neutron activated particles and debris from reactor purging(cleansing) operations entered the river. Riverbed sediments and floodplain soils of the Hanford Reach constitute a sink for many of the pollutants released to the environment by Hanford's operations.

WASTE MANAGEMENT AND ENVIRONMENTAL RESTORATION (EM)

What has been the fate of the 100 million curies discharged to the Columbia River? Most of that radioactivity was very short lived and rapidly decayed away. Studies based on random sampling of locations along the Columbia River from Hanford to the coast are cited by DOE-RL to indicate that radionuclides do not pose a significant human health risk. However, random sampling of locations along the river ignores hydrodynamic characteristics of the river and the particle size of radionuclides discharged into the Columbia River. There are areas along the river where radionuclides will preferentially settle out. Declassified documents suggest, that during Hanford Operations, 98% of particles associated with fuel rod ruptures were less than 30 microns in size. Water velocity, river bottom roughness, particle density, shape, and size dictate sediment deposition. This suggests that radionuclides could preferentially concentrate in sediment deposits in the silt-size range.

Any shoreline activities which effect the flow of the Columbia River risk the remobilization of contaminants entombed within river sediments. For example, erosion is now occurring on Locke Island caused by a change of flow within the river. This change may be due to landslides in the White Bluffs, which result from irrigation tail-water. Even today, it is possible for the public to come into direct contact with neutron activated particles (cobalt-60) in the river sediment. On June 7, 1995, employees of the U.S. Department of Energy and its contractors, EPA, Washington State Departments of Health and Ecology met to decide on protocol to determine the risk associated with cobalt-60 particles known to be on D Island in the river. The Nez Perce Tribe is concerned that

contaminants left in place at depth in the soil column after the Hanford Site's remediation will result in further degradation of the Columbia River. If irrigation occurs on lands north of Gable Mountain and Gable Butte, the subsequent rise of the water table under the 100 areas could potentially remobilize contaminants and provide a pathway for contaminants to reach the Columbia River. A "Wild and Scenic Designation" of the Hanford Reach will help to protect public health and the environment by preventing the remobilization of contaminants entombed within the river's sediment and the shoreline's soil column.
Chromium polluted ground water at levels toxic (greater than 11 ppb of hexavalent chromium) to salmon redds (salmon nests) is entering the Columbia River. The pore-water sampling program confirms that hexavalent chromium is present in the gravel beneath the Columbia River. Salmon redds are present in these gravel areas. Sediments in areas near D reactor and H reactor have been sampled and have been shown to contain levels greater than 11 ppb of hexavalent chromium. The Nez Perce Tribe ERWM has proposed that DOE consider pumping ground water from the areas of highest hexavalent chromium concentration, treating this ground water, and then injecting this treated ground water in well bores along the bank of the Columbia where toxic ground water is entering the river. The injection of treated groundwater in the bank of the river could create a hydraulic barrier which would prevent concentrated ground water from reaching the Columbia River. Nez Perce ERWM staff also is concerned about the chromium plume associated with N reactor.

Cultural Resources

Another Hanford restoration project where the Nez Perce have been active participants concerns the restoration and revegetation of the former Environmental and Molecular Sciences Laboratory (EMSL) site. Human remains which were believed to be those of Indians were discovered during preliminary construction activities in 1994. As required by law, all construction on the project was postponed for 30 days during which time representatives from the four affected tribes developed recommendations on how to proceed. The Nez Perce Tribe ERWM agreed with the other tribes that the tribal members should be left where they were originally interred. As a result of these findings the EMSL site was moved to another location. The former EMSL site was recontoured and a native revegetation effort was undertaken. With funding from DOE, the four affected tribes established a native locally-adapted seed bank and nursery in 1993. Native plants such as sagebrush, rabbitbrush, buckwheat, Indian rice grass, and needle and thread grass have been successfully reintroduced on some areas of the EMSL site. These revegetation activities are planned to continue until 1998.

Technology Development

The Nez Perce ERWM have recently reviewed a sonic drilling method which is a relatively new technology used in FY 95. Wells drilled in FY95 used this method but the technology failed to meet the design criteria as substantial formation damage occurred during drilling and/or completion. The following problems were encountered when using this method:

1) Formation damage.
2) Inability to collect samples with a representative grain-size distribution.
3) Inability to collect undisturbed cores.
4) Possible formation damage increase with increasing hole-size.

5) The drilling method may not produce waste minimization a valid claim beneath the water table.

Mitigation

The Nez Perce advocate including the costs and plans for habitat restoration in the budgeted cleanup activities. Restoration activities include site characterization, revegetation, mitigation, and monitoring plans. The Nez Perce Tribe and other affected tribes have had direct involvement in restoration activities of the Environmental Restoration Disposal Facility (ERDF), a Hanford project. The ERDF is an inland disposal facility that will eventually be filled with contaminated material that is being removed from waste sites along the Columbia River. Native plants were salvaged prior to construction of the ERDF and transplanted to areas on the north slope of the Columbia River that had been disturbed as a result of cleanup activities. Since the ERDF site was constructed on undisturbed sagebrush habitat, the Nez Perce Tribe is also actively involved in providing input to the ERDF Mitigation Action Plan. The Nez Perce Tribe is also actively participating in other Hanford related issues which includes land use planning, protection of state and federal plant and animal candidate species, participation in cultural and ecological surveys, Columbia River impact assessment concerns, groundwater contamination, and review of focused feasibility studies and proposed plans.

As a result of these actions, the Nez Perce Tribe can claim that it is one of the key players that help DOE-RL save millions of taxpayers dollars. Although the Nez Perce Tribe has not always been properly consulted with, the Tribe is proud to say that it has played an important role.

The benefits of these efforts will help the Nez Perce Tribe in forming and promoting a tribal environmental restoration team that could be developed as a private enterprise for future Hanford remedial activities. This example of combining environmental justice with environmental restoration technology transfer could be a positive example that can be replicated at sites across the nation.

Public Education and Information

In 1983, the Nez Perce Tribe was found to be "affected," as defined by the 1982 Nuclear Waste Policy Act. A year later, the Tribe was funded to start the negotiation under the Consultation and Cooperative Agreement. The negotiation occurred in May 1987 and the Tribe was involved with the Basalt Waste Isolation Project (BWIP), in Hanford, a deep geologic repository project. Along with the stoppage of BWIP, funding for the Nez Perce also ceased. DOE created Environmental Restoration and Waste Management (ERWM, later changed to EM) in 1990, and the affected status of the Nez Perce Tribe was re-affirmed a year later. In 1991, the Nez Perce Tribe applied for a planning participation grant, which was funded in 1992, thus creating its ERWM Department. Since then, ERWM staff and its technical consultants provided technical and cultural comments and recommendations to many documents and activities that have been requested by DOE-RL. Concerns and issues of the Nez Perce Tribe include regulatory status, cultural or natural resources management, transportation emergency responses, risk assessment, Tribal religious freedom, technical/management capabilities, and others. On technical related issues at DOE-RL, the Nez Perce Tribe is a member of the Hanford Advisory Board (HAB), the Hanford Natural Resources Trustee Council (NRTC), and the Site Technology Coordination Group (STCG). On the regional and

national level, the Nez Perce ERWM is a member of groups such as DOE-HQ's Community Leaders Network (CLN), Stakeholders Working Group (SWG), and the Interstate and Regulatory Technology Working Group (IRTWG), and the State and Tribal Government Working Group (STGWG). The Nez Perce Tribe has regularly and consistently sent its technical representatives to meetings, conferences, and workshops which are sponsored by DOE, American Nuclear Society (ANS), Weapons Complex, US Environmental Protection Agency (EPA) Western and National Governors Association (WGA/NGA) and private agencies. It has also sent representatives overseas to visit a nuclear facility in Sellafield, England, and attended the 5th International Conference on Radioactive Waste Management and Environmental Remediation (ICEM) in Berlin, Germany and Global '95 in Versailles, France. Nez Perce ERWM have presented papers in WM'94 and WM'95, and Global '95. Currently, ERWM staff expertise encompasses geophysics, geology, bioresource engineering, statistics, wildlife biology and environmental science. With this expertise, the Nez Perce Tribe plans to contract cleanup related activities which will further its goals of building a scientific/technical infrastructure while protecting its treaty retained rights.

CONCLUSION

In the few short years that the Nez Perce Tribe has been officially involved at the Hanford site, the U.S. Department of Energy funding has allowed tribal participation to reach the necessary level of technical and program sophistication that is currently producing cost savings and building a stronger regional economy. The Nez Perce Tribe is aware of the possible funding shortfalls and makes no mystery of the fact that the Nez Perce ERWM Department is moving into the contracting arena. The Nez Perce ERWM Department is adding value to every federal dollar spent by transitioning some of its functions into the marketplace, by building an educated and technically capable workforce, by improving coordination of efforts on site, by piloting an effective tribal participation core program, and by offering new insights to environmental management practices and strategies.

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TRANSLATING THE NEZ PERCE WORLDVIEW INTO ENVIRONMENTAL MANAGEMENT STRATEGIES

by
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ABSTRACT

The Nez Perce Tribe has developed the technical, scientific, and management expertise to offer input, in close government-to-government consultation, recognizing the federal trust responsibility to indian tribes, on many different levels of a national nuclear weapons contamination cleanup program, After a few short years of digesting the ongoing environmental management practices of the U.S. Department of Energy and its contractors, the U.S. Environmental Protection Agency, and the Washington Department of Ecology, the Nez Perce Tribe Department of Environmental Restoration and Waste Management is ready to offer its perspective on developing strategic directions and goals for the cleanup of the Hartford reservation in eastern Washington. The paper introduces the Nez Perce worldview and moves on to develop a concrete example of how

that worldview can be incorporated in setting cleanup goals and endstates. This paper examines the efficacy of developing values-based mission statements and performance measures to provide for cleanup, protection of Nez Perce treaty rights, continued practice of traditional Nez Perce activities, and multiple uses for the local non-Indian population, The paper concludes by showing the benefits of taking a landscape approach, in close consultation with the Nez Perce Tribe, to site-wide integration of remedial actions.

INTRODUCTION

The Nez Perce Tribe has been involved in nuclear waste issues since 1987 and the Basalt Waste Isolation Project (BWIP). Nez Perce involvement has always been somewhat constrained by the level of funding and the enormous task of wading through Department of Energy documents. Since 1992, however, the Nez Perce Department of Environmental Restoration and Waste Management (ERWM) has been attempting to define its own agenda. We are moving beyond document review, and into shaping strategic policy directions. One such strategic policy direction is incorporating a Nez Perce worldview into environmental management practices, from risk assessments to technology development. This paper will introduce the Nez Perce worldview and how that worldview and accompanying values shape the Tribe's environmental management decisions. We will also touch on how the value-adding practice of incorporating this worldview might help others' making environmental management decisions.

Today, as the Nez Perce ERWM Department matures, we are defining our own path in Nez Perce terms that others can understand. This path follows the same direction as those of the Tribe's traditional past, however we are using radically different technologies to help us along this different new path. We feel pressure to develop this new path from internal sources as a way of incorporating new, and necessary, technologies into our lives in a way that minimizes the cultural impacts. We are also developing this new path for external consumption. People and governments are making decisions that affect our lives. We are developing processes to inform strategic directions in environmental management that incorporate Nez Perce values, in the hope that such processes and directions will find greater acceptance. This education through process development has the added benefit of introducing Nez Perce values, making them less "mystical," doomed to be forever too "ethereal" to be the basis for any serious decision making.

We cannot expect anyone to understand Nez Perce values and interests as the Nez Perce do, however, we can translate, or transliterate, these concerns into the larger society's lingua franca to ensure we are not marginalized. But more importantly, we must transliterate this worldview for the larger society, because with our help, we believe that the United States, and the world, can achieve more fundamental environmental protection with less red-tape in a shorter period of time.

VALUES, RIGHTS, AND CONCERNS

The Nez Perce people have been here since time immemorial and think about the consequences of their actions in terms of hundreds if not thousands of years. While planning that far in advance may seem impossible, it reveals a culturally conservative worldview, one that attempts to implement a survival plan based on thousands of years of collective wisdom. This wisdom is passed down from generation to generation in a context that brings the past alive and encourages thoughtfulness for generations to come.

Before a child could walk, he was taught cultural and traditional values, the parents knowing that experience was the greatest teacher began teaching their children to honor and respect the properties of Mother Earth, as provided by the Creator. The children were taught that these properties of Mother Earth, the land, animals, plants and forests were never owned, but to be borrowed and used for their subsistence and they were to give thanks for their use. Nez Perce children were taught to always leave/give something in return to replace what had been taken. For centuries, each year the bands, families, friends and neighboring tribes gathered in the springtime for a thanksgiving feast called, "Ka-oo-yet." Giving thanks meant sharing salmon, roots, and traditional foods as a tribute to Mother Earth and the Creator. Thanks was again given in Autumn for the harvest of traditional foods. This practice continues today, however the salmon are endangered, traditional foods and berries are becoming scarce, and access to traditional use areas are increasingly limited. Maintenance of these traditions and values ensures the protection of resources through reverence for life and humility before creation. This reverence for life and humility, for the Indian, is inseparable from the identity of individuals, families, bands, and tribes.

The contrasting view of environmental protection found in the non-Indian world of the United States frustrates many Indian people because it is media specific, contaminant specific, and divided into human and ecological risks, administered through "stove-piped" programs and funding. The contrasting, reductionistic view of problems and solutions is evidence of a different worldview, a worldview that does not necessarily lend itself to environmental protection easily. The Nez Perce worldview translates into practice through time consuming, and often contentious dialogue. This dialogue is conducted within a set of given cultural parameters. The introduction of new ways of understanding threats and solutions (risks and technology developments) in the arena of nuclear weapons cleanup is both a cultural and democratic challenge.

OPEN DECISION PROCESS

How questions are asked is sometimes almost as important as the questions themselves. A tribal dialogue on plutonium disposition, for example, cannot take place without presenting the technical and political considerations against the array of tribal legal, cultural, political, and natural resource considerations. Developing this complex matrix of concerns to arrive at an answer can be a daunting task, but questions such as plutonium disposition must be considered by the Tribe. Crafting an answer might be a time consuming process, but the longer it is discussed, the more it is related to other daily tribal concerns, the more it will become part of Nez Perce reality. The long-lived nature of plutonium and other radioisotopes requires the Tribe to incorporate these concerns into the collective Tribal experience. Lessons learned from this question and answer exercise can be transferred to non-tribal society as an example of establishing low-tech longevity of information within a society.

The first step to opening this dialogue must be based in the traditional Nez Perce value practice of seeking an answer respectfully and with a commitment to honor the answer. We must carefully ask the whole question of plutonium disposition in a way that shows respect that we have for those we are asking. A carefully asked question will hopefully elicit a carefully considered answer. Perhaps the Tribe will not be willing to

share the whole answer(s), or may not give the answer the U.S. wants to hear, or may even give an ambiguous answer(s) that can stimulate further dialogue that can lead the U.S. to a better solution. In any case, it is a question we must ask of our Tribal public, and one that will prove to be useful, not only for the example of incorporating tribal values into decision-making processes, but also in how tribes create and sustain open and democratic processes.

For the Nez Perce Tribe formal decisions are made within a constitutional structure; however, there is more to forming that opinion than the simple voting at General Council, or by resolution from the Nez Perce tribal Executive Committee (NPTEC). The way decisions are made, through the various formal and informal mechanisms, constitutes the Nez Perce decision making process. The scope of the decision making process varies from decision to decision; however, in attempting to answer a question such as plutonium disposition the Nez Perce ERWM Department is not only looking for the answer. It is also attempting to give the process greater transparency for the benefit of the Nez Perce people, and for the benefit of our partners in the field of environmental management.

The formal structure of the Nez Perce decision making process is found in the 1961 Constitution, as revised. The Nez Perce tribe has an elected body consisting of nine members called the Nez Perce Tribal Executive Committee (NPTEC). This governing executive committee is responsible for the protection of Tribal sovereignty, including protection of Tribal rights in Nez Perce ceded territory and areas over which the Nez Perce exercise off-reservation treaty rights, and promotes the health, education and welfare of the Tribe. The NPTEC members are elected from the Tribal membership by the Nez Perce General Council, which includes all eligible voting Tribal members. Both the General Council and the NPTEC are governed by the Revised Constitution and Bylaws of the Nez Perce Tribe. The Constitution and Bylaws describe the duties and responsibilities of the NPTEC and General Council. Any enrolled Nez Perce Tribal member may address any issue before the NPTEC or the General Council.

Prior to the adoption of the Constitution and formation of the Tribal Council, all issues were discussed before a Tribal General Council meeting, called for that specific purpose. The General Council Chairman, Vice-Chairman, and Secretary were elected by special election. The meeting, called by the Chairman, lasted no more than two days to discuss the issue at hand. This early General Council system resembled a council of elders, as they discussed Treaty rights, traditional practices, and cultural impacts before making a decision. Today, the General Council is attended by both young and old, but special committees are usually reserved for elders to research issues and evaluate impacts to culture, traditional practices, and treaty rights. These special committees make recommendations to the General Council and these are then often forwarded to the NPTEC for final resolution. This process ensures direct democracy, and several cultural and political screens that will make a decision "Nez Perce."

SOVEREIGNTY: DUTY TO NEZ PERCE COMMUNITIES

In conjunction with the Nez Perce duty to protect its treaty rights, the environment, culture, and welfare, is the duty to educate its members and neighboring public to its activities. The Nez Perce Tribe assumes many different roles and therefore has the responsibility to protect and enhance these relationships in a uniquely Nez Perce way.

The Nez Perce Tribal is a governmental entity with certain powers and authorities derived from its inherent sovereignty, status as land owner, and delegations from the Federal government. The Nez Perce Tribe exercises its powers and authorities to serve its members and to regulate activities occurring on the reservation. The Nez Perce Tribe is a cultural entity, accordingly charged with the responsibility of protecting and transmitting that culture which is uniquely Nez Perce; the Nez Perce tribe is also the beneficiary within the context of federal trust relationship, and consequently has obligations to other Indian Tribes.

The U.S. Department of Energy planning activities touch, in some way or other, each of these roles of the Nez Perce Tribe--and particularly so when the Tribe acts in its cultural, treaty, and trustee roles. The understanding iterated in this paper reflects the diverse interests and responsibilities of the Nez Perce Tribe.

Nez Perce vis--vis the Department of Energy

The relationship between the Nez Perce Tribe and DOE around the planning issues is defined by the trust relationship that exists between the federal government and the Tribe, by the DOE American Indian policy, and by the mutual and generally convergent interests of the parties in the efficient and expeditious cleanup of the Hanford site and other areas. For example, the Nez Perce Tribe sees itself not only as an advisor to DOE, but also as a previously untapped human resource pool of scientific and engineering personnel that will be available to assist the Department in the 21st Century. The Nez Perce Tribe also sees its members as a pool of technically trained and certified labor force for environmental restoration and decontamination and decommissioning work. The cooperative agreement between the Nez Perce Tribe and the U.S. Department of Energy outlines an approach that will integrate these and other roles into a comprehensive Nez Perce-DOE program.

Relationships with Tribal Members and the Reservation Community

The Nez Perce Tribe views its primary responsibility toward its members as protection of Tribal treaty rights. However expansive that responsibility may be, Tribal leadership would be remiss if that were the only focus of the proposed relationship between the Nez Perce Tribe and DOE. The Nez Perce Tribe also has an obligation to educate its members and the reservations community of its and DOE's activities at Hanford. This will mean reaching into the Headstart Program; elementary and secondary schools; and the Tribal reservation communities and general public. The Tribe also must examine the employment and economic and business development opportunities presented by the cleanup at Hanford and must create programs that will enable Tribal members to realize these opportunities.

Relationship with Non-Human Community Members

Native American healing practices have been historically misrepresented, either through ignorance or with exploitative intentions, however, as an aspect of the Nez Perce duty to our non-human community it seems necessary to explain, at least partially, what this relationship is. For centuries the Indian has used natural resources and borrowed from the land for his subsistence, taking roots, berries, and herbs for foods and medicines. The animals, birds, and fish were his clothing and food. His knowledge of the land and the resources it provided gave him the power and control of traditions, culture, and religion of his choosing. He was taught by his elders and parents from early childhood and eventually

became superlative in the use, knowledge, and care of natural/cultural resources. These powers were enhanced when and if an individual received a vision, or "wa-ya-kin," thus gaining knowledge and spiritual support from an animal or thing. These special powers were recognized by his people throughout his lifetime, usually by taking the name of the animal or a piece of the animal that he wore as part of his apparel (i.e. "rabbit-skin leggings"). In addition to the spiritual knowledgeof the special animal, the individual became versed in the different types of herbs, roots, barks, leaves, and berries to be used as medicines that could be taken by eating, drinking, smoking, application to wounds or the skin, or by other means. Application of the medicines was usually accompanied by chanting, singing, rattles, and sometimes dancing, which actions were requests for assistance from the spirits of the special plants and animals.

Relationships with Other Indian Tribes and Tribal Organizations

The Nez Perce Tribe views its role within the DOE planning context as operating on two levels. The first, and most important, is obviously the Tribe's role as advisor to the site-specific cleanup of the Hanford site. However, the Tribe realizes that its consultation with DOE will be viewed by both the Department of Energy and other Indian Tribes whose interest in the DOE environmental management planning process are not as compelling, as representatives of American Indian Tribes in general. This is not an unrealistic concept. The fact is that the Tribe must prepare and act on these two levels. It must carry out the oversight, monitoring, and analysis necessary to be a strong advocate for Tribal rights at the Hanford site and must be able to extrapolate from the site-specific analysis the generalized concerns that are relevant to American Indian Tribes across the country. We believe, the transportation issues and planning for human resource development fit into this category. The Nez Perce tribe believes that it can and must operate on these various levels to act as a credible representative of general Tribal interests in environmental management decisions.

With this introduction to the values, decision making process, and responsibilities to the various communities of the Nez Perce Tribe we move on to a discussion of basing environmental management decisions on more project-specific information.

DEVELOPING VALUES-BASED MISSION STATEMENTS AND PERFORMANCE MEASURES

The current Hanford mission statement ("cleanup Hanford") is really a "means" goal rather than an "ends" goal. Hanford is being remediated for reasons in addition to the fact that cleanup is good. However, site-wide target endstates have not been defined, and therefore performance measures that apply during remediation have not been clearly defined. The dangers of beginning massive cleanup before endstate goals have been defined should be readily apparent. For example, proceeding with excavation before defining the target endstate could result in permanent loss of habitat, whereas initially setting an environmental management goals might result in the selection of remedial technologies that are less intrusive. It is not sufficient to begin excavation on the assumption that eventually it will become clear when to stop, and that appropriate restoration measures can be designed at some time in the future when endstate land uses have been clarified. Similarly, a temporally phased cleanup might result in initial cleanup to brownfield standards, thus allowing industrial development in an area later targeted for ecological preservation. Both of these examples violate the decision

principle of preserving future options for natural resource management and multiple land uses.

If a higher site-wide goal, such as "preserve the eco-cultural landscape," is articulated early in the decision process, then several things happen. First, agreement about higher-level goals may reveal commonalties among the Nez Perce Tribe and other potential endstate "user groups" that were formerly unrecognized. Second, the list of technical options could likely increase. Third, conflicts about each isolated action may diminish if site-wide, holistic, values-based goals are developed and enforced. Fourth, both external groups and internal program managers will be able to relate individual actions to real progress.

Scope of Analysis Required to Support Broad Management Goals

Strategic planning efforts frequently fail to recognize that the choice of a strategy can significantly influence the endstate. For instance, a narrowly defined risk reduction strategy automatically relies on certain types of information and excludes other types. Conventional environmental safety and health (ES&H) strategies typically focus solely on reducing human exposure, with only minimum attention to environmental and ecological goals, and no attention to cultural values-based goals (Fig. 1). If reducing (or avoiding) human exposure is the sole risk performance measure, then other types of risk (accident probabilities, probability of ecological impacts, probability of cultural impacts, and so on) may not receive adequate attention of budget. Programmatic risk may actually increase if performance measures are too narrowly defined and if the decision logic fails to identify uncertainties in the path forward. Decisions are more stable (i.e. technically defensible and politically acceptable) if they are linked to broad management goals; this in turn requires a broader (but not necessarily data intensive) information base. There is much more to health than just the absence of exposure, and the Nez Perce identify at least three types of health: physical, spiritual, and emotional. Further, there is more to risk reduction than just reducing the probability of adverse human health effects. One approach that is inherently more suited to indigenous values and holistic perspectives is Comparative Risk (Fig. 2). Comparative Risk (7,8) is clearly more comprehensive than human exposure-based assessments. It also captures Quality-of-life concerns, which we have termed "Culture" to include concepts about Nez Perce culture and the ability to practice traditional activities. The three types of impacts (health, environment, and culture) together give an indication of the impacts to, or health of, holistic Circle of Life. The Circle of Life, reflects the culmination of indirect health effects, the conclusions of community-based knowledge on how contaminants affect the community's cultural identity, knowledge of elders about what natural resources really mean to Nez Perce people, and inherent rights of all species (modified from Cole, (2)). In Nez Perce reality, the Circle of Life also reflects the concept that there is a single "essence" that extends to the bounds of the Earth, that every natural thing originating from the Earth represents a local or mobile manifestation of that essence, that each thing is connected to every other thing through the shared essence, that therefore harm to any of those components also harms the whole.

Fig. 1

Fig. 2

When assessing risks in practices, a tribal risk model might incorporate information about culture-specific foods, medicines and other materials

obtained from the study area, provided that the assessment is designed to avoid the necessity of revealing confidential information. In fact, most standard exposure equations cannot truly use this information in the first place, because species-specific uptake factors are usually unavailable. Therefore, careful assessment design can adequately reflect exposures that might reflect exposures that might occur during a set of traditional activities without compromising confidentiality. Similarly, ecological species of particular cultural importance will likely be different from threatened and endangered species, but artificial (computer) foodweb models that list multiple species but lack species-specific contaminant transfer factors will limit the usefulness of gathering sensitive or confidential information.

The importance of including the full list of stressors in the evaluation cannot be overemphasized. Because decisions about environmental remediation or waste management are usually focused on chemicals and radionuclides and the environmental media or facilities that contain them, other stressors may be excluded from the decision process. For example, Hanford habitats and treaty-reserved rights are at just as much risk from political (zoning, boundary creep, reduced budgets), legal (denial that treaty rights exist, adverse land use designations), institutional (closed decision processes, denial of access to traditional lands and resources), physical (expanding infrastructure, habitat fragmentation, clean fill mining) and aesthetic (noise, visual impacts, shrinking buffer zones around sacred sites) impacts as they are from contamination. If this is recognized early in the decision process, it is more likely that values-based mission statements will adequately reflect Nez Perce concerns. If it is not recognized, then decisions typically become technical problem statements that lose direct links to values and may ultimately be rejected or cause costly mid-course project corrections.

TRANSLATING TO ENVIRONMENTAL MANAGEMENT STRATEGIES

The last step in the planning process is to evaluate alternative means of achieving each target land use; since the target endstates were clearly defined in the previous step, alternative solutions can be more precisely selected. In addition, cost, schedule and worker exposure during remediation are modifying factors in project implementation rather than insurmountable obstacles that might automatically preclude further consideration of the preferred endstate.

At Hanford, a common misperception is that traditional indigenous land uses, habitat preservation and general recreational uses (e.g. bird watching, fishing, hunting) are entirely different endstate designations that are mutually incompatible. A related misperception is the "pristine or nothing" label applied to tribes when they push for maximum remediation and restoration. Ultimately, the Nez Perce Tribe has a trust duty to their members to protect treaty rights, human health and natural resources. However, once this is acknowledged as the ultimate goal for the site, then a wide range of mutually-compatible activities is possible during interim phases of cleanup.

CONCLUSION

The nuclear waste cleanup experience gained by the Nez Perce tribe continues to grow and gain technical sophistication. The Nez Perce Environmental restoration and Waste Management Department is involved in a number of remediation, environmental restoration, technology development, and waste management activities. After a few short years of

digesting the current processes and schools of thought involved in these various activities, the Nez Perce Tribe is attempting to synthesize these "technologies" with its values, offering an alternative way of conceptualizing one path to holistic environmental management goals. The Nez Perce Tribe subscribes to the theory that close consultation is an educational tool for everyone involved. Through this educational process the Nez Perce Tribe offers to share its collective wisdom and practical advice, obtained through thousands of years of trial and error, and maintained by culturally conservative practices. The Nez Perce Tribe has a duty to contribute to solutions for the problem encountered in the filed of environmental management. In return, the Nez Perce Tribe expects that through this educational process of consultation others will begin to grasp the full scope of humankind's duty to the environment.

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Session 22 -- ACADEMIA/GOVERNMENT/INDUSTRY PARTNERSHIPS FOR EDUCATION & TECHNOLOGY

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WERC/FAST INITIATIVE

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ABSTRACT

The WERC/FAST Initiative embodies the merits of two successful programsthe Waste-Management Education and Research Consortium (WERC) and the Fast-track Advancement of Significant Technologies (FAST). In the accomplishment of its mission, WERC hosts an annual environmental design contest, the only environmental process design competition of its kind in the United States that provides university participants with design, as well as practical, experience on real environmental problems. The intent of FAST is to leverage a small amount of funding from the U.S. Department of Energy (DOE) to make potential users at DOE sites more comfortable with existing technologies to solve specific problems by providing quality treatability testing to document performance. This synergistic relationship of the two programs concentrates on moving promising technologies selected during the WERC environmental design contest to a field-ready level. DOE, through the Office of Environmental Restoration, has tasked the FAST Program to provide financial support to the universities selected in the WERC competition to perform treatability studies and performance testing. This testing is conducted in cooperation with WERC and the DOE site sponsors of the problems.

INTRODUCTION

Many mechanisms exist for the promotion of environmental technology, particularly those that are considered to be "innovative." The U.S. Environmental Protection Agency (EPA) has its SITE (Superfund Innovative Technology Evaluation) program, the VISITT (Vendor Information System for Innovative Treatment Technologies) database, and numerous publications to spread the word about environmental technologies. The U.S. Department of Defense (DOD) has prepared technology screening matrices, publications on technology demonstrations, and environmental management "good news" stories. The U.S. Department of Energy's (DOE's) technology development program issues programmatic and site-specific summaries of technology demonstrates and related projects, has conducted workshops to promote technology transfer, and has distributed technology information in a "baseball card" format. Despite all these efforts, technology vendors and providers continue to complain that their technologies are not used and environmental restoration problem holders continue to complain that existing technologies are inadequate to fulfill their needs.

Some of the barriers that hamper technology use include the following factors:

Sources of technology information, such as publications and databases, are not always reliable because much of the information is supplied by the technology vendors and providers to promote their products or services and often do not include adequate data on performance and cost.

The belief that just because a technology worked at one site, it won't necessarily work at another (i.e., the "my-side-is-different" syndrome).

DOE sites, used to doing everything in-house, sometimes suffer from the "not-invented-here" mentality.

Decisions to use a particular technology are often made by committee (e.g., EPA, State, community, and responsible party) or require adherence to a stringent permitting process. These conditional factors can contribute to second guessing about the acceptability of a technology.

Those responsible for cleanup of environmental problems often want a "risk-free" solution.

The agencies tasked with the lion's share of the nation's cleanup effortsEPA, DOD, and DOEhave several efforts under way that focus on removing barriers to technology use. The Federal Remediation Technologies Roundtable and the Western Governors' Association DOIT (Develop on-Site Innovative Technologies) committee are among the multiagency organizations that are working on these issues. DOE has established the National Environmental Technology Applications Center to promote the commercialization of technologies by assisting them through the so-called "valley of death" where technologies languish for lack of a champion. Environmental centers have also been established at other DOE sites (e.g., Oak Ridge National Laboratory, Lawrence Livermore National Laboratory, Rocky Flats Environmental Technology Site), and programs (e.g., Technology Connection Program) and hot lines have been started that focus on dissemination of technology information. DOD has aggressively pursued the use of new technology in its cleanup of McClellan Air Force Base in California and through its Air Force Center for Environmental Excellence. In addition, the State of California's EPA has established a program to certify environmental technologies that have proved successful in remediating sites. However, many of these endeavors are in their infancy and some are still trying to formulate their strategies for promoting environmental technologies.

For individuals in the field who are responsible for the cleanup of an environmental problem in the very near future, is there a more expedient way for them to understand and accept the inherent risk of using an innovative technology? We believe there is, and it's known as the WERC/FAST Initiative.

BACKGROUND

The WERC/FAST Initiative consists of two separate programs, established for two different objective. In combination, these programs provide a unique opportunity for problem holders to identify and to become comfortable with innovative solutions. The term "comfortable" is used to convey the psychological state that decision makers have when they are willing to accept the risks of using an innovative approach. WERC is the Waste-management Education and Research Consortium, which is a nonprofit collaborative program whose members include universities, colleges, and national DOE laboratories located in New Mexico. The mission of WERC is to expand the nation's capability to address issues related to the management of all forms of waste through education, technology development, and information transfer. As part of its mission, WERC hosts an annual Environmental Design Contestthe only environmental process design contest of its kind in the United States that provides university participants with design, as well as practical, experience on a real environmental restoration (ER) or waste management (WM) issues. Many of the design problems are actual DOE ER and WM projects sponsored by sites that are actively seeking solutions.

FAST is the Fast-track Advancement of Significant Technologies, which was initiated in 1995 by DOE's Office of Environmental Restoration and leverages a small amount of funding to make potential users at DOE sites more comfortable with existing technologies. This reassurance is primarily accomplished by providing high-quality treatability studies to document performance. The intent of FAST is to be fast and independent, to provide head-to-head testing with baseline technologies to benefit

technology owners that cannot afford quality testing, and to allow these technology owners to retain all proprietary rights. Because the FAST Program directly involves DOE sites in the selection of technologies, it fosters a working relationship between industry and the users at the DOE sites.

The synergistic relationship in both WERC and FAST Program provides an opportunity to develop an approach that embodies the strengths of each program and enhances their respective capabilities. This relationship, known as the WERC/FAST Initiative, concentrates on moving promising technologies selected by the sponsoring sites during the WERC environmental competition to a more field-ready level. DOE, through the Office of Environmental Restoration, has tasked the FAST Program to provide financial support to the universities selected at the environmental design contest. The universities, through cooperation with WERC and the problem sponsors, conduct treatability studies and performance testing.

FAST PROMOTES OWNERSHIP

At the heart of the FAST program is the "blood, sweat, and tears" approach. When an individual or organization strongly supports a position, it typically is not because someone else has said that "this position is good for you;" rather, it is because the individual or organization has put its own efforts into the realization of that position. When we invest our own blood, sweat, and tears into a position, we are likely to support it and take the risk associated with implementing that position. An excellent example of this sociologic trait is the Habitat for Humanity program. Under this program, the recipients of housing upgrades participate in the rehabilitation or construction of a structure and, therefore, invest "sweat equity" to gain ownership. As a result, they develop pride in a home that does not generally occur when the work is provided solely by others.

Table I presents a comparison of the basic goals and principles of the Habitat for Humanity program and how the FAST mission emulates this program.

Table I

FAST implements this approach through site involvement in the identification of existing technologies that have potential benefit to solve their environmental problems but lack sufficient performance testing to receive their acceptance. To acquire this important performance test data, the FAST Program provides fundingbut only if the sites interested in the technology agree 1) to leverage site funding and/or provide in-kind support (e.g., engineering, laboratory analyses, health and safety) to perform the performance testing; 2) to work with the FAST Program to identify the performance objectives and the test plan for the technology; and 3) to seriously consider using the technology if performance testing indicates its acceptability. By directly involving DOE sites in the selection of technologies, a working relationship is promoted between industry and the users at the DOE sites.

THE VALUE OF THE WERC ENVIRONMENTAL DESIGN CONTEST

The value of the WERC Environmental Design Contest to a DOE site can be expressed best by the enthusiastic sentiment that one 1995 sponsoring site, the Hanford Site, had for the competition. Hanford personnel stated, they "... would recommend participation [in sponsoring a problem] to any organization seeking novel solutions to technical problems for a relatively low investment. The research and testing that took place would

have cost between 1 and 2 orders of magnitude to duplicate. It is unlikely that much, if any, of the work would have been approved for in-house development owing to the uncertainties of results versus cost...."

WERC estimates that for every problem sponsored by a site, the participating universities provide a combined 40,000 hours of research labor. The return on investment for each site that sponsors a problem is conservatively estimated to be 20 to 1. Each graduate and undergraduate university team prepares a technical paper on its technology, presents oral and poster presentations, and performs benchscale demonstrations.

For the 1995 competition the DOE site sponsors were Rocky Flats (treatment of solar pond sludges), and Hanford (retrieval of salt cake from high-level waste tanks). More than 20 universities accepted the challenge of developing practical solutions for each of these problems.

FASTHOW IT LEVERAGES WITH WERC

The common thread between WERC and the FAST Program is the insistence that problem holders invest their resources (money and expertise) into the process of identifying and testing potential solutions. This investment of sweat equity furthers the likelihood that the problem holders will assume ownership of the technologies and actively pursue their implementation. This common thread strengthens the synergism between the FAST Program and WERC.

THE 1995 WERC/FAST INITIATIVE

During the 1995 WERC Environmental Design Contest, the judges identified five entries with promising solutions for the two specific DOE problems being used in the competition: treatment of solar pond sludges (Rocky Flats) and retrieval of salt cake from high-level waste tanks (Hanford Site). Criteria used to select the technologies for FAST sponsorship were developed with site participation. Approximately 25 individuals representing DOE, DOE contractors, EPA, State regulators, engineering consultants, industry, and academia judged the WERC competition. The two DOE sites reviewed the recommendations of the judges and concurred that three of the solutions had sufficient interest to the sites to warrant further refinement of those technologies to a more field-ready level. Through FAST, WERC supported students during the 1995 summer months to refine the technologies. The selected universities and their technologies are

University of Oklahoma, Norman: modified borehole mining system for material removal (for Hanford Site problem).

University of Alabama at Huntsville: a removal system that uses a pneumatic steel-pellet blaster device with magnetic retrieval (for Hanford Site problem).

University of Idaho, Moscow: use of naturally occurring apatite to immobilize heavy metals in solar pond sludges (for Rocky Flats problem).

Each university was assigned specific objectives to achieve for its technology. The University of Oklahoma needed to perform some initial scoping tests to determine a good water temperature and jet-cutter nozzle design for the bench-scale tests of its borehole mining system. Additional quantitative data were required to identify parameters and values that are most critical to material removal: salt removal rate versus water temperature, nozzle orifice design, water pressure and velocity, water flow rate, and distance between nozzle and the salt for a range of the parameters.

The University of Alabama at Huntsville approach uses a steel-shot blasting technique that is similar to sand blasting with the advantage of

magnetic recovery of the shot from the waste stream. Tasks included obtaining a suitable air blast gun from a commercial vendor and conducting quantitative tests to compare its performance to salt cake removal with sand blasting. Other activities consisted of determining optimum air pressures and airflow and selecting steel shot and nozzle sizes to assess the potential of this technique for cleaning the walls of tanks at the Hanford site.

The University of Idaho needed to determine if the naturally occurring mineral apatite would be an appropriate admixture to stabilization mixes for immobilizing heavy metals in solar pond sludges at Rocky Flats. This technique has been used at Superfund abandoned-mine sites and has been investigated by Pacific Northwest Laboratory. Team assignments included tests to identify the performance range of various types of apatite at different ratios of salts, water, and pozzolan to determine an optimized formulation for immobilization of heavy metals while maintaining structural integrity of the final waste form.

General objectives for each technology consisted of 1) identifying outstanding issues relative to each technology that need resolution before full-scale implementation can occur, 2) preparing a fact sheet about each technology, and 3) demonstrating each technology at either the Hanford Site or Rocky Flats and giving an oral presentation of the findings to DOE and the sites' operating contractors.

Kick-off meetings were held at Hanford and Rocky Flats in July with all the parties involved in the 1995 WERC/FAST Initiative, including students and site representatives. Students were able to tour the sites and see the problems that their technologies address. Basically, they were afforded the opportunity to "kick the tires" of the problem. Guidelines were developed for the testing to be performed by the students during the summer and were agreed upon by the sites. All the students' activities have not been completed.

The University of Idaho presented the findings of its chemical engineering students to representatives of Rocky Flats, WERC and FAST. Test results indicate that the use of apatite in the proposed grout formulation could stabilize the process sludge to meet waste acceptance criteria. The best apatite material seems to be commercially available fish bones rather than mineralized apatite. Rocky Flats representatives will now use this information to assess the use of apatite on actual radioactively contaminated sludges. The use of apatite as a geochemical barrier for a mixed-waste repository at the site is also being considered.

Presentations to Hanford Site personnel by teams from the University of Oklahoma and the University of Alabama at Huntsville also offered solutions to problems. Results achieved with Oklahoma's water-jet device (34.5 megapascals [5,000 pounds per square inch] unconfined compressive strength) on hard salt cake had a fivefold to sixtyfold increase in removal rate with 66C water in comparison with 16C water. Alabama's pneumatic steel-pellet blaster retrieved salt cake at a rate of 10 cubic inches per minute. While that rate probably is not fast enough for bulk retrieval, this method could be used for cleaning tank walls or internal portions of structures. The Hanford sponsors noted that "...this is the only entirely dry waste retrieval method to have even limited success on material this hard." The management of the High-Level Waste Retrieval program exemplified the site's interest with a request to the two universities, in cooperation with WERC, to support the refinement of

their technologies for use with the robotic manipulator arm being perfected at the Hanford Site.

IS THE WERC/FAST INITIATIVE SUCCESSFUL?

One sign of success is the commitments by the problem holders at the sponsoring sites to advance the technologies to the next step in readiness for field use. Both Rocky Flats and Hanford affirmed their intents to use these technologies. This step of embracing the technologies the students developed in the WERC environmental design contest, enhancing the technologies by leveraging funds from FAST and in-kind support from the problem holders, and actually moving toward field-ready applications of the technologies demonstrates the success of the WERC/FAST relationship.

Another sign of the success of a WERC/FAST Initiative is the desire of other sites to participate and to allocate their resources into sponsorships. Although this is the first year for the WERC/FAST Initiative, three sites have agreed to fund the sponsorship of problems for the 1996 WERC Environmental Design Contest. The Idaho National Engineering Laboratory problem involves the remediation of a series of 1,900- to 190,000 liter (500- to 50,000 gallon) underground mixed-waste storage tanks and surrounding soil. Savannah River Site's problem is the remediation of radioactively contaminated vegetation with a method other than incineration. The third problem is a waste management issue at Rocky Flats involving the treatment of industrial machining filters contaminated with a hazardous solvent and plutonium particles. These sites will provide additional support in the form of in-kind services to develop test plans under the FAST portion of the initiative and will offer consultation and mentorship to the universities selected to further the field implementation of their solutions.

THE WERC/FAST INITIATIVE MEETS DOE'S NEEDS

The WERC/FAST Initiative stands up to DOE's tough demands for strategies that are customer driven, practical, cheap, and flexible. One of the criteria for acceptance into the WERC/FAST Initiative is needthe technology must meet the needs of DOE sites, regulators, and stakeholders. Because the students who compete in the WERC Environmental Design Contest must use technologies that they are confident will work, existing technologies are selected and are often innovatively adapted. This use of existing technologies avoids long-term research and development efforts and benefits DOE sites by furnishing more performance data for site-specific applications. Small investments in performance testing that are leveraged through both WERC and FAST produce large results for the DOE complexproven technologies. The inherent flexibility of FAST allows adaptability, such as the WERC/FAST Initiative, to meet changing real-world needs and offers opportunities with solutions to environmental problems.

ACKNOWLEDGMENTS

This work was performed under DOE Contract No. DE-AC04-94AL96907 for the U.S. Department of Energy; Daniel Varhus is the DOE Grand Junction Projects Office project manager. This program would not be possible without the support of our DOE Headquarters sponsor, Lawnie Taylor, Office of Program Integration. Abbas Ghassemi of the University of New Mexico, Albuquerque, was instrumental in the development of the WERC/FAST Initiative; that aspect of FAST was possible because of his backing.

STUDENT DESIGN PROPOSAL FOR MECHANICAL MOBILIZATION AND RETRIEVAL OF RADIOACTIVE WASTE FROM UNDERGROUND STORAGE TANKS

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ABSTRACT

A student design team produced a conceptual design and planned a project for safe and cost effective removal of radioactive waste from underground storage tanks. The proposed design concept differed from baseline sluicing and remotely controlled manipulator arm technologies. Advantages of the design concept were identified as simplicity in deployment, operation, automation and maintenance, as well as potential for improved reliability over the manipulator arm technologies. Little or no additional radioactive waste would be generated. Legal and regulatory requirements, and health and safety considerations were taken into account in the project planning. All applicable legal and regulatory requirements were reviewed and actions were identified for compliance with these requirements. Legal and regulatory requirements included compliance with general radiation protection standards, standards for packaging and transport of radioactive waste, environmental regulations, procedures for decommissioning and decontamination of equipment and community right-to-know laws. A health and safety plan was produced which included monitoring of radiation levels, administrative and engineering controls and an emergency action plan. A plan for ensuring good community relations was identified which included public information meetings and information brochures describing the reasons for and the risks associated with the proposed project. A business plan and economic assessment indicated the viability and cost effectiveness of the proposed approach, based on present value project costs for waste removal from two tanks in a 34 month period.

INTRODUCTION

A group of eight students, five mechanical engineering seniors, one chemical engineering senior, one industrial hygiene senior and one MBA student produced a conceptual design for a mechanical system capable of mobilizing and retrieving radioactive wastes from underground storage tanks. The waste composition, storage tank specifications and other design requirements were those issued by the Waste-Management Education and Research Consortium (WERC) for the 1995 International Environmental Design Contest (1). The storage tanks were assumed to be located in Las Cruces, New Mexico, according to the contest rules. The single-shelled, underground storage tanks specified in the competition rules contain high level radioactive waste, varying in consistency from hard saltcake to sludge. Access to the interior of the tanks and,therefore to any equipment operating inside of the tanks, is severely restricted due to radiation levels. Because of this, any proposed waste retrieval system should be very reliable, simple to maintain and must be easily automated. The proposed concept met these requirements and avoids past practice sluicing which creates additional radioactive waste, and is significantly simpler than multi-degree-of-freedom manipulator arms, the other baseline technology.

RETRIEVAL SYSTEM DESIGN CONCEPT

After an extensive design tradeoff study, a rotating cutting device to mobilize the saltcake supplemented by an air conveyance system to remove the waste from the tanks was chosen for further consideration. The contest mailings (1) stipulated that the baseline, multi-degree-of-freedom manipulator arm technology and past-practice sluicing should be precluded from the proposed solutions. Several other alternative concepts were also considered but were determined to be inferior to the chosen one based on a sum of weighted attributes criterion or Pugh chart (2) analysis. This evaluation, however, did not include detailed design considerations or a detailed cost analysis because of time constraints.

Fig.1

Preliminary testing using a recipe for simulated saltcake supplied with the contest mailings revealed that typical cutting tools alone would not be likely to break up the hard saltcake to a form suitable for air conveyance, at least not in a time and energy efficient manner. However, it was hypothesized that, because of the brittle nature of the saltcake, its resistance to fracture could be significantly reduced by first "scoring" it (i.e. cutting a series of parallel grooves in it). Breakup of the material could then be accomplished by subsequent cutting with a high speed cutting wheel perpendicular to the grooves. Experiments with the simulated saltcake validated this approach. The contest mailings stipulated that an existing air-conveyance system was available for use with the retrieval system design.

To deploy the aforementioned scoring and cutting mechanism, it was proposed to use a telescoping arm with a central support, as illustrated in Fig. 1. A cutting head unit consisting of high speed abrasive scoring wheels with counter-rotating breakoff disks at right angles (shown schematically in Fig. 2) was to be mounted at the end of the telescoping arm. The arm could be folded into the central support so that the entire unit could be lowered into the tanks through existing central risers. The telescoping nature of the arm allowed the full range of the tank diameter to be accessed by the cutting head. The machinery would be powered with hydraulic motors because of their reliability and favorable torque-speed characteristics with relatively low weight, as compared to electric motors (3).

Fig. 2

The sludge material in the tanks was described as having a varying consistency but similar to that of peanut butter. In addition to breaking up the hard saltcake to a form suitable for air conveyance, this sludge would also need to directed into the air conveyance ducts. Thus,impellers would also be mounted on the cutting head unit to impart momentum to the sludge, forcing it into the air conveyance system.

The student design team believed that this conceptual design offered advantages over the baseline technologies. Because of the relative simplicity of the deployment mechanism in comparison to multi-degree-of-freedom manipulator arms, operation of such a system could be easily automated. Further, automation would be significantly less expensive than for manipulator arms and the reliability of such a system would be significantly improved over remotely controlled manipulator arms, again owing to the relative simplicity. Finally, the proposed system would create little or no additional radioactive waste, though it might be necessary to cool the cutting blades so that some coolant fluid would be

introduced into the radioactive waste stream and a heat exchange and recirculation system would be required for the coolant.

HEALTH, SAFETY, REGULATORY AND COMMUNITY RELATIONS ISSUES

In conjunction with the conceptual design, regulatory requirements, health and safety considerations and community relations issues were reviewed. In order to complete a waste retrieval project using the proposed technology, licensing requirements, radiation protection standards, packaging and transport regulations, environmental regulations, equipment decommissioning regulations and community right-to-know laws must be complied with (4-7). These were thoroughly reviewed and required actions for compliance were identified based on a New Mexico project site. A health and safety plan involving personal protection equipment and monitoring of workers for radiation exposure, employee training, engineering control of radiation levels and an emergency action plan was produced. A community relations plan including public meetings was also proposed.

General Radiation Protection Standards

The New Mexico Radiation Protection Standards (5) require everyone who receives, uses, transfers, possesses, or acquires a minimum amount of specific radioactive materials to hold a license (5). Therefore, licensing by the NRC would be required to perform the waste retrieval. Radiation exposure, regulated by OSHA and the NRC, should also be kept to levels that are As Low As Reasonably Achievable ("ALARA"), taking into account current technology and the benefits of public health and safety.

Decontamination

Decontamination is essential for maintaining the "ALARA" standards for the reuse, repair, and decommissioning of equipment used in the retrieval system. Decontamination removes a high percentage of the radioactive particles that adhere to the equipment, allowing for safer transport and maintenance. The chemical agent formerly used for decontamination of radioactive equipment was CFC-113. However, the Clean Air Act of 1990 bans the production of all known sources of atmospheric chlorine, including CFC-113. As a result, an alternate cleaning agent such as perfluorinated hydrocarbon surfactant solution would be required (6). It would be necessary to provide a decontamination area inside the above-tank facility, and decontamination could be performed through the use of high-pressure sprays.

Packaging and Transport

The packaging and transport of radioactive materials is governed by the NRC, 10 CFR 71; the DOT, 49 CFR 170-1891 (4); and the New Mexico Radiation Protection Regulations 3-700 & 4-260 (5). These laws pertain to the transport of licensed radioactive material outside the confines of the plant or place of use. Licenses for packaging and transportation would have to be obtained from the NRC, the DOT, and the NMED (New Mexico Environmental Department). Licensing would be required to transport samples of the tank waste off-site to be analyzed, since the tank contents must be analyzed before any retrieval begins.

The proper packaging requirements would have to be observed in accordance with the waste classification status (10 CFR 71 sub-part D, 49 CFR 173 sub-parts A and B and 49 CFR 178 sub-part K) (4). The packaging to be used to transport samples would have to be approved by the NRC. An application for package approval submitted to the NRC must include a detailed package description, a package evaluation, details of a quality assurance program, and the identification of the proposed fissile class.

The application requirements are described in full detail under 10 CFR 71 sub-part D (4).

Environmental Regulations

The Council on Environmental Quality (CEQ) has several requirements that must be met as mandated by the National Environmental Policy Act (NEPA). Environmental information documents (EIDs) would have to be prepared and submitted to the EPA for any proposed action that could have a potential impact on the environment. An environmental review process is required. The EPA will provide an environmental assessment to determine whether or not there is an impact significant enough to warrant the requirement of an Environmental Impact Statement (EIS). A record of the decision must be made available to the public. Further information on the procedures for implementing the requirements of the CEQ is located 40 CFR 6 (4).

Decommissioning

Decommissioning of the facilities and the equipment used is regulated by the EPA, the NMED, and the NRC. At the present time, there are no set standards. Each case is assessed individually, and a decommissioning procedure must be approved by all three agencies. The current practice is to reduce risk to below 10-4 to 10-6 and to reduce radiation levels to below external radiation standards set by the NRC (6). Retrieval equipment would need to be decommissioned in compliance with all regulatory decisions.

Emergency Planning and Community Right-To-Know

The Emergency Planning and Community Right-To-Know Act of 1986 (EPCRA), Title III of the Superfund Amendments and Reauthorization Act (SARA), requires that the public have access to information regarding the presence of hazardous substances in their communities. State and Local Emergency Planning Committees (SEPCs and LEPCs) must be notified when there is a release of reportable quantities (RQ) of a hazardous substances that may adversely affect the community. Reports must be made available to the public providing interested parties with names and quantities of the hazardous chemicals released. In the event of a release of an RQ, compliance with notification requirements as detailed in 40 CFR 302 (4) would be necessary.

Health and Safety Plan

All activities related to the operation of the waste retrieval system would be conducted in compliance with federal and state regulations for safe work practices as well as a proposed health and safety program. Federal health and safety regulations that must be followed include subparts of OSHA's general industry standards (29 CFR 1910.96) and subparts of the NRC's standards (10 CFR 20) for protection against radiation (4).

The proposed health and safety plan possessed the following components: monitoring and surveying to identify and quantify potential hazards, administrative controls and engineering controls to minimize those hazards, and an emergency action plan. This written health and safety plan was intended to be made available to all employees as well as all federal, state, and local authorities.

The main health and safety concerns associated with the waste retrieval are centered around the presence of alpha, beta, and gamma radiation within the single-shelled tanks. These types of radiation are caused by the presence of radioactive materials within the tanks, specifically 137Cs, 90Sr, 90Y, 238U, 239Pu, and 241Am (1). The health and safety program emphasized these concerns, but also addressed all other health

and safety concerns associated with the proposed project. The following discussion illustrates major components of the proposed health and safety program.

Monitoring

Continuous monitoring of radiation levels and airborne radiation levels in the repair unit, the control center, and the support center would be monitored to ensure a safe work site for employees and compliance with all appropriate regulations. All employees would wear film badges (to provide a legal record) and dosimeters (to provide an immediate reading) during their shifts. Monitoring would also aid in the selection of personal protective equipment (PPE) as it became necessary.

Administrative Controls

Employee training was a major element of proposed administrative controls. All site workers involved with the retrieval would be required to take a 40 hour training course and be certified to handle and remediate hazardous waste (29 CFR 1910.120) (4). Employee training would also be developed to address such areas as hazard awareness, hazard avoidance, proper use of PPE and limitations of that PPE (when applicable), evacuation procedures, the purpose of controlled work areas, Right-to-Know issues, the use of Material Safety Data Sheets (MSDS), and confined space entry. A medical surveillance program would be implemented to ensure employee health prior to and during employment. PPE is an administrative control that would be used as a last resort, but PPE would be readily accessible to employees for emergencies. The PPE would include gloves and clothing to minimize contamination during maintenance procedures. Employees would be required to go through a decontamination process after performing maintenance tasks that require contact with the waste. Decontamination facilities would be provided, and monitoring would be performed to verify adequacy of decontamination. Safety glasses would be worn by all employees. Respirators could be used if they were determined to be necessary during exposure monitoring, and if used, a Respiratory Protection Program would be implemented. Controlled work areas would be posted according to OSHA and NRC regulations to further ensure employee safety and site security, and audible warnings would be installed to alert employees entering high radiation areas. An industrial hygienist would be available during all retrieval operations to ensure compliance with the site health and safety program.

Engineering Controls

In the case of exposure to radiation, the controls would be primarily the automation of the retrieval system and the reduction of employees' time spent near the waste. Other forms of control such as shielding might be necessary to reduce employee exposure to "ALARA" levels, and the specifications of necessary shielding could be calculated once the radioactive content of the waste was known. The retrieval method was designed to minimize employee contact with the retrieval equipment, the amount of time that employees are in high radiation areas, and thus the employees' proximity to the waste. The retrieval system would be entirely automated, including the handling of the waste once it is out of the tanks and is being pumped into the waste transportation vehicles for subsequent transport to the waste retrieval facility. Employees would not be inside the above tank facilities (except in isolated areas) when waste was being pumped out of the single-shelled tanks and into the vehicles. Employees might have to enter the facility for tasks associated with moving vehicles out of the secondary structure when they are full and

when maintenance tasks are required. Again, when employees were engaged in these tasks, waste would not be pumped from the tanks, thus reducing potential exposure. The waste transportation vehicles would be designed so that employees working in the repair unit would not be exposed to levels above 0.2 mrem/hr (1). The above ground facility would provide general ventilation (as well as exhaust ventilation if air monitoring proves it to be necessary) to reduce the potential for airborne radiation.

It was determined to be imperative that the contents of the single-shelled tanks be analyzed before any retrieval could begin. This analysis would allow shielding requirements for the process to be calculated. Analysis results would also be used to determine the design of the waste transportation vehicles according to the appropriate regulations. The design of the above-ground facility and secondary structure would also be checked against the analysis results to determine if they adequately provide secondary containment to ensure that radiation was not released to the environment.

Emergency Action Plan

An immediate evacuation warning signal would be installed in the interim retrieval facility, in compliance with the regulations in 10 CFR 20 (4). The emergency action plan would include procedures for dealing with emergencies, as well as names of the local emergency agencies and federal and state regulatory agencies to be contacted in the event of an accident. Designated areas for first aid kits would be identified. MSDSs would be used to determine first aid procedures in the event that medical attention were necessary.

Community Relations

Good public relations would be imperative for success of the proposed project. A well informed community would help the project gain the support of regulatory agencies, help insure funding, and allow the project to proceed quickly since it would be clear that the rights, health, and environmental security of the local community would be protected. The statutory and regulatory requirements have been established in CERCLA as amended by SARA (40 CFR 300), and NEPA (40 CFR 1502-1503, 1506) (4). At sites where CERCLA response actions are taken, there are community relations requirements and procedures for specific situations in which public participation must occur. The community relations program would comply with these regulations.

It would be important to keep the community informed both before the project begins and as it progressed. Public information meetings would be held on a regular basis. These meetings would inform the community of the project's status as well as any changes in proposed actions.

Brochures would be distributed which outline the reasons and methods for retrieval, risk associated with the project, and emergency procedures that are in place for potential accidents. Possible questions and concerns the public may have would be anticipated, addressed, and responded to in these brochures. A telephone hotline would also be set up so that community residents could receive information about the project.

Another approach proposed to gain community trust was to create public participation programs where members of the public would be directly involved in the initial stages of planning. These participation groups would then oversee the project until completion.

BUSINESS PLAN AND ECONOMIC ASSESSMENT

A general description of project costs is given in Table I. The Gantt chart method for project planning was used to estimate the duration of the project and project scheduling. The proposed project allowed for retrieval of waste from two tanks within a 34 month period. the major phases of the project would be equipment design and construction and testing, facilities construction and retrieval from two tanks and equipment decommissioning. The total cost of equipment, required tank access hole enlargements and facilities construction was estimated to be $9.61 million, representing 65% of the total project cost. Costs associated with licensing requirements were omitted because they would be the same for any proposed retrieval method.

Table I

Economic Assessment

The assessment summarized in Table I was based on assumptions which produced the most conservative (i.e. the highest) estimates of project costs. A discount rate of 5% was used to compute present values and this was based on conservative estimates suggested by the EPA (21). This discount rate is used for CERCLA hazardous waste projects which have similar risks to the one described here. Tax and depreciation costs were omitted in order to make a cost comparison between baseline technologies and the proposed technology. The total estimated tax benefit based on the assumption of no salvage value for the equipment was $4,125,200.

CONCLUSIONS

A student design team proposed a mechanical removal method and planned a project which could safely and effectively remove radioactive saltcake and sludge from underground storage tanks. The mobilization and retrieval system differs from the baseline sluicing and remotely controlled manipulator arm technologies. The design concept utilized a rotating cutting head to break up the hard saltcake and this would be deployed by a telescoping arm and central support structure as an integrated system. This design was seen to offer several advantages. These advantages included the facts that it does not create significant additional radioactive waste, it offered simplicity in automation potential, deployment, and maintenance functions, and it offered significant potential for enhanced reliability compared to the use of a remotely controlled, multi-degree-of-freedom manipulator arm.

All applicable legal and regulatory requirements associated with the proposed project were reviewed and the actions which would have to be taken to comply with these regulations were identified. A health and safety plan which would ensure compliance with OSHA regulations and guarantee employee exposure to as low as reasonably achievable ("ALARA") levels was discussed and planned. An emergency action plan was also outlined.

A business plan was developed to assess project costs and scheduling requirements for a 34 month project duration. This business plan and an economic assessment indicated the cost effectiveness and viability of the proposed project. Costs comparable to the baseline, remotely controlled manipulator arm technology can be achieved, and the student design team believed that there was considerable potential for cost savings.

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ACKNOWLEDGEMENTS
This summary is based on the work of eight Clarkson University students whose entry in the WERC 1995 International Environmental Design Contest, April 17-21, 1995 in Las Cruces, NM took fourth place for Task I. They are: Ryan Boas, Kyle Dorr, Stephanie Franke, Brandy Gallagher, Mark Nystoriak, John Papalia, Bill Sharland and Doug Walters.

STUDENT-DESIGNED HIGH LEVEL IN-TANK WASTE RETRIEVAL SYSTEM AND MIXED WASTE REMEDIATION SYSTEM

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ABSTRACT
The Waste-management Education and Research Consortium (WERC) had requested designs for a high level radioactive waste mobilization and retrieval system for underground storage tanks and for a sludge

remediation process. The Montana Tech Environmental Design Team (MTEDT) designed creative, cost-effective methods for the given problems. One group within the team designed an In-tank waste Removal Integrated System (IRIS). WERC had requested that the waste be removed from the underground tanks within a six month time frame, and restricted the design from using a robotic arm or from sluicing. IRIS is a mobile system, relatively simple in design. Design parameters will allow removal of waste from a 0.75 Mgal tank in 5.8 months. The two major systems involved with IRIS are the pneumatic fracturing system and the air system. Other components include liquid/slurry retrieval pumps. The in-tank components are handled by a hydraulic cylinder system, with telescoping trusses. Air system components above ground handle collection of particulate, warming/dehumidifying the air stream, and elimination of dust emissions. The projected costs and operation/maintenance on IRIS, for waste removal from eight tanks, has been estimated to be $15.7 million. Another group within the team designed a two-step, low maintenance, semicontinuous Solvent Extraction and Contaminant Recovery Technology (SECRT), which is designed to be flexible for various contaminant levels and transportable. WERC had restricted the design from utilizing cementation or vitrification. SECRT's first step involves a liquid/solids separation and counter-current extraction of radionuclides, organics, and metals using a chelating agent and surfactant mixture. The second step involves sending liquid through the heating chamber of a two-stage, forced-circulation evaporator/crystallizer, where 70% of the contaminated liquid stream is flash-vaporized. The residual liquid is cooled and sent to a solvent/water separator, where the solvents are collected and marketed as either an industrial solvent or incinerator fuel. The water is further filtered to dischargeable levels. SECRT is designed to handle 7200 gallons per day (gpd) in contaminated liquid and 1800 gpd in pond sludge. The projected cost of treatment of the pond sludge is $4.66 million, or $3.88 per kilogram of sludge. MTEDT's plans also address legal issues, health and safety concerns, and community involvement.

PROCESS DESIGN FOR HIGH LEVEL IN-TANK WASTE RETRIEVAL SYSTEM

Preliminary Activities

A complete characterization of the tank interior will be conducted prior to starting waste retrieval operations. Characterization will include sampling for explosive chemicals such as hydrogen, ferrocyanide, and organic compounds, as well as depth profiling of waste moisture content. Water content data will be gathered with a modified neutron probe. The Montana Tech Environmental Design Team (MTEDT) has assumed that pumpable liquids will be removed from the tank. If a tank has less than three risers excluding the central riser, additional risers will be installed to accommodate monitoring and air circulation equipment. Also, nonessential in-tank hardware will be removed and decontaminated. A 20 foot by 50 foot concrete, load-bearing pad will be constructed adjacent to the tank. Mobile units containing air treatment equipment, a boom hoist, the decontamination chamber, and a containment unit will be assembled on the bearing pad.

Removal Process

An In-tank waste Removal Integrated System (IRIS) will be employed to mobilize and retrieve waste from the single shell tanks (SST's). IRIS consists of dual telescoping hydraulic booms with an integrated Waste Mobilization and Retrieval Module (WMRM) located at the end of each boom.

The WMRM is capable of solids fracturing, air conveyance, and liquid/slurry removal.

To mobilize and retrieve a full tank of waste, two sets of dual telescoping booms with identical technology (IRIS Jr. and IRIS Sr.) will be used. IRIS Jr. is a smaller version of IRIS Sr. and is necessary to enter a full tank of waste; however, for tanks having a minimum of 18.5 ft of clearance between the tank dome and the waste, only IRIS Sr. will be used.

Figure 1 illustrates IRIS's six major components: the IRIS deployment hoist, Vertical and Rotational Support System (VRSS), Support Column, Telescoping Boom, Angular Displacement Cylinder (ADC), and the WMRM, in addition to the air conveyance system. IRIS will initially be lowered into the tank through a 36 inch central riser and will be locked into the VRSS. Utilizing the VRSS, Telescoping Boom, and the ADC, a Programmable Logic Controller (PLC) will allow IRIS to maneuver within the tank.

Fig. 1

A pneumatic fracturing device located on each WMRM utilizes eight reciprocating air cylinders with chisel attachments to fracture the waste into sizes easily handled by the air conveyance system (see Fig. 2.). The PLC will operate the Intermittent Step and Repeat Removal Process (ISRRP) for solids fracturing. The ISRRP is similar to the procedures used in computer controlled milling applications and incorporates removing a layer of material in one plane, lowering the WMRM a prescribed depth, and removing another layer of material. To begin the ISRRP, IRIS will be lowered to it's set position in the tank. The ADC will then retract, moving the WMRM upward and outward, and the telescoping boom will be fully extended (IRIS Jr. can extend 30 ft, and IRIS SR. can extend to the interior tank wall.). Once in position, the telescoping boom will force the pneumatic fracturing device into the waste until a two inch depth is reached. The telescoping boom will then retract until clear of the waste, rotate through an angle along the circumference which allows a slight overlap of the previous position, and will again be extended into the waste. This process will be repeated until an annulus of waste around the perimeter is completely removed. The telescoping boom will then retract as the ADC extends, moving the WMRM radially inward, so that the fracturing of the next inner annulus can be completed. This step will be repeated until the WMRM reaches the center of the tank. At this time, the aforementioned process will be repeated for the next layer.

Fig. 2

As the waste is fractured, it will simultaneously be removed from the tank by an air conveyance system. Once outside of the tank, the waste will enter a pulsed-jet baghouse and will then be temporarily collected in a Collection, Storage, and Transfer (CST) station. From the CST, the waste will be transported to a Waste Retrieval Facility (WRF).

When liquids or slurries are encountered, the ISRRP will be interrupted. IRIS will be removed from the tank, and a submersible pump will be attached to the WMRM. IRIS will then be returned to the tank and directed to the liquid/slurry through manual operator control. Once the liquids/slurries are pumped, IRIS will be withdrawn from the tank and the pump will be removed. IRIS will then be redeployed, and the Step and Repeat process will again be enacted. Once the waste is removed, it will be temporarily placed in the CST station until it is transported to a WRF.

Wet or damp waste that cannot be pumped will be dried by hot, dry air convection and will then be pneumatically fractured and removed as a solid.

After IRIS has removed the bulk of the waste, an abrasive wire brush assembly will be attached to the WMRM to dislodge residual waste from the tank walls. This step will guarantee removal of greater than 99% of the waste.

Post-Retrieval

When retrieval at one tank is completed, the system will be transported to the next tank. IRIS Jr., if required, can be deployed at the next tank while IRIS Sr. finishes operations at the first tank. This practice significantly reduces total project time. As IRIS is removed from the tank it will be decontaminated by remote CO_2 blasting in the decontamination chamber (1). After decontamination, retractable shielding on the deployment hoist will enclose the booms to limit exposure until the system is redeployed at the next tank. Decommissioning will include the recycling of steel and the entombment of non-recyclable materials (2).

Project Time Requirements

The IRIS system will completely retrieve waste from a 0.75 Mgal tank in 5.8 months. This estimate is based upon retrieval of the contents of Hanford tank SX-105 (3), and laboratory retrieval rates of various waste types. Emptying a 0.5 Mgal tank requires 4.1 months and emptying a 1.0 Mgal tank requires 7.4 months. The total time required to retrieve waste from eight tanks (assuming four 0.5 Mgal and four 0.75 Mgal tanks) is 46 months. If IRIS Jr. and IRIS Sr. retrieve waste from two separate tanks concurrently, the total retrieval time will be reduced to 28 months.

PROCESS DESIGN FOR SLUDGE REMEDIATION

Waste Characterization

Analyses of the pond water and the pond sludge were provided by WERC for each of the three ponds. These data describe a chemically complicated hazardous waste which is characterized by broad ranges of nitrate, salts, metals, radioactivity, and total organic carbon (TOC). The complexity of the waste contaminates restricts the available remediation technologies; WERC further restricted treatment options by stating that vitrification and cementation methods were unacceptable. MTEDT developed a two-step, low maintenance, semicontinuous Solvent Extraction and Contaminant Recovery Technology (SECRT) which is designed to be flexible for various contaminant levels and transportable for use at future sites.

Step One of SECRT

Supernatant water is initially separated from the settled sludge. This water is then pumped from the covered storage tank, illustrated in Fig. 3, and sent to the contaminated liquid stream (CLS). The CLS is treated in Step Two. Recycled heated water (RHW) from Step Two is used to remove additional salts and to slurry the remaining solids from the original tank into hydrocyclone A; make-up water may be added if necessary to achieve proper slurry consistency. Mixing the RHW and the sludge results in the transfer of water-soluble contaminates from the sludge into the aqueous phase. After passing through hydrocyclone A, the liquid stream is sent to the CLS, and the sediment slurry travels to the first contaminate extraction mixer (CEM) in the counter-current extraction system.

Fig. 3

The first and second CEMs operate counter-currently using a chelating agent and surfactant mixture to remove radionuclides, organics, and

metals from the soil matrix. In the first CEM, sediment mixes with the recycled extractant liquor from the second CEM. Hydrocyclone B separates the slurry from the first CEM into a spent liquor stream which travels to the CLS and a sediment stream which travels to the second CEM. Fresh chelating/surfactant solution is added to the second CEM, and the liquor stream recovered from hydrocyclone C is sent to the first CEM while the sediment travels to the final water wash.

The counter-current system utilizes the extractant most efficiently by precleaning the most contaminated sediment with the liquor stream the second CEM and then contacting fresh extractant solution with the precleaned sediment. The second CEM mixes the cleanest extractant with the cleanest sediment, thus maximizing the difference in pollutant concentration and promoting dissolution of contaminates.

After hydrocyclone D, a high-efficiency centrifuge separates the sediment from the wash water, and dewaters the sediment to approximately 80% solids by weight (4). Liquid collected from the centrifuge travels to the CLS, and the solids are tested to determine proper disposal options. Disposal alternatives for delisted sediment include land application and possible marketing. If low-level radioactivity is present, shallow trench burial is a disposal option, or the sediment may be transported to a radioactive waste facility in Salt Lake City, Utah (5). High-level radioactive waste (HLW) may be disposed of at DOE facilitated sites.

Step Two of SECRT

The CLS travels through the heating chamber of the two-stage, forced-circulation evaporator/crystallizer (EC) where the waste stream temperature is raised under pressure. This process allows the liquid to reach boiling temperatures without actually volatilizing, thereby preventing scaling inside the chamber. The heated stream then travels to the low-pressure portion of the EC where the waste is concentrated and 70% of the CLS is flash-vaporized (6). The vapor stream, comprised of water and volatile organic compounds (VOCs), travels to the condenser while a brine waste containing metals, salts, radionuclides, and nonvolatilized organics exits at the bottom of the EC. A slip stream of the exiting brine waste slurry is recycled into the heating chamber. The recycling rate of the slip stream controls the CLS retention time in the EC and consequently determines the solid crystal content of the ultimate waste product. Depending on the radioactivity levels, the waste is dealt with accordingly, as mentioned in the previous section.

The vapor stream exiting the EC travels through a pipe coiled around the CLS input stream. This heat exchange process serves to preheat the CLS and to precool the vapor before it reaches the condenser. The condenser cools vaporized organics and water vapor to a liquid phase. Residual vapors leaving the condenser flow through an activated coconut-shell carbon unit to remove remaining organics and then flow through a High Efficiency Particulate Air (HEPA) filter to remove any potentially entrained radioactive particles (5). Condensate travels to a solvent/water separator where solvents will be collected and marketed as either an industrial solvent or incinerator fuel. The condensed water travels through a granulated, activated carbon filter where any remaining organics are adsorbed to produce dischargable water. A portion of the condensed water is recycled to the storage tank and the final wash portions of Step One via the RHW stream.

RESULTS OF TECHNICAL ASSESSMENT

In-tank waste Removal Integrated System

The Telescoping Boom consists of a double-acting telescoping hydraulic cylinder and two telescoping trusses. The double-acting telescoping hydraulic cylinder controls the length of the boom. The telescoping truss provides bending and torsional support for the boom and provides housing for the air conveyance lines, hydraulic lines, pneumatic lines, and the pump lines. A 55 gpm hydraulic fluid pump with a maximum operating pressure of 1500 psi will be used to maneuver the cylinders and the trusses. The ADC controls the angle at which the telescoping boom is maintained relative to the support column and will be operated by a 10 gpm hydraulic fluid pump with a maximum operating pressure of 3,000 psi. The ADC will apply a maximum force of 12,000 pounds.

The WMRM performs three separate tasks: 1) Pneumatic Fracturing: The pneumatic fracturing device uses four levels of chisels to mobilize the solid waste into sizes easily handled by the air conveyance system. Since one WMRM is located on each boom, a total of 16 chisels will be used. A 64 cfm airstream at 90 psi will be used to drive the 16 chisels. 2) Air Conveyance: Fractured waste will be captured and removed from the tank in a high velocity air stream utilizing a duct velocity of 7200 fpm. 3) Liquid/Slurry Removal: The liquid/slurry retrieval pump will be placed in the WMRM and submersed into any pockets of liquid and/or slurry encountered during the removal process. The pump will be capable of handling 70 ft of head and 75% solids by volume. The telescoping booms will be capable of fracturing the waste at a rate of 1.34 cfm. Dry material will be removed at a maximum rate of 5.34 cfm by the pneumatic conveyance system (2.67 cfm from each telescoping boom). A submersible pump will remove the liquid/slurry at a maximum rate of 50 gpm.

The pneumatic conveyance and drying system incorporates a closed air movement system to contain radioactive particulate. The air conveyance system is capable of conveying the mobilized waste to the surface at a rate of 500 pounds per minute. Each of the two waste removal pipes will convey the particles into cyclones mounted with the turbo blowers. From the two blower systems, the airstream will be joined by an additional exhaust ventilation line from the tank. Flow will then be directed to a pulsed-jet baghouse, which is currently the best available technology for particulate removal (7). The particles will be collected in a hopper at the bottom of the baghouse. Here, another conveyance system will interface with the CST. The airstream then passes through parallel desiccant dehumidifiers/preheaters and is returned to the tank. Exhaust from within the secondary containment will be vented through a double HEPA filter bank. Three lines are utilized to exhaust air and/or particles at a total rate of 7,200 cfm. Two lines return hot, dry air to the tank at a total rate of 7,000 cfm, maintaining a slight vacuum in the tank. Airflow is directed downward and deflected out to ensure uniformity of air distribution.

Solvent Extraction and Contaminate Recovery Technology

SECRT is designed to process up to 7,200 gallons per day (gpd) of contaminated liquids and 1,800 gpd of pond sludge. All water and slurry lines are constructed of stainless steel to resist corrosion. MTEDT performed laboratory settling experiments to estimate the ratio of supernatant liquid to sludge volume. Based on these experiments, MTEDT determined that approximately 5,500 gallons of supernatant liquid shall be pumped from each storage tank at a rate of five gallons per minute (gpm) and sent to the CLS. Recycled heated water (RHW) is used to dilute the remaining sludge to a slurry containing approximately 30% solids by

weight for easier pumping. This slurry is pumped through a hydrocyclone at a rate of 3000 gallons per hour (gph). All hydrocyclones used in SECRT were designed using on estimated dry particulate specific gravity of 2.0 with the following size distribution: 100% finer than 4.75 mm, 70% finer than 75 um, 20% finer than 40 um, and 5% finer than 20 um (8). The slurry exiting the hydrocyclone through the one-inch diameter outlet pipe will contain 60-65% solids (9). The slurry is transferred into a covered, 500-gallon polyethylene tank and mixed by means of a 3/4-horsepower shaft mixer operating at 420 revolutions per minute (rpm). An extract solution of EDTA and a proprietary surfactant is added to the slurry to achieve an approximate volume ratio of 3:1 liquids to solids (10). To ensure thorough contact of extractant with particulate, the solution is mixed for 15 minutes--the optimum agitation time as determined by MTEDT lab experiments. This process is repeated in two additional mixing tanks--one using fresh extractant solution and the other using RHW. Finally, an industrial, high-efficiency centrifuge dewaters the cleaned sludge to approximately 80% solids (4).

Supernatant from the storage tank, liquids from each hydrocyclone, and liquids from each centrifuge process are pumped through the forced-circulation EC at a rate of five gpm. A full-scale treatability study shall be conducted to determine the flow rate of the EC slip stream necessary to achieve maximum crystal content in the waste product. A high crystal content (due to increased water evaporation) minimizes the volume of hazardous waste product produced. The condenser discharges condensate at a temperature of approximately 125F(11). Condensate is directed through a separation unit where organic solvents are removed from the water. Excess vapors and the condensed water are routed through two separate activated carbon units to complete the remediation process. A portion of the pure water is recirculated via the RHW to be utilized as an extracting solvent.

METHODS FOR TESTING UNIT OPERATIONS/PROCESSES

Methods for IRIS

The fracturing rate for both saltcake and sludge was determined experimentally by measuring the time required to fracture a known volume of material with a single pneumatic chisel. A particle size distribution after fracturing for both saltcake and sludge was determined by sieve analysis. Bench scale waste removal rate was determined by measuring the time required to vacuum a known volume of material. Both the sludge drying rate and the critical moisture content at which the sludge can be fractured were determined by measuring the drying time of a known amount of sludge in a 104C oven. A desiccant was placed in the oven to reduce the moisture content of the drying air to simulate full scale conditions. The critical moisture content of saltcake was determined by mixing simulants of varying percentages of moisture and recording the highest percent of moisture content at which a fracturable solid could be attained.

Methods for SECRT

Using 0.2 M and 0.4 M EDTA, 1.00 M acetic acid, and 1.00 M citric acid, MTEDT conducted experiments on 22 test samples containing various metals, metal-salts, and pulverized kaolin. Supernatant water decanted from each of the test samples, and immersion solvent extraction was performed on the remaining clay. The decanted water, extraction solvent, and cleaned clay were first filtered, then digested, and finally analyzed using an Inductively Coupled Plasma (ICP) analysis.

A series of five immersion extractions were conducted on triplicate surrogate samples using a deionized water (DI) wash for the initial extraction, 0.1 M EDTA with 0.5 M Na2HCO3 (sodium bicarbonate) solutions for the next three washes, and a final DI water wash. Each solvent was mixed with the surrogate sludge and the resulting slurry was centrifuged. The liquor was collected, digested, and filtered for ICP analyses to determine the metals removal efficiency of each wash. In addition, residual contaminant quantities from the extractant solutions were approximated by evaporating a known quantity of extract liquor from each wash.

LAB RESULTS

IRIS Lab Test Results

Table I lists the results of the lab tests conducted. A single pneumatic chisel was used for the fracturing tests; therefore, all values listed are for bench scale unless otherwise noted (F = full scale, B = bench scale).

Table I

SECRT Lab Test Results

Although water removes a significant portion of adsorbed metals from course soil fractions, the addition of a chelating agent enhances the metals removal from the fine soil fractions. Both EDTA solutions in Experiment One exhibited consistently high removal efficiencies for the added metals without removing the metals associated with the pulverized kaolin. Therefore. EDTA was selected on the basis of economics, safety issues (i.e. concerns associated with acid extractant), and the lack of performance differences among the four acidic solutions tested. Based on literature searches (12), previous knowledge, and lab results, 0.1 M EDTA combined with a 0.5 M Na2HCO3 buffer solution was chosen for use in the bench scale and Experiment Two. Also, an oxidizing agent is not included in the extractant solution because it will not improve the removal efficiency of cadmium chromium, and nickel (13)--which are the only metals present in the WERC surrogate. At the beginning of Experiment Two, the optimum centrifuge rate was determined to be 2,000 rpm for a spin time of 15 minutes which reduced the sludge to 65% solids by weight. Experiment Two resulted in high removal efficiencies for calcium, chromium, and nickel, which are predominately extracted in the EDTA washes, as shown in Fig. 4.

ECONOMIC ASSESSMENT AND BUSINESS PLAN CONSIDERATIONS

All of the projected costs and operations/maintenance on IRIS are shown in Table II.

Fig. 4

Table II

If additional risers are necessary for a particular tank, a charge of $4 million would be added for each additional riser required.

The business plan and assumptions include the following to meet project goals:

Three shifts of four people would be required for operations, including a Health Physics Technician and supervisor per day.

All equipment is shipped FOB to the WERC site.

Labor costs associated with major equipment replacement or repairs are not included.

Utility costs are per New Mexico Public Utility Commission mandated rates.

All necessary risers are assumed to be in place.

Research and Development costs include setup of pilot operation.

All of the projected costs and operations/maintenance on SECRT are shown in Table III.

Table III

The cost of treatment per kg is $3.88, which includes $1.08/kg for capital costs, $0.53/kg for chemical costs, $0.06/kg for research and development, and $2.22 for operation and maintenance costs. There is also a potential that may be realized for a small amount of revenue, since the recovered solvents may be marketable.

The following is a list of process and cost assumptions:

75 gal/hr sludge feed rate from original tank.

Mobilization & set up: 20 days; demobilization: 10 days.

50 days of operation with 10% downtime.

24 hrs/day, 7-day workweek.

178,357 gal of sludge to remediate;

Personnel per shift to include 1 supervisor, 3 technicians, 2 maintenance personnel (on call)

Labor costs associated with major equipment replacement or repairs are not included.

Utility costs are per New Mexico Public Utility Commission mandated rates.

Due to unknown speciation of TOC, exact quantity of surfactant required is unknown--listed quantity is a reasonable estimate.

LEGAL, HEALTH, AND REGULATORY CONSIDERATIONS

A comprehensive site safety and health plan (SSHP) to ensure compliance with all legal, health, safety, and environmental regulations is an integral part of the overall design of IRIS and SECRT. The intent of this plan is to protect all life forms from the potential risks associated with exposure to ionizing radiation. Therefore, it includes all necessary criteria to keep exposures "as low as is reasonably achievable" (ALARA). These criteria are divided into safety and health, environmental, contingency planning, and legal considerations.

Safety and Health

For the duration of the mobilization, retrieval, or remediation of the waste, 29 CFR 1910.96 and 29 CFR1910.120 regulations will be adhered to. All applicable Nuclear Regulatory Commission (NRC) requirements (including 10 CFR part 20, subparts A through O, and appendices) will be followed as well. To achieve compliance with all regulations, a site-specific characterization, safety and health survey will be done to establish baseline contaminant levels compared to normal background levels. Ionizing radiation will be continuously monitored through the use of personal monitors, and area sampling as set forth in 10 CFR 20.1501. Based on sampling data, an exclusion zone (EZ) will be determined. Limited access to the zone will be granted to perform initial equipment setup and routine maintenance. A contamination reduction zone (CRZ) will be established in which a personnel decontamination station and auxiliary and emergency equipment will be located. The area outside the CRZ is the support zone and should be considered clean with no more than 0.002 mrem/hr of exposure. The support zone will contain the OCMS located up to 1300 feet from the tank undergoing retrieval operations. To ensure controlled access, the EZ will be fenced. The fence will be supplemented with visual monitors and alarms. All personnel entering the EZ shall wear the following personal protective equipment (PPE): Tyvek coveralls with hood, Tyvek gloves and boot covers, and full face air purifying

respirators with radionuclide cartridges. In addition to the PPE, all personnel working at the site shall wear appropriate dosimeters such as thermoluminescent dosimeters (TLD) and film badges. All personnel will be admitted into a radiation health and protection program to monitor exposures to ionizing radiation. At no time shall anyone receive a dose greater than 0.2 mrem/hr or a cumulative annual dose of 500 mrem, nor shall any organ or tissue receive 50 rem cumulative dose. The maximum dose for the eye is 15 rem. If at any time these doses are exceeded, further abatement measures shall be implemented immediately.

Environmental

All piping, duct work, and system components containing waste shall be double walled and shielded with a high density material with a half value layer adequate to protect personnel and the environment from gamma radiation. Alpha and beta radiation can be absorbed or shielded with normal construction materials to prevent them from being released into the environment. If a controlled release of air should be required, it will pass through a pollution control system to catch radioactive particulate. Controlled releases will be monitored to ensure regulatory compliance.

Contingency Planning

If an accident occurs and results in either a severe injury or a release of waste into the environment, emergency procedures will be immediately initiated to control the accident. The procedures shall be based on a worse case scenario at full operational conditions. A coordinated abatement effort between local health and emergency personnel will be implemented to minimize any adverse effects.

Legal Considerations

The Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) governs the remedial activity on a hazardous waste site. The immobilization and retrieval design process was developed to comply with the criteria established in the National Contingency Plan (NCP):

Threshold Criteria: Overall protection of human health and environment; and compliance with Applicable or Relevant and Appropriate Requirements (ARARs).

Primary Balancing Criteria: Long term effectiveness and permanence; reduction of toxicity, mobility, or volume through treatment; short-term effectiveness; implementability; and cost.

Modifying Criteria: State acceptance and community acceptance.

ARARs are regulations that pertain to the environmental work at a CERCLA hazardous waste site. ARARs for this project can be obtained from the following legislation: National Environmental Policy Act (NEPA); Atomic Energy Act of 1954; Resource Conservation and Recovery Act (RCRA); Clean Air Act; Clean Water Act; and from Emergency Planning and Community Right to Know Act.

PLAN TO ESTABLISH COMMUNITY RELATIONS

The community involvement plan has two primary purposes: 1) develop an understanding of public concerns and respond to such concerns; and 2) nurture positive community relations. In order to achieve these goals, the public involvement plan will include the following features:

1) Community input interviews. The initial phase of the community involvement plan will consist of thirty interviews conducted with a broad

cross-section of subjects. The goal of this activity is to identify the spectrum of community concerns. The IRIS project must remain flexible and responsive to concerns raised in this portion of the plan.

2) Availability of information. WERC must take the initiative in disseminating information in order to assure that the public is well informed. Furthermore, this information must be presented in a non-technical format which is accessible to the general public. Information presented here will include progress reports, press releases, fact sheets, explanation of concerns as WERC perceives them, explanations of how WERC arrived at certain decisions, and any other requirements under Superfund Amendment and Reauthorization Act, Title III.

3) Public meetings. Public meetings will be used as a forum for WERC to provide information and for the public to voice its concerns. Public meetings shall involve concerned individuals as well as citizen groups.

4) Central public information contact. The WERC employee overseeing press releases shall also act as a point of contact for the public. This individual will provide the public with personal contact and will allow the company to track public response to the IRIS project.

CONCLUSION

WERC has explored two main waste removal systems for the underground tanks: sluicing and a robotic arm. IRIS is a much more attractive system than sluicing for the primary reason that IRIS creates no additional waste. IRIS will also offer a significant improvement over existing robotic arms because of IRIS's mechanical simplicity. Past arm-based systems have not been cost-effective due to high maintenance requirements and a low degree of mechanical availability. The MTEDT strongly believes that this is not an inherent problem with arm based systems, but rather that it is symptomatic of a design that is overly complex for the task at hand. We believe that IRIS will represent a significant improvement over past arm-based systems in the areas of cost-effectiveness and mechanical dependability, because of the inherent simplicity of its design. The major structural components of the system are simple hydraulic cylinders. Because IRIS has a relatively simple design, we believe it will retrieve waste at a significantly lower cost. We estimate typical retrieval time for a 0.75 Mgal tank to be 6 months and the total cost for eight tanks to be $15.7 million. Our estimated cost represents significant savings compared to recent government cost estimates for this project. In 1993 the General Accounting Office reported a cost estimate of $15 billion to retrieve waste from 177 tanks at the Hanford site, representing an average cost of $85 million per tank. MTEDT believes that cost-effectiveness and mechanical dependability are precisely the type of innovation the Tank Waste Remediation System program needs.

SECRT's two-step process remediates the pond sludge and water resulting in cleaned sediment, high quality effluent, concentrated solvents, and a minimal quantity of hazardous waste. Step One uses a multiple extraction process to remove contaminates from the sludge; the second step utilizes a forced-circulation evaporation system for volume reduction to concentrate the hazardous contaminates for disposal. Rather than destroy the solvents that are volatilized in the evaporator, SECRT's collector/separator recovers the marketable solvents for future sale. Additionally, SECRT condenses and collects the water vapor evolved from the evaporator for either reuse in the process or discharge as industrial quality water. These pollution prevention measures create beneficial products from an otherwise useless waste stream destine for disposal. The

total cost for the pond sludge and water remediation is $4.66 million, or $3.88/kg of waste. As a unique feature, SECRT is very transportable, and after treatment at the WERC site, may be dismantled and relocated to treat similarly contaminated sites. SECRT is highly adaptable to waste types and contaminant levels.

ACKNOWLEDGMENTS

The following individuals were responsible the design and proposal work associated with IRIS: Julie Evans, Philip Hannifin, Chris Kaiser, Frank Raab, and Shannon Wilson. Additionally, the following individuals were responsible for the design and proposal work associated with SECRT: Jodi Bingham, Cindy Bock, Allison Malsam, Annette Murphy, Christi Witt, Jodi Woods, and Matt Zietlow. Gratitude is extended to our team adviser, Dr. Herold (Butch) Gerbrandt.

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TASK II: EVAPORATION POND SLUDGE TREATMENT

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ABSTRACT

The University of Idaho Environmental Design Team (UIEDT) has completed a preliminary treatment design for Task II as described by WERC. This task involves 178,357 gallons (35 wt% solids) of contaminated sludge which has been removed from evaporation ponds and is temporarily being stored in 10,000 gallon tank. In compliance with the requests of WERC and the requirements of the USEPA, the treatment process remediates the sludge to a form which meets 30 year Land Disposal Restrictions (LDR) and Department of Transportation (DOT) shipping requirements. Also considered in the process design are the social and ethical responsibilities to the workers and community.

The remediation process designed by UIEDT focuses on the maximization of the permanent immobilization rather than stabilizing the contaminants in a support matrix. This not only ensures long-term safety, but also helps to minimize process costs by eliminating waste transported to a landfill. Each part of the treatment plan has been chosen and optimized to maximize process efficiency, minimize overall costs, and guarantee successful remediation. The proposed plan involves two primary phase treatments; the volatile organics with bioremediation and the heavy metals with immobilization.

The initial task in the process is to convert one of the existing storage tanks into a "treatment tank" where sludge from the remaining tanks will be pumped prior to treatment. Sludge in the treatment tank will be heated to volatilize off the organics, separating them from the high salt concentrations which are toxic to the microbial populations responsible for remediation. After being recondensed, the organics are sent through a two stage bioremediation process. The anaerobic reactor degrades the most complex organics into simpler forms over a nine day period. Effluent from the anaerobic stage is fed directly into the aerobic reactor where the remaining contaminants are degraded in a ten hour period. Streams exiting from the bioremediation phase are void of any restricting contaminant concentrations and can be released straight into the local sewer.

Simultaneously with the volatilization of the organics is the immobilization of the heavy metals. A reactive hydroxy apatite is fed into the treatment tank at 1% weight of the solids and mixed for a 24 hour period. The apatite provides nucleating sites for a reaction resulting in the precipitation of metal-substituted apatite minerals, which have leach resistance typically below EPA drinking water standards and almost permanent durability. The apatite has also proven to lower salt concentrations. After treatment, the soil and liquid phases of the sludge are separated by rotary vacuum drum filters. The resulting water stream has been approved for release into the local sewer and the solid phase can be returned to the land where it is useful not only as fill, but also as a continuing treatment process for any further leached contaminants from the site. The proposed treatment process can be completed in 100 days, with and operating schedules of eight hours per day, seven days per week during the summer months. Total remediation

costs for this site are approximately $166,000 or $0.93 per gallon of contaminated sludge, but would decrease significantly with and increase in the volume treated.

INTRODUCTION

WERC of New Mexico has requested a process design to stabilize/remediate 178,357 gallons of contaminated sludge which has been removed from shallow evaporation ponds and is currently being stored in a series of 10,000 gallon tanks. Prior evaluations of the site have eliminated grouting as a viable treatment and have also determined that the treated waste must meet 30 year stability criteria. In response, the University of Idaho Environmental Design Team (UIEDT) has developed a treatment process which will not only remediate the contaminated sludge at this site. but can also be transported and applied to contaminated slurries at other locations.

For the remediation, UIEDT's process involves bioremediation to treat the organic contaminants and a hydroxy apatite immobilization for the heavy metals and to reduce the salt concentrations. The following table includes the treatment standards that must be met:

Table I

The proposal discussed in this report focuses on the technical process description, including bench scale evaluations. However, the importance of a comprehensive treatment plan is recognized, and so the proposed design also includes an economic assessment, environmental requirements, health and safety evaluations, and consideration of a community relation plan.

SLUDGE TREATMENT

The UIEDT has developed a process to treat 178,000+ gallons of industrial waste sludge previously removed from evaporation ponds and currently being stored in a set of eighteen 10,000 gallon tanks. It has been a goal of the team to develop a treatment process, not only for this particular problem site, but one which can also be adapted and transported for use at other contaminated sludge sites.

Several aspects of handling a contaminated waste must be taken into account when determining viable treatment options. Listed or non-listed waste? In-situ or ex-situ treatment? Fixation or stabilization? On-site or off-site disposal? The treatment methods must be evaluated and the design concerns prioritized. When actually choosing a remediation design, UIEDT considered the following priority:

- reduce contaminant concentration to specified limits
- meet regulations for exiting streams (water. clean soil, etc.)
- minimize or eliminate waste to landfill
- maximize amount of clean soil returned to the land
- minimize costs
- maximize efficiency and up-time of process
- maintain safety as a top priority.

Process Design

Focusing primarily on the maximization of long-term (permanent) immobilization, UIEDT has chosen to actually remediate the sludge rather than stabilizing the contaminants in some form of matrix. Doing so not only ensures long-term safety, but also helps to minimize costs by eliminating waste transported to a landfill. The proposed process involves two primary phase treatments -- organics with bioremediation and heavy metals with immobilization. The 178,357 gallons of sludge is

treated in a series of 22 batches, each containing approximately 8,100 gallons of sludge.

The 10,000 gallon tank, which initially contains only 8,400 gallons of sludge, will be immediately adapted into a treatment tank where sludge from the remaining tanks will be pumped prior to treatment. Three hundred-sixty-five pounds (1% by wt.) of reactive phosphate hydroxy apatite will be added to the 8,100 gallon sludge batch being fed into the tank and mixed for 24 hours. Introduction of the apatite into the sludge, which contains mobile metals and other inorganic contaminants. provides nucleating sites for a reaction resulting in the precipitation of metal-substituted apatite minerals. Metals encompassed in apatite have almost permanent durability and leach resistance that significantly exceeds other stabilized waste forms because the apatite mineral structure is very stable over a wide range of environmental conditions and for geologically-long time periods. The leachability of metals sequestered in apatite are typically well below EPA drinking water standards, have proven insensitive to pH changes over the range of 2 to 12.5, and the bioavailability of ingested metals is minimal. Other advantages of using hydroxy apatite include small material requirements, rapid kinetics, and low cost. The apatite reaction requires less than 1%(wt) of reactant and under 24 hours for formation while the common grouting techniques tend to require as much as 30%-50%(wt) and up to 30 days for "set-up.'' (1) Since the phosphate mineral is naturally occurring in a number of forms ranging from ore to shark teeth and fish bones, it has the potential of being obtained for little more than transport costs from places such as fish canneries. (2)

During the same 24 hour period that the apatite reaction is being completed, a saturated steam coil will be used to evaporate off the organic compounds by raising the bulk sludge temperature from ambient (~27C) to the mixed boiling point of the organic compounds (~121C). During this time, it is also expected that approximately 5% of the water will evaporate off, helping to ensure the removal of the volatiles. UIEDT needs to separate the volatiles from the bulk sludge since the biological organisms used in the organic treatment are sensitive to the salt concentrations found in the sludge. For the organic compounds that might remain in the sludge and not evaporate, UIEDT will add 90 pounds (0.25%wt) of activated carbon to the treatment tank. Activated carbon is a common technique used to trap organic compounds because of the large number of reaction sites and the immediate reaction. However, using activated carbon as an actual treatment has the disadvantage that the used carbon itself must be treated, which can be more expensive than initially treating the volatiles. (3)

A simple, water-cooling condensing unit will trap the volatilized compounds and return them to a liquid phase prior to their being fed into a storage reservoir awaiting biological treatment. Biodegradation of the organics has been chosen because it is one of the few treatment options available that permanently removes the volatiles rather than trapping them or releasing them into the atmosphere in diluted concentrations. An anaerobic reactor is responsible for the degradation of tetrachloroethene (PCE), and will begin the degradation of methylene chloride, 2-butanone, and other volatile organic compounds (VOCs). The most difficult component to degrade is the PCE because it contains four chlorine groups surrounding a double bonded carbon. A reductive dechlorination process has been chosen since it optimizes PCE and VOC

degradation. The biodegradation pathway involves the use of an unidentified microbial population obtained from the anaerobic digester sludge of the primary digester at any local sewage treatment plant. (4) The anaerobic reactor is fed in batches of ~5,000 gallons of condensed organics every ten days, with nine days being required for the actual degradation and one day to transfer batches. Fresh nutrients should be fed at a volume of 0.2% (15 gal) every 24 hours with an equal amount being removed (and later recycled) so that the total concentration remains constant. The reactor is packed with one inch polypropylene mesh spheres for a growth support structure, operates at ambient temperature, and should maintain a pH in the range of 6.8 to 7.5.6

An aerobic reactor is then used to complete the degradation of the anaerobic bioreactor endproducts: trichloroethylene (TCE) and remaining VOCs. A process focusing on the degradation of TCE is used because it is the most difficult constituent left to treat due to the presence of three chlorine groups. Methanotrophs obtained from the upper three inches of a sedimentary pond are used as the microbial population and the methane monoxygenase enzyme responsible for the degradation is produced with the aid of a 5% methane and 3% propane mixture acting as an inducer.7

A continuous stirred tank reactor (CSTR), which is simple and easily available on an industrial scale, will be used as the aerobic reactor. The effluent stream from the anaerobic reactor is fed directly into the aerobic reactor where it is continuously mixed (200 rpm) and fed the inducer at a rate of 0.45 gpm for ten hours. Like the anaerobic reactor, the aerobic degradation is completed at ambient temperature and a pH of approximately 7.2. (5)

The effluent stream leaving the bioreactors primarily contains water, converted compounds, and biomass, and are void of any restricting contaminant concentrations. UIEDT can therefore release it straight into the local sewer for treatment at the waste-water treatment facility. (6)

The sludge remaining in the treatment tank has been sufficiently treated by the hydroxy apatite and activated carbon to immobilize the contaminants in both the water and soil phases. It is pumped directly from the tank into the feed reservoir for a series of two rotary vacuum drum filters (RVDF's). The RVDF's are efficient solid-liquid separators which work by drawing slurry from the feed tank onto a rotating drum, where a cake is formed on a filter cloth as a created vacuum pulls the filtrate through internal drainage channels. (7) Since it is not necessary for extremely high purity in the exit streams, which allows us to take advantage of lower capital, operating, and maintenance costs, RVDF's were chosen over other separation techniques such as sediment centrifugation or plate filtration.

Two streams emerge from the RVDF's: a slightly moist (<15%) soil cake and a water stream. (7) The soil can be moved with a small front end loader and returned to the original evaporation ponds as fill, with the additional benefit of being in a position to trap any further leached contaminants from the site with the hydroxy apatite reaction. (2) The water stream contains no detectable contaminants except for the salt concentrations which result in a moderately high TDS (total dissolved solids) level. Because waste-water facilities use biological systems for treatment, it could be expected that the salts would create a serious problem. However, since the total liquid stream containing the salts is only 138,000 gallons (released at less than 5.800 gallons/day), the high pH stream would be quickly neutralized by the much larger stream

contributions made by the rest of the community. Release of the water stream into the sewer has been approved by the waste-water treatment facility in Las Cruces. (8)
Fig. 1.
The treatment time for each of the 22 batches is 250 hours. based on the 24 hour requirement for the apatite reaction, 9 days for the anaerobic remediation, and 10 hours for the aerobic remediation. Since it can be done simultaneously with the biological treatment. the time necessary for the RVDF's and transporting the soil is encompassed within the 250 hours. The overall time. however, does not depend upon 250 hours for each batch since the bioreactors have been scaled up to accommodate approximately 3 batches at a time, allowing the "treatment tank" stage to run on a constant basis. But, it is necessary to account for the time needed to pump the sludge, volatiles, or water from one stage to the next. The total treatment time is approximately 100 days when also including set-up and training, down time, and take-down.
ECONOMIC ASSESSMENT
In order to determine the feasibility of a proposed project, it is necessary to complete an economic evaluation. The intent of this evaluation is to provide a cost estimate of the entire treatment process, based on three cost categories: capital. operating. and other associated costs. The cost estimate has been completed using both vendor quotes and a series of cost tables and charts found in Plant Design and Economics for Chemical Engineers by Peters and Timmerhaus. The assessment also assumes that detailed variations of the "six-tenths rule" applies to our equipment and that costs of similar processes are closely related to the costs involved in this process. An accuracy of approximately -30% to +30% has been reached with the resulting cost estimates.
The following economic assessment is based on the following:
Total Process Time -- 100 days
Set-up, training, inoculum, start-up ~ 2 weeks
Treatment ~ 72 days
Down time ~ 8-10%
Take down and clean-up ~ 1 week
Operating Schedule -- 7 days/week, 8 hours/day
Equipment Costs -- amortized by % depletion of total production capability over 15 yr. (10 yrs of actual operation)
Various operating and associated costs -- assuming a percentage of capital investment
Capital Costs
For this evaluation, capital costs include equipment and start-up costs, with items such as site preparation and R&D falling under associated costs. Expenses for installation and any necessary electrical or process control are included in the given equipment costs. The equipment costs are amortized on a percent depletion basis, assuming a total "life" of 7,000,000 gallons (180,000 gal/1 00 days for 15 years, with 10 years of actual operation time).
Table II
Table III
Operating Costs:
Labor

The proposed process will run eight hours per day, seven days per week. Since the process requires mostly supervision rather than a constant need for "hands-on" labor, there is not a need for very many employees. During the normal business week (Monday - Friday). the person hired as project engineer/manager will work a typical 40 hour week. A lead operator will also work a 40 hour week, but from Wednesday through Sunday to overlap with the engineer so that a supervisor will always be available. The operators will work approximately 40 hour weeks on rotating schedules. and will be on duty two at a time. UIEDT might need to adjust the 8 hour shifts and worker schedules periodically on those days between anaerobic bioreactor batches for transfer and cleaning. UIEDT will hire the operators locally, while the engineer will be brought in and given a living stipend of $125 per day.

Table IV

Utilities

The predominant utility cost is for electricity to run the pumps, agitators, and vacuum drums. Energy costs are assumed to be $0.08/kW-hr. Steam at 100 psig for the heating coil is priced at $3.50/1000# and the condenser cooling water is $0.10/1000 gallons. The sewer water costs are estimated at $2.20/1000 gallons.

Landfill

If land application is not approved at the site, UIEDT must take the treated sludge to a class C landfill. Landfill costs can be assumed to be $100 per cubic yard, and transport costs to the landfill is approximately 25% of the total landfill cost, which would result in a total disposal cost of almost $25,000.

Table V

Other Associated Costs

Not included in capital or operating costs are other necessary expenditures that can make up anywhere from 5-15% of the total process cost and vary significantly depending upon the actual process. Included in these are permit and regulation costs which are generally the obligation of the responsible party and vary from site to site (as a result, the value included here is only approximate permit fees for this site, and doesn't include legal costs).

Table VI

TOTAL PROCESS COST: $166,000

The total process cost of $166.000 is equivalent to $0.93 per gallon or $145 per ton of the contaminated sludge treated. Operating costs arc equal to 80% of the total cost, and labor itself is almost 70%. The bioremediation phase accounts for the majority of the treatment expenses, with the apatite treatment itself costing less than $40 per ton. Because of the nature of the design, the total cost per ton would decrease significantly with an increase in the volume treated.

REGULATORY AND LEGAL CONSIDERATIONS

Remediation and stabilization of the hazardous waste while preventing contamination of the surrounding land is the primary concern of the University of Idaho Environmental Design Team (UIEDT). In order to maintain credibility with the community and receive cooperation from governmental agencies WERC and UIEDT must comply with applicable environmental laws and regulations.

The legal and regulatory issues that are most important are those governing permitting, waste transport, generation and disposal of waste, and water and air quality. The primary regulatory instruments that are of

concern to UIEDT are the Resource Conservation and Recovery Act (RCRA-HSWA), Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), Hazardous Materials Transportation Act, (HMTA), Clean Air Act, Clean Water Act, OSHA, and finally state administered laws. (6,9) UIEDT will contract the services of a professional environmental consultant with extensive knowledge of federal and local regulations in addition to general permit requirements. Although professional assistance will be obtained, it is useful to discuss general regulatory requirements.

Permitting information is contained in 40CFR part 124. Permits are usually issued by the EPA, state of operation or both. Currently under the Hazardous and Solid Waste Amendments of 1984 (HSWA), until states receive specific HSWA authorization, all permits are handled as joint issuance, assuming the state has the authority to implement pre-HSWA provisions. Signatures by EPA and the state are necessary to provide the facility with authority to operate under RCRA. If the state has no authorization, EPA assumes authority to issue the permit. Specific requirements needed to receive the permit are not covered here, but can be found in EPA's Permit Applicant's Guidance Manual for Land Disposal,. Treatment, and Storage Facilities. (10)

The generation of wastes and the regulations covering those actions are the next concern of UIEDT. As mentioned earlier a myriad of regulatory activity surrounds the generation of disposal of wastes on all government levels. UIEDT will apply, to become a Corrective Action Management Unit (CAMU) through RCRA as regulated in 40CFR sections 260.264. 265,268.270. 271. These rules promulgate provisions under subpart S for CAMUs to be used for the purpose of facilitating remediation waste management activities at RCRA facilities. More specifically for a site remediation project under CAMU any waste managed within the CAMU s, which was generated as part of the corrective action at that facility (i.e., remediation waste) would not be subject to RCRA regulatory disposal requirements. Thus, waste generated from corrective action at the facility may be placed within the CAMU to the technology-based levels established under the RCRA land disposal restrictions (LDR) program. As a CAMU remediation project UIEDT would still be required to meet the remediation standards set by RCRA. (10)

The next concern of UIEDT is the transport of hazardous waste from the site. The transportation of hazardous wastes is regulated by both the Hazardous Materials Transportation Act (HMTA) and RCRA. (9,11) Section 3003 of RCRA required EPA to establish standards for transports and coordinate activities with the Department of Transportation (DOT). (10) WERC is also considered a TSD (Treatment, Storage and Disposal Facility) and is subject to 40CFR Sections 264, containing requirements applicable to new facilities or those possessing a permit. Treatment is any method, technique or process designed to change the physical, chemical, or biological character or composition of any hazardous waste so as to recover energy, or biological character or composition of any hazardous waste so as to recover energy or material resources from the waste, or to render such waste non hazardous or less hazardous: safer to transport, store, or dispose of; or reduce in volume [Section 260.10].

As a TSD, the WERC site should obtain an EPA identification number by submitting EPA form 8700-12 to the regional EPA office (Region VI, Dallas TX. 214-655-6700) or authorized state (New Mexico, 505-827-2924). Depending on the necessity of and business arrangement with the landfill,

WERC may be required to submit a detailed analysis of the transported waste as detailed in Sections 264/265.13.27

Location of UIEDT s operating waste remediation unit may also fall under jurisdiction of the EPA or state environmental authorities. Under Section 264.18. there are location restrictions which include:

Facility must be located 220 feet from an active Holocene fault.

Facilities are not allowed on a 100 year flood plain unless it meets the exemption conditions contained in [Section 264.18(b)]

Location subject to wetland restriction in accordance with Executive Order 11990(Protection of wetlands)

Further information on site-specific location standards can be found in EPA s Permit Writer's Guidance Manual for Hazardous Waste Land Storage and Disposal facilities. February, 1985 (OSWER Directive No. 9472-004).

Security measures are also the responsibility of WERC and are contained in Sections 264/265.14, requiring the prevention of unknowing or unauthorized entry of persons or livestock to the active portions of the facility. Security measures may include one or all of the following: 24 hour surveillance systems around the facility, signs, and barriers to prevent entry to the facility. Regulations governing these activities follow Sections 264/265.14(a)(b)(c).

Regulations regarding training of employees for the TSD facility is outlined in Sections 264/265.16. The training program must familiarize the employees with emergency procedures and emergency equipment shut-off controls. Regulations contained in Sections 264/265.32 require the facility to maintain an internal communications system, external communications system and adequate fire-fighting and response equipment. Communication with local authorities in the case of an emergency is required by sections 264/265.37(a).

WERC is also required to develop a contingency plan that describes the actions facility personnel must take in response to fires, explosions or releases [Sections 264/265.51 (a)]. A well developed contingency plan will follow the sample contained in OSWER Directive No. 9523.00 10.

UIEDT expects the site to close cleanly in accordance with Sections 264/265.11:264/265.112, and will not be subject to post-closure requirements [Section 264/265. 110]. Within 60 days of the closure, a certification must be prepared and submitted to the EPA [Sections 26z./265.225]. WERC may be subject to financial responsibility and liability requirements, Sections 204/265.14C(b) and 264/255.147, respectively. (10)

The Clean Air and Water Acts are of concern to UIEDT and WERC. UIEDT will make a significant effort to remove all contaminants from the water using bioremediation, and the use of phosphate ore. The processed water will be sent to a waste water treatment plant for final purification, after UIEDT has obtained a waste water discharge permit through the State of New Mexico Water Quality Control Commission. (12)

The only concern of UIEDT in the area of air quality is the volatilization of contaminants during remediation. Volatilization will be avoided by using a closed remediation system, thus the WERC remediation with respect to the Clean Air Act is free, for all intents and purposes, from regulatory requirements.

Overall, the areas of concern for UIEDT with respect to the proposed remediation are permitting, waste generation rules, transport, disposal requirements and finally, air and water quality. The regulations have

been presented primarily from a federal perspective, but it is equally important to recognize local and state authority, as mentioned earlier.

HEALTH AND SAFETY

The UIEDT regards the health and safety of the community and workers as top priority,therefore; UIEDT agrees to accept the responsibility of maintaining a safe working environment. The major objectives to uphold include: compliance with industrial safety standards (ACGIH. AIHA. and NIOSH), adherence to regulations (OSHA, EPA, State of New Mexico, City of Las Cruces), maintaining high worker morale, and achieving the most efficient level of clean-up. (13)

Before any work may begin, a Site Health and Safety Plan including medical surveillance of employees and documentation on site worker training is required by OSHA. Also encompassed in the General Health and Safety Plan are: the organizational structure, comprehensive work plan, training programs, standard operational procedures and informational programs. (13)

To meet these objectives in the clean-up process, UIEDT is primarily concerned with the maintenance of good ventilation, proper use of methane gas, worker protection/well trained workers, and a contingency plan in case of an accident. Maintaining good ventilation throughout the process is necessary to minimize the airborne contaminants. The bioreactors produce a substantial amount of CO and the mixing of the hydroxy-phosphate as well as the use of the drum filter could allow for aerosols. Consequently, a hood on-site would be beneficial for both the collection of endproducts as well as for proper air monitoring, which is a factor in the quality assurance program. A daily sample would be recorded in order to compare values and stay within regulations detailed in the FDR's. (10) In the event of a high reading, a warning signal would be activated which would be directly relayed to the project manager for response.

The use of methane gas in the aerobic chamber of the bioreactor demands special attention. The explosive nature of methane gas when it comes in contact with air is a problem that must be addressed. (14) First, all employees would be knowledgeable of the precautions necessary and what to do in the case of a gas leak. Next, specially designed pressure valves and reinforced sealed tanks will be purchased to insure the safe utilization of this gas. In addition, the above ground tanks will be stored in a more secluded area of the site approximately 300 yards away from the main location. Lastly, a methane and propane mixture was chosen because of the increased safety level as well as the enhanced biodegradation of the volatile organic compounds under these conditions. (5)

Protecting the employees is essential for a successful process. It is important that universal precautions are used in all situations, which involves the assumption that all materials in this process are a threat to the health of any exposed person. Following this assumption, protective clothing must be worn and safety regulations must be followed and strictly enforced at all times. This includes appropriate documentation of daily procedures in the event of having to backtrack the operation. In addition, all workers must have proper training about the contaminants on site and be familiar with the contingency plan. This type of instruction is fundamental to the proper execution of this process. To attain this knowledge, each employee will be required to take a 40 hour OSHA hazardous materials course which teaches precautions and appropriate preparation as contained in 29 CFR 1910.120 HAZWOPER TRAINER. (13)

Unfortunately the possibility of an accident will always be present, even if the requirements are met and the proper procedures are followed in this situation, a contingency plan would be necessary for the maintenance of UIEDT's high safety standards. The plan should include a flowchart which would clearly state the steps that must be taken to contain contamination and who to contact. The procedures outlined in the Material Safety Data Sheets (MSDS) for the chemicals involved in the process will guide emergency response and containment action. (15) Due to the precautions taken by the UIEDT, the worst case scenario is not likely, but the employees will be trained to be prepared for almost any situation.

COMMUNITY PLAN

The UIEDT recognizes the importance of developing a good relationship with the community regarding the removal and treatment of hazardous wastes. Cooperation will be essential for a successful process. To start the formation of a good community relationship, the following objectives regarding community relations should be instigated: encouraging public involvement, employment of effective communication, and the instigation of widespread education.

Encouraging public involvement in the remediation plan is a method for attaining cooperation from the community. The people in the surrounding areas have a right to be informed and a right to participate in any decisions that may affect their health or property. (18) It is imperative that WERC acknowledges these privileges and intends to involve the public from the beginning by allowing them to review any proposals before the important decisions are made. Any suggestions should be considered and a possible open forum setting could prove advantageous because this would provide an opportunity for individual expression. Additionally, articulation should be encouraged through a "suggestion box" in the city hall. A monthly committee composed of WERC employees and administrators could meet to answer these questions and distribute the answers in a newsletter format.

Effective communication is essential in any relationship, especially when one of the parties is less informed. Most importantly, WERC should develop honest open ties with the citizens. To attain this alliance, any concerns should be addressed immediately and questions should not be avoided but answered in an straightforward fashion. Furthermore. jargon and technical language should be kept to a minimum because they serve as barriers to effective communication. This includes limiting acronyms, abbreviations and raw data; instead expanding personal experiences and anecdotes. Ultimately, the interactions should be on a more personal level instead of on an abrupt detached level.

Finally, the instigation of widespread education would help to produce a more informed public. The significance of providing informative seminars where the general public can learn more about the processes involved in hazardous waste treatment is well documented. If the people can be taught the importance of treating the wastes and how it can benefit them in the future, they will most likely be more willing to cooperate. Another way to educate the public is to use the media in an effective manner. In today's society, many busy Americans depend on the media to provide information, so WERC should be open and accessible to reporters and members of the media.

In conclusion, when the people are kept involved and informed, they are often more interested, realistic, and cooperative. These factors can mean

the difference between a successful project and a project plagued with continuous problems.

FINAL CONCLUSIONS

UIEDT's proposed treatment design for the contaminated evaporation pond sludge fulfills the requests and meets the requirements presented by WERC. The treatment process chosen -- involving bioremediation, apatite immobilization. and a solid-liquid separation technique -- is designed and optimized to be a successful, efficient and cost effective alternative. With this process, the contaminated sludge is remediated to a form which allows the solid portion to be returned to the site for land application and the liquid portion to be fed directly into the local sewer. Literature and experimental results confirm that the contaminant concentrations are reduced to levels well below the requirements for 30 year LDRs (WERC rqmt.), DOT (WERC rqmt.), and also for UIEDT's intended application. Analysis and reviews also confirm the permanent stability of the process, resulting from the fact that the sludge is remediated rather than just stabilized.

This proposal includes a preliminary technical design for the described treatment site, and also addresses the economic, environmental, safety, and social considerations. Additional information concerning the site would be required for a more detailed process design, and further on-site tests should be performed to confirm the large scale applicability of the process.

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22-5

STABILIZATION OF A SOLAR POND SLUDGE FOR LAND DISPOSAL

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ABSTRACT

A solidification and stabilization (S/S) technique was developed for a chemically complex mixed waste sludge containing nitrate processing wastes, sewage sludge, and electroplating wastwaters. The sludge has high concentrations of nitrate salts; cadmium, chromium, and nickel concentrations of concern; and low levels of organic constituents and alpha and beta emitters. Because this waste is classified as both a listed hazardous waste and a low-level radioactive waste, land disposal

in a mixed waste landfill was identified as the only suitable method for the long term management of the sludge.

Preparation of the sludge was required to ensure compliance with all land disposal regulations. The S/S process identified and optimized for this sludge included sulfide reduction of nitrate and precipitation of metallic species, followed by evaporation to dryness and solidification of the dry sludge in recycled high density polyethylene with added lime. A modified Toxicity Characteristic Leaching Procedure was used to determine required treatment chemical dosages and treatment effectiveness. An economic assessment of the cost of this project showed over 50% of the total cost can be attributed to the transport and disposal of the solidified product in a mixed waste landfill.

The implementation of this process will involve compliance with several environmental and health and safety regulations. RCRA hazardous waste generator and NPDES permits are needed for this process. OSHA standards require adequate monitoring and protection of on site employees from exposure to radiation and chemical hazards.

INTRODUCTION

Solidification and stabilization (S/S) are established and widely used technologies for the preparation of hazardous wastes for land disposal. These techniques effectively reduce the mobility of hazardous constituents and provide a product with a suitable structural integrity for compliance with RCRA Land Disposal Restrictions (LDRs) (40 CFR 268). S/S has been incorporated into 28% of the CERCLA record of decisions written between 1982 and 1992, making it the single most widely used technology for hazardous waste sites (1).

Many S/S applications involve the use of cement, with or without a pozzolanic material, to bind the waste material into a solid monolithic form (2). These materials are most widely used because of their low cost and readily available processing equipment. The major limitation of cement-based S/S applications is the potential for incompatibilities between constituents in the waste and the cementation reactions. This is particularly true for wastes with a high ionic strength; many types of salts increase the setting times and decrease the durability of the cured concrete (3).

The research described here was conducted to find a suitable S/S process for a chemically complex mixed waste. The waste sludge has been dredged from solar evaporation ponds and is currently waiting for further processing before its ultimate disposal. The evaporation ponds had been used to reduce the volume of a wide variety of wastes including nitrate processing solutions, sewage sludge and electroplating wastewaters. The sludge is characterized by high concentrations of nitrates and heavy metals (Cd, Cr, Ni), low concentrations of several volatile and semi-volatile organic compounds, and low levels of alpha and beta radiation emitters. The precipitation of insoluble inorganic complexes and wind blown dust both contribute to the approximately 30% solids concentration in this sludge.

Regulations addressing two specific characteristics of this waste material effectively limit the ultimate fate of this waste to land disposal. First, the presence of alpha and beta radiation emitters classifies this waste as a low-level radioactive material. Storage in a regulated land disposal unit is the only option for the long-term management of these wastes. In addition, since the ponds were used for the treatment of electroplating wastewaters, the entire waste stream is

considered a listed waste (F006) and, thus, any byproducts derived from the treatment of this waste will also be considered a listed waste. The preparation of this waste for a mixed waste landfill by S/S was the only option suitable under current RCRA and NRC regulations.

The objective of this paper is to present the procedure and rationale used to develop an appropriate S/S process for this chemically complex waste sludge. Several factors were considered in this assessment; the most critical criteria included:

- immobilization of the heavy metals to comply with the Universal Treatment Standards (UTSs) defined under the LDRs of RCRA (40 CFR 268);
- reduction of the concentration of the highly reactive nitrates;
- production of a solid material with a structural integrity for compliance with RCRA and NRC regulations; and,
- minimization of the volume of waste requiring land disposal.

Considerations for the regulatory, economic, and health and safety issues that are impacted by this design are also discussed.

PROCESS IDENTIFICATION

Solidification Processes

The most important criterion for choosing a suitable binder to solidify the waste material is the compatibility between constituents in the waste and the characteristics of the solidification agent. Of the many constituents in this waste, those contributing to the high ionic strength were expected to cause the greatest incompatibilities with S/S techniques. Table I presents the approximate total concentration of primary ions in the sludge, illustrating the extremely high concentrations of sodium, potassium, and nitrate ions. Organic constituents were present at only low concentrations (10-100 ppb) and were not expected to cause any incompatibility problems. Literature sources describing the application of S/S processes to other complex waste streams were reviewed to identify an appropriate technique for this particular waste.

Table I

A significant amount of work has been completed showing the feasibility of encapsulating complex chemical wastes in polyethylene (4-5). In this process, dried waste is mixed with the polyethylene in a heated extruder. Microencapsulation of the waste in the plastic matrix effectively reduces the amount of leaching and greatly increases the structural durability of the waste material. The use of post-consumer polyethylene has the added benefit of using a waste product to treat a waste.

Polyethylene encapsulation has been used successfully for nitrate salt wastes with low levels of radioactivity (6-8). This technique has been shown to greatly reduce the leachability of salts; provide a solidified matrix that greatly exceeds durability standards set for both hazardous and nuclear wastes; and, be readily implemented at a full-scale operation (6,8). Previously published research on the polyethylene solidification of nitrate salts indicates, however, that the heavy metals may not be sufficiently immobilized by this process. For example, Kalb et al. (6) report a concentration of 3.6 mg/L chromium in the leachate generated from the toxic characteristic leaching procedure (TCLP). This concentration is sufficiently low to prevent the waste from being identified as hazardous based on its toxicity characteristic (40 CFR 261.24). However, for the land disposal of listed wastes, the TCLP leachate must have concentrations below the Universal Treatment

Standards. The UTS for chromium is 0.86 mg/L (40 CFR 268), well below the concentration achieved by Kalb et al. (6).

Based on the successful implementation of polyethylene encapsulation of nitrate salt wastes, it was decided to pursue this technique for the mixed waste considered in this research. The overall process for the solar pond sludge, however, requires additional processes: the waste must be dried prior to solidification in polyethylene; and, the heavy metals immobilized to meet the stringent UTSs for land disposal of a listed waste.

Chemical Stabilization Processes

Stabilization of metallic species is accomplished by conversion of the metal into an insoluble complex by way of chemical reaction. The insoluble solid form can then be separated from solution. Metals are often rendered insoluble by precipitation in a metal-sulfide, metal hydroxide, or metal carbonate form, or reduction to metallic form by reaction with a borohydride. For economic reasons sulfide and hydroxide precipitation are the more attractive options.

At the high pH conditions found in the waste, precipitation of metals as metal hydroxides would seem a reasonable approach. All three metals of concern (Cd, Cr, Ni) form fairly insoluble hydroxide complexes. Aqueous concentrations of less than 1.0 mg/L are theoretically achievable (9). Were it not for the amphoteric behavior of metal hydroxides, all of the metals would be stabilized by this method to a degree which could approximate the desired insolubility. However, the metal hydroxides are increasingly soluble at both low and high pH, and the pH of minimum solubility changes with metal and with solution characteristics. No optimum pH value exists at which all three of the metals of interest can be rendered insoluble to the desired degree.

Cadmium and nickel readily form metal sulfide species (CdS, NiS) which are also fairly insoluble. Nickel forms a sulfide complex of similar solubility as the nickel hydroxide complex, while binding cadmium as CdS reduces its aqueous solubility in dilute solution to approximately 10-7 mg/L at pH 11 (9). This greatly reduced cadmium solubility is desired in order to render the metal immobile and meet the UTS.

On addition of the sulfide based reducing agent to the waste, other oxidation/reduction reactions also take place. Nitrate salts in the mixture will be reduced to nitrite and the nitrite to more reduced nitrogen complexes. Reduction of nitrate salts to other nitrogen species is beneficial in that both nitrate and nitrite salts are strong oxidizers. In dried form, they can react violently with combustible materials, resulting in violent combustion or explosion (10).

Summary of Appropriate S/S Process

Based on a review of the chemical characteristics of the solar pond sludge and literature applications of S/S technologies, an overall strategy involving the stabilization of the waste through the addition of sodium sulfide and lime and solidification in polyethylene was identified as appropriate for this waste. The addition of sodium sulfide has the dual role of precipitating both nickel and cadmium as very insoluble complexes and reducing the nitrate to less reactive nitrogen species. The reduction of the nitrates has an added benefit of reducing problems associated with incompatibility. The addition of hydroxide as lime (CaO) ensures the maintenance of a high pH so that the precipitation of chromium in the hydroxide form is maximized.

One drawback to this process is the generation of a secondary by-product stream as water is separated from the waste material prior to encapsulation in polyethylene. Traditional separation processes for water and solid, such as filtration were rejected due to the very high dissolved solids concentration - including radionuclides - that would remain in the filtrate. Instead, evaporation of the water was identified as the best technique. Pilot-scale testing by Faucette et al. (8) identified that a horizontal thin film evaporator produced a dry salt from an aqueous nitrate salt waste with the most desirable physical characteristics for polyethylene solidification.

EXPERIMENTAL METHODS

Due to the very high ionic strength in the solar pond sludge, it is not feasible to predict the appropriate dosages of treatment chemicals based on chemical equilibrium calculations. Experimental testing was necessary to optimize the overall S/S process identified above.

A surrogate waste which approximated the ionic composition of the raw waste as indicated in Table I was prepared from reagent grade chemicals, water and kaolinite. Radioactive materials present in the solar pond sludge were omitted from the surrogate waste in order to avoid worker exposure to radioactivity. This mixture was allowed to react overnight before proceeding with treatment. Chemicals used to treat the surrogate waste mixture were sodium sulfide, lime, and polyethylene pellets.

Sodium sulfide was mixed with the surrogate waste mixture in separate batch experiments and allowed to react for 12 hours on a wrist action shaker before testing pH, nitrate and nitrite concentration in the mixture. In this manner the sodium sulfide dose required to fully reduce the nitrate and nitrite species was determined. Nitrate and nitrite concentrations were determined using commercially available colorimetric test strips (EM Quant, EM Science, Gibbstown, NJ).

Once the range of sodium sulfide dose required for reduction was determined, batch treatment of the waste was performed in a more limited range around this dose to determine the optimum dose for immobilization of the cadmium as CdS. Identification of the optimum dosage was determined based on the concentrations of metals in the TCLP extract from treated and solidified samples. Lime, if added for a particular test, was added to the dried waste mixture prior solidification. The drying and solidification steps mimic the procedure used to stabilize the waste in the full treatment process.

Drying was performed by heating the sulfide treated portion of the waste in a distillation apparatus until nearly dry. The waste mixture was removed from the distillation flask into a blender (Waring, Winsted, CT) for grinding to facilitate mixing with polyethylene.

A bench-scale mixing and extrusion apparatus was constructed to perform the mixing of treated, dried waste and polyethylene pellets. An aluminum tube with a 4.7 cm inner diameter and 17 cm in length was fitted with aluminum end caps. The tube was filled with a mixture of waste, lime and polyethylene pellets, placed on a laboratory hot plate and wrapped in heating tape along its entire length. The hot plate and heating tape were controlled by a rheostat in order to prevent excessive heating of the mixture. As the mixture melted in the tube, it was periodically mixed manually with a stir rod. Approximately one hour was required to fully melt and mix the components. The polyethylene and treated waste mixture was extruded from the tube by removing the end caps and pushing the mixture from the tube into a metal mold using a metal rod fitted to the

diameter of the tube. After cooling for ten minutes, the monolith was removed from the mold and used for further testing.

Leaching of the contaminants from the monolith was performed by a slight modification of the standard technique for TCLP. As metal species were the only components of interest, small polyethylene bottles (Nalgene) were substituted for the zero-headspace reactor. A bandsaw was used to cut the monolith into cubes not larger than 9.5 mm per side. Extraction fluid for an alkaline sample (TCLP Fluid #2, 40 CFR 261) was prepared and its pH tested to be 2.880.05. The final solution pH was verified and recorded. The extract was isolated from the waste cubes and diluted as necessary for analysis.

Metals analysis of the leachate was performed by graphite furnace AAS using a Perkin-Elmer Zeeman 5000 spectrophotometer. The method of standard additions (11) was utilized to minimize matrix interference effects.A Universal Testing Machine (Tinius Olsen) was used to perform the unconfined compressive strength test on the final monolith according to a standard testing protocol, ASTM D 2216-91. Freeze-thaw durability of the monolith over 12 cycles of freezing and thawing was determined by ASTM D 4842-90.

EXPERIMENTAL RESULTS

The initial batch tests were conducted to identify the sodium sulfide dosage required to reduce the nitrate species. Treatment with doses of sodium sulfide between 0 and 150 g/L were considered. The nitrate in solution was reduced to below detection following addition of approximately 85 g/L of sodium sulfide. Nitrite concentration was near the detection limit in the untreated waste and remained below the sensitivity of the test strips at all sodium sulfide dosages.

The dosage of sodium sulfide needed to stabilize the cadmium as CdS was then determined. Dosages ranging from 70 to 120 g Na_2S per liter of the surrogate waste were considered. The treated waste was then dried, mixed with polyethylene on a one-to-one basis by weight, extruded, cooled and subjected to the TCLP. The average cadmium concentration in the TCLP leachate as compared to Na_2S dose is given in Fig. 1. The UTS limit for cadmium of 0.19 mg/L is indicated by the horizontal line. Only one Na_2S dosage tested, 110 g/L, was able to meet the UTS requirement. Nine separate extractions were performed at this dosage to verify this result. Subsequent tests were performed using this treatment dosage.

A series of lime doses were added to the treated dried waste to ensure precipitation of the chromium in the mixture as $Cr(OH)_3$. Figure 2 displays the results of average chromium concentration in the TCLP extracts. These samples were prepared from 50% by mass polyethylene, the indicated percent by weight of lime and the balance dried waste. The horizontal line reflects the UST of 0.86 mg/L. Three individual tests at a 10% lime dose showed the UST could be achieved by this process. Nickel concentrations in the leachate from this product averaged 0.17 mg/L, well below the UTS of 5.0 mg/L.

Fig. 1 & 2

The polyethylene monoliths also met requirements for durability. The NRC requires a minimum unconfined compressive strength of 60 psi for generic waste forms containing low-levels of radioactivity (5). The waste material generated in this work had a measured compressive strength greater than 450 psi. This value was not affected by 12 cycles of freeze and thaw conditions. The freeze/thaw test also showed long-term

durability with insignificant mass loss (0.05%) over the period of this procedure.

Water evaporated during the drying of the waste was condensed and tested. There were no detectable concentrations of cadmium and chromium in the condensate, although a significant amount of hydrogen sulfide gas was emitted during the drying, which dissolved in the condensate. In the laboratory, hydrogen peroxide was used to promote the oxidation of the sulfide to sulfate ions. The treated condensate contained 54 mg SO4/L.

OTHER DESIGN CONSIDERATIONS

Regulatory Constraints

The Resource Conservation and Recovery Act (RCRA) and regulations under the Nuclear Regulatory Commission place severe limitations on the handling and final disposition of this waste material. The combined identification of this waste as a listed hazardous waste (F006) under RCRA, and a low-level radioactive waste effectively limits the ultimate management of this waste to landfilling in a mixed waste facility. Prior to landfilling, however, the waste must be processed to meet the constraints of the Land Disposal Restrictions defined under RCRA (40CFR 268). These restrictions require that there be no free water associated with the waste, that the leachate generated from the waste product by the TCLP have concentrations less than the Universal Treatment Standards, and that the waste have a suitable structural integrity. The testing summarized above indicates that treatment of the waste with sodium sulfide (110 g/L), evaporation of water, and mixing/extruding with lime (10%) and melted polyethylene (50%) produced a product meeting the requirements of the LDRs.

Although the on-site treatment of this waste will be conducted under a CERCLA record of decision, a RCRA permit will still be required to obtain status as a RCRA hazardous waste generator (40 CFR 262.12 and 262.20). This permit will provide the EPA identification number required for manifesting and transporting the solidified waste to an off-site disposal facility. The waste should be transported by a licensed hauler in trucks that are placarded according to 49 CFR 177.

Because this process generates a secondary waste stream, the evaporated water, additional regulations for this byproduct must also be considered. This stream is considered a listed waste since it is derived from the treatment of a listed waste. Analysis of the water evaporated and condensed during the drying operation showed that it contains no detectable levels of the heavy metals included in the F006 waste. Radionuclides in the original waste are present in ionic forms and will be concentrated with other salts as the water molecules are vaporized. Thus, it is recommended that an application for the delisting of this waste material be completed. This procedure is lengthy but will greatly reduce the volume of waste material that will require disposal in a landfill.

Assuming that the evaporated water is successfully delisted, additional regulations limiting discharges to the atmosphere (Clean Air Act) or a water body (Clean Water Act) will have to be considered. Significant concentrations of dissolved hydrogen sulfide gas were detected in the condensed water. While this gas will not render the water "hazardous, " it cannot be freely discharged to the environment. Laboratory treatment of the condensate showed that the sulfide ions can be successfully oxidized to sulfate ions. The concentration of sulfate ions (54 mg/L) in the condensate is very low relative to the proposed Safe Drinking Water

Act maximum contaminant level goal for this constituent of 500 mg/L. There should be no trouble obtaining a National Pollutant Discharge Elimination System (NPDES) permit for the discharge of the condensed water to a local surface water body.

Health and Safety

Additional regulations restricting the actions and exposure of employees to hazardous materials will also be applicable to this treatment process. These regulations include subparts of OSHA general industry standards (29 CFR 1910) and CERCLA (40 CFR 311)

The primary concern for the workers involved with the waste solidification process is the potential for exposure to alpha, beta, and X radiation. The X radiation is a byproduct of Bremmstrahlung reactions as beta particles are attenuated by shielding materials. With worst case assumptions, it was determined that all employees must maintain a distance of at least 1.5 m from all process equipment to ensure that radiation dosages are maintained below the allowable limits (29 CFR 1910.96). Film badges should be worn to monitor exposure levels and a lockout/tag out fence employed to maintain an adequate distance between employees and process equipment.

Monitoring and personal protective equipment should also be available for other potential hazards to employees. Grinding equipment for the treated dried waste should be enclosed and ventilated to minimize the inhalation of metal-laden dust. Other process equipment should be ventilated to ensure compliance with carbon monoxide and hydrogen sulfide air quality limits. Carbon monoxide can be produced during the polyethylene extrusion process while the hydrogen sulfide is a byproduct when sodium sulfide is added to an aqueous solution. The high pH of the waste will help to minimize the H_2S production.

Economic Considerations

The cost of processing 180,000 gallons of the solar pond sludge in a 6 month period was assessed. The cost of disposal of the solidified product in a mixed waste landfill was the most significant cost associated with this process. Rates for disposal vary between $360 and $1000 per cubic yard. Compared with equipment rental, utility and personnel expenditures, the disposal cost represents 50 to 75% of the total budget. Thus, any additional steps to minimize the volume of waste would lead to a reduced project cost.

Other researchers have used much lower polyethylene to waste ratios in their application of this technique for the solidification of waste materials. Monoliths with as little as 30% polyethylene have been successful for both incinerator ash (4) and nitrate salts (7). This variable was not considered in this research. With the small scale equipment used to conduct these experiments, it was difficult to achieve a homogeneous distribution of the waste product in the melted polyethylene. This became more problematic with a decreased fraction of polyethylene. The decreased polyethylene:waste ratio would substantially reduce the costs associated with the purchase of polyethylene and transport and disposal of the solidified product. With 30% polyethylene, however, it is expected that the concentrations of cadmium and chromium in the TCLP extract from these samples would increase. Kalb and Adams (7) found that with an decrease in the polyethylene fraction from 50% to 30% the cadmium concentrations increased by nearly five fold. Similar increases with the waste used in this work would restrict this solidified waste from land disposal. Further experimentation with full-scale

equipment would be required to adequately assess if higher waste loadings -- used to reduce the disposal costs -- can still met regulatory limits.

SUMMARY AND CONCLUSIONS

The solar pond sludge considered in this research was a complex mixture characterized by a very high ionic strength, elevated concentrations of heavy metals and low levels of radionuclides. Because of the presence of radionuclides and the classification of this sludge as a listed waste under RCRA, disposal in a mixed waste landfill was identified as the only appropriate management technique for this waste. With the complex chemistry of this waste, it was necessary to treat the sludge prior to solidification for land disposal. Two primary objectives were considered during the chemical stabilization of the waste: reduction of the reactive nitrate concentration and immobilization of heavy metals.

Experimental optimization was used to fully develop a S/S process for the solar pond sludge. The overall process includes the addition of sodium sulfide (110 g/L) to reduce nitrates and precipitate cadmium; evaporation of water; the addition of lime to precipitate chromium; and microencapsulation in a polyethylene monolith. The stabilized and solidified product met federal standards for durability and leachability making it acceptable for disposal in a mixed waste landfill.

An economic assessment of the cost of this project showed over 50% of the total cost can be attributed to the transport and disposal of the solidified product in a mixed waste landfill. Further work at a pilot-scale should be completed to assess the possibility of reducing the percentage of polyethylene in the final solidified product.

The implementation of this process will involve compliance with several environmental and health and safety regulations. RCRA hazardous waste generator and NPDES permits will need to be obtained. OSHA standards will require adequate monitoring and protection of on site employees from exposure to radiation and chemical hazards.

ACKNOWLEDGEMENTS

This research project was originally undertaken by a group of Clarkson University students participating in the 1995 Environmental Design Competition sponsored by the Waste-management Education and Research Consortium (WERC). This consortium is based in Las Cruces, NM and the competition directed by Dr. Abbas Ghassemi. The Clarkson students received the "Best Bench-Scale Demonstration" and "First Place" awards for their proposed solution to this problem.

Additional experiments were conducted by Michael Reardon and Matthew Abbato, Master of Engineering students, and AAS analysis completed by Ms. Eleanor Hopke, Laboratory Technician.

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SPARTAN ENVIRONMENTAL TECHNOLOGIES PROPOSAL FOR REMEDIATION OF POND SLUDGE

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ABSTRACT

The Waste-Management and Research Consortium (WERC) requested a proposal to remediate and stabilize sludge, which was pumped from three solar evaporation ponds into 10,000 gallon tanks. The sludge was composed of a combination of sediments, water and dissolved and undissolved salts. The contaminants of concern in the sludge included heavy metals, low level radionuclides and organic chemicals (2). Because the sludge contained both hazardous compounds and radionuclides, it was required to be managed as a mixed waste regulated under the Resource Conservation and Recovery Act (RCRA) (1) and the Atomic Energy Act (AEA) (3).

The process proposed by Michigan State University ensured the sludge would be treated, stabilized, transported and disposed of in accordance with all applicable federal, state and local regulations. The design criteria that was used for this proposal is as follows:

- Minimization of the volume of mixed waste to reduce total remediation and disposal costs
- Maximization of the rate at which the sludge is processed
- Compliance with all federal, state and local regulations
- Development of positive community relations
- Minimization of negative health impacts by observing stringent safety considerations

At the time of the proposal the costs of disposing of a mixed waste was approximately 100 times greater per cubic foot than for disposing of a low level radioactive waste (4), thus the primary goal was to reduce the

volume of mixed waste in order to lower disposal costs. Volume reduction was achieved through a remediation process which removed the heavy metals and organic contaminants from the pond sludge. The proposed process was a two step acid extraction, in conjunction with cyanide oxidation followed by chromium reduction and metals precipitation. The treated sludge could then be disposed of as a low level radioactive waste outside the jurisdiction of RCRA.

INTRODUCTION

In 1995, the Waste-management and Research Consortium (WERC) requested a proposal to remediate and stabilize sludge from three solar evaporation ponds. Proposals were requested in an attempt to spark new ideas in the minds of students and to evaluate innovative technologies for the remediation of this site. Michigan State University formed a design team consisting of 22 students from four departments: civil, environmental and mechanical engineering and crop and soil science. These students formed a "company" called Spartan Environmental Technologies (SET) to investigate the problem and seek solutions which would be both physically and economically feasible. The following report details SET's research and the proposed process for remediation.

TECHNICAL ASSESSMENT

The sludge was composed of a combination of sediments, water and dissolved and undissolved salts. The contaminants of concern in the sludge included heavy metals, low level radionuclides and organic chemicals (2). Because the sludge contained both hazardous compounds and radionuclides, it was required to be managed as a mixed waste regulated under the Resource Conservation and Recovery Act (RCRA) (1) and the Atomic Energy Act (AEA) (3).

Several treatment and/or stabilization alternatives were investigated and evaluated based upon the following design criteria: time, economics, feasibility, efficiency, waste minimization, safety and community relations. Table I details the treatment alternatives that were investigated. At the time of this proposal, the options available for mixed waste disposal were extremely costly (4). Therefore, particular consideration was given to those processes demonstrating significant mixed waste volume reduction.

Table I

The proposed treatment process was a two step acid extraction, in conjunction with cyanide oxidation, chromium reduction with sodium sulfite followed by metals precipitation with sodium hydroxide and sodium sulfide (Fig. 1). Research done by SET showed that acid washing effectively removed the metals from the sludge. The proposed design would allow for more than 98% of the solid waste generated to be delisted, resulting in only 1,100 gallons of the solid material requiring disposal as mixed waste.

To prepare the mixed waste for transport and disposal, stabilization using polyethylene was chosen. Polyethylene greatly reduced the disposal volume over commonly used technologies. In addition, the stabilized waste passed compressive tests, toxicity characteristic leaching procedure (TCLP) tests and was shown to be stable in hostile environments (11).

Fig. 1

PROCESS DESIGN

The radioactive and non-radioactive metals were extracted from the sludge using sulfuric acid. The pond sediments contained an appreciable concentration of cyanide. To prevent the release of hydrogen cyanide

(HCN) during extraction, the sludge was first treated with sodium hypochlorite (NaOCl) to oxidize the cyanide (CN-) to carbon dioxide and nitrogen gas (N2). Ozone was studied as an oxidation alternative for sludges with high concentrations of phenol, pyrene or other organic chemicals. Recent work has shown the applicability of using ozone to oxidize these compounds in soils and sediments (7,13).

The complete removal of the radionuclides from the sediments was not feasible. Therefore, the treated sediments were neutralized with sodium hydroxide, dewatered using a plate filter press and could be disposed of as a LLW. Prior to disposal, it was recommended that a delisting petition be submitted.

Following acid extraction, the acid solution containing dissolved metals was treated with sodium sulfite. During cyanide oxidation, trivalent chromium was oxidized to hexavalent chromium. Sodium sulfite reduced the chromium back to its preferred trivalent state. Sodium hydroxide and sodium sulfide were added to precipitate the metals out of the acid solution as metal hydroxides and metal sulfides. Ferric chloride was added to aid in flocculation and settling of the metal precipitates. The metal precipitates were dewatered using a plate filter press. The extracted water from dewatering was returned to the inlet of the metals precipitation/ sedimentation tank to undergo a second acid extraction. The dewatered mixed waste was stabilized with polyethylene in a screw extruder and would ultimately be disposed of at a mixed waste facility. The effluent from the precipitation/ sedimentation tank was neutralized using sulfuric acid and discharged to a publicly owned treatment works (POTW). The treatment process is illustrated in Fig. 1. The equipment and chemicals for the proposed design are shown in Table II.

Table II

PROTOTYPE DESIGN

A prototype was designed to demonstrate the feasibility of the proposed treatment plan. The design was based on extensive laboratory analysis of the surrogate sediment. The surrogate sediment was prepared according to specifications provided by WERC (2,14). In accordance with the proposed full scale design, the metals were extracted from the surrogate sludge in a two stage process, first with 6 N sulfuric acid followed by 0.1 N sulfuric acid. The sulfuric acid solutions were added to the sludge in a 5 L basin equipped with a mixing shaft. Following extraction, the supernatant was siphoned into a second 5 L basin where 6 N sodium hydroxide, 25 mg/L sodium sulfide and 10 mg/L ferric chloride were added. The mixture was flocculated and allowed to settle. The extracted metals were first dewatered with vacuum filtration and then stabilized using polyethylene.

Due to the expense of a screw extruder, a steel mold was used to simulate the ability of polyethylene in retaining the waste during TCLP tests. The mold used for stabilization of the precipitated metals was a 5" x 4" x 4.5" block of steel that has a hollow interior measuring 4" x 3" x 3". The top of the mold was machined so one inch of steel would fit inside the mold to compact the mixture of polyethylene and waste metals. Eight vents were strategically placed to allow excess polyethylene to bleed from the mold. The bottom of the mold was attached using six countersunk machine screws, allowing for easy removal of the solidified block. The mold was placed in a compacting device consisting of two fixed steel plates. A 4 ton hydraulic jack was used to compact the mixture. A movable plate was then placed under the mold to maintain the pressure on the

mixing during melting. The hydraulic jack was then removed and the mold was placed in an oven for 16 hours at 180C. Following the 16 hour heating period and a cooling period, the sample was ready for disposal.

LABORATORY ANALYSIS AND RESULTS

Laboratory treatability studies were conducted using a surrogate sediment containing cadmium, chromium and nickel which are representative of the metals present in the actual pond sludge. Studies were conducted to determine the following:

- the efficiency of metal extraction from the sediment
- the removal of metals from the acid extract
- the ability to stabilize the material to meet RCRA standards

Preliminary studies were conducted to determine the efficacy of solvent extraction using di(2-ethyl-hexyl) phosphate (DEHPA) in hexane for the removal of the regulated metals from the sediments. The extraction coupled with acid washing produced favorable removal efficiencies. However, chromium could not be separated from the solvent phase. Without the ability to recycle the DEHPA, the cost increased significantly making the process economically unfavorable. In addition, the kaolin in the sediments could not be easily separated from the solvent phase. These problematic areas dictated the discontinuation of the solvent extraction studies.

Preliminary acid extraction studies were conducted using the original surrogate formulation provided by WERC (containing 1.7 wt % kaolin) (2). The results indicated that significant metals removal was achieved when the pH of the solution was less than 1. Subsequent studies were conducted using the revised formulation updated by WERC (containing 11 wt % kaolin) (14). To compensate for the possibility of lower metals removal due to a higher clay content, a more concentrated acid solution was used.

The efficiency of the acid extraction procedure was evaluated using a mixture containing approximately 410 mL of 6 N sulfuric acid and approximately 1,200 g of surrogate sediment. The acid and sediment were mixed for two hours followed by separation by filtration. The acid extraction process was repeated twice using 1,800 mL of 0.1 N sulfuric acid.

TCLP tests were conducted on the untreated sediment as well as the treated sediment after each extraction step. The results for these analyses are provided in Table III. TCLP and compressive testing of the stabilized metals sediment were also conducted and the results are shown in Table IV. These results indicate that sufficient metals removal from the sediments, to meet land disposal requirements, was achieved after two acid extractions. Results for the stabilized material shows that samples passed both compressive strength requirements and TCLP test requirements. After stabilization, the mixed waste met regulations for disposal at a suitable site.

Table III

Table IV

The metals present in the acid extract were precipitated out of solution by raising the pH of the extract using sodium hydroxide. As shown in Fig. 2, except in a narrow pH range (Cd, pH 9), the treated extract slightly exceeds the most stringent of standards for the discharge of a new source metal finisher to a POTW (40 CFR, Part 433). For this reason, combined metal hydroxide/ metal sulfide precipitation was used in the process in order to meet the standards.

Fig. 2

Polyethylene was chosen as the stabilization technique because of its ability to pass TCLP and compressive tests at high waste loading as indicated in Table IV. Advantages of polyethylene include: it's ability to withstand bacterial degradation, radiation effects, freeze/thaw conditions and biodegradation (10).

ECONOMIC ASSESSMENT

The proposed cost for the treatment and disposal of the pond sludge at the time of the contest was estimated at 5.8 million dollars or $1,840 per ton of sludge. The costs were determined after a thorough investigation of the capital, operating, material, transportation, labor and legal fees (see Table V).

Table V

The costs for the capital equipment were based upon the purchase of the equipment. These costs were amortized over 10 years at an interest rate of 9%. It was assumed that 1 year of equipment operation will be charged to this project to account for actual operating time, down-time for maintenance, or idle time. It was also assumed that the salvage value of the equipment at the end of 10 years will be $0.

BUSINESS PLAN

The proposed sludge remediation process would be completed in several stages over a period of 18 months. Figure 3 represents the proposed time schedule for the process.

Fig. 3

LEGAL AND REGULATORY CONSIDERATIONS

All proposed action for this remediation would be accomplished in compliance with all applicable federal, state and local regulations. The major statutes that would affect the remediation are as follows: AEA (3), RCRA (1), National Environmental Policy Act (NEPA) (46), Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) (47), Clean Water Act (CWA) (48), and the Occupational Safety and Health Act (OSHA) (49).

The pond sludge in the tanks contained several metals, organic chemicals and radioactive constituents (gross alpha and beta) that were present at levels that required it to be managed as a mixed waste. In accordance with EPA published guidance (50) waste containing low level radioactivity regulated under AEA and hazardous waste regulated under Subtitle C of RCRA are subject to both sets of statutory requirements (4). The strategy of the proposed process was to remove all hazardous constituents necessary to delist the waste and allow for disposal outside the jurisdiction of RCRA. Standards in 40 CFR 260.20 outline the procedures for delisting waste.

Transportation Considerations

Following the treatment of the pond sludge and the stabilization of the extracted radioactive metals, all the waste would be packaged and transported for disposal. All packaging, manifesting and transporting of the waste would be accomplished in accordance with the NRC requirements set forth in 10 CFR 71 and the Department of Transportation (DOT) requirements in 49 CFR 173. In order to determine the proper packaging requirements for radioactive material, the radionuclides in the material and the activity of the material would have to be identified.

Disposal Considerations

The waste streams created from the proposed process that would require disposal were the stabilized mixed waste, the treated pond sludge and the process wastewater. The stabilized mixed waste could potentially be

disposed of at Envirocare Inc. of Utah. The radionuclides and the listed waste code would have to be known prior to arranging for final disposal. The treated pond sludge would be considered a Class A LLW based on the information provided by WERC. The sludge could be disposed of at a licensed commercial LLW disposal site or at a DOE disposal facility. The wastewater resulting from dewatering during the treatment processes would be disposed of through a local POTW in accordance with the NPDES permit for the POTW and with the requirements set forth in 40 CFR 413 and 10 CFR 20. This wastewater would be appropriately analyzed for radioactive, hazardous and other characteristics to ensure this compliance.

HEALTH AND SAFETY CONSIDERATIONS

Several health and safety issues would have to be considered if the proposed process were implemented at the actual site. Safety training would be provided to all employees involved in remediation activities in the following areas: Right-To-Know Regulations (SARA, Title III), Radiation Safety, Emergency Response, Hazardous Waste Management (RCRA), and the use of Materials Safety Data Sheets (MSDS). OSHA and NRC regulations will be followed for all on-site operations.

Awareness seminars would be held to inform the community about the health and safety measures being utilized. An emergency evacuation plan would be in place for the site and its surrounding areas. Emergency response training would be made available not only to police, medical, and fire personnel, but also to community response teams so they may properly respond to hazardous waste and chemical emergencies. A qualified site safety officer would be available during remediation activities.

COMMUNITY RELATIONS PLAN

To establish a strong relationship with the community, a Community Relations Committee (CRC) would be formed, consisting of community leaders and representatives and environmental professionals. The primary role of the CRC would be to hear and evaluate public concerns relating to the remediation project. The following issues would be emphasized throughout the duration of the project:

Community participation is valued.

The proposed process significantly reduces taxpayer cost by minimizing the quantity of mixed hazardous waste that needs to be disposed of in a secured facility.

A toll free hotline would be provided to answer questions and concerns raised by the public. There would be separate toll-free numbers for Spanish speaking individuals and a TTY phone system for the deaf.

A well developed public relations plan would ease the worries of the community and make the project a welcome one.

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CLEMSON UNIVERSITY'S VITRIFICATION RESEARCH LABORATORY: A DOE/INDUSTRIAL/UNIVERSITY JOINT EFFORT

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ABSTRACT

Clemson University's Vitrification Research Laboratory has been serving the needs of government, industry and education since its creation 3 years ago this spring by assisting in vitrification processing trials of surrogate DOE waste streams in commercial equipment. The use of surrogates allow runs to be conducted at very low cost compared to processing radioactive wastes, and permit trials that push processing and compositional limits. A waste composition that devitrifies into refractory rock at the bottom of the melter, or aggressively consumes the melter's refractory lining, is a relatively minor inconvenience in the Vitrification Lab. In a radioactive waste processing facility however, such occurrences would be very serious in terms of lost time, incurred expense and the creation of additional waste.

Waste treatment equipment suppliers can use our laboratory as a test bed for their equipment, thus gaining experience and knowledge regarding performance and improvement opportunities. Users of the technology, such as waste processors and those with waste streams, have the opportunity to tap an objective source of information obtained from Clemson University's past experience. Or, previously untested waste streams can be treated

using the Vitrification Lab's resources to evaluate the effectiveness of several treatment technologies.

Clemson University and its students benefit from the training and experience afforded by the studies conducted at the Vitrification Lab. Potential employers are identified for the students, the university derives support for research from the work conducted, and students, professors and the Vitrification Lab staff have the opportunity to publish the results of their work. Additionally, equipment manufacturers, waste processors and owners of waste streams that have a need for well educated engineers with practical experience on real-world equipment and processes will find the Vitrification Lab's undergraduate and graduate students well suited to their needs.

This presentation will describe the laboratory and its equipment, the surrogate waste streams tested, the results achieved and lessons learned. Work currently underway will be described, as will plans for future expansion and diversification.

INTRODUCTION

In 1992, Clemson University, DOE/Westinghouse Savannah River Company, Envitco, Inc. and Stir Melter, Inc. cooperated to form a vitrification research laboratory within the Department of Environmental Systems Engineering at Clemson University. Since then the lab has conducted a number a vitrification process development campaigns in support of DOE's mixed waste management programs. Studies are also conducted for the industrial partners to support their efforts.

The laboratory currently houses three melters, including a DC arc unit from Electro-Pyrolysis, Inc., and employs six full time and two part time staff. A number of graduate and undergraduate students are also involved in thesis projects or work/study programs.

CAPABILITIES

The current laboratory facility, shown in Fig. 1, was constructed in the winter of 1992-93 directly behind the L. E. Rich Environmental Research Laboratory at Clemson University. The vitrification laboratory building is a pre-engineered metal structure, 12.2m x 15.2m x 4.9m high at the eaves, insulated, heated and air conditioned. Three melters (1) are currently housed in the lab. Of these, the two joule-heated melters are owned by the Clemson Universities Research Foundation and the DC arc graphite electrode melter is owned by DOE/WSRC. The existing lab space is fully utilized by the three melter units and their auxiliary power, feed, offgas and data acquisition systems. In fact, with the addition of the DC arc melter in March 1995, warehouse space had to be leased for storage of materials and supplies previously housed in the lab. Clemson University plans to expand the Vitrification Lab in the spring of 1996, as discussed later in this paper.

Fig. 1

Processing

Envitco EV-16 Joule-Heated Melter - The melter was first operated in our laboratory in January, 1993 and has produced over 3175 kg of glass from surrogate wastes since then. A schematic is shown in Fig. 2.

Fig. 2

The Envitco melter has an 0.46m x 0.46m x 0.36m high (0.076m3) refractory lined melting chamber and is capable of continuously processing glass at rates exceeding one-half ton per day . Glass temperatures in excess of 1500C have been measured in the vessel, while the glass exiting the melter is typically in the range of 1300-1400C. This melter has a 100 kW

power supply and uses molybdenum electrodes. It is capable of cold top or hot top operation and has been operated with both dry or slurry feeds. In cold top operation with dry feed, the surface temperature of the feed layer is of the order of 200 C. This cooler zone of feed material decreases metal emissions by condensing and refluxing volatile metals into the melt.

Because of its high temperature capability, this melter allows a wide compositional range for a given waste material. The melter has achieved continuous melt rates of 30 kg/h on dry feed.

The Envitco melter is equipped with both dry and slurry type feeders. The dry feed system is equipped with a scale and feedback control circuit to ensure a constant feed rate. The slurry feed system utilizes a continuous loop to ensure that feed material is kept in suspension in the slurry. At the melter, a calibrated pump feeds the slurry into the melter.

WV-0.25 Stir-Melter - Clemson University's WV-0.25 Stir Melter is one of three stirred melters in operation. It has an Inconel 690 melt chamber with a 0.15m x 0.15m x 0.30m high working volume and an Inconel 690 stirring electrode. A schematic of the melter is shown in Fig. 3. Maximum processing temperature is limited to ~1070C because of the materials of construction. The melt chamber has an internal "tea pot" spout with an external drain tube, allowing for continuous operation. The melter has separate ports for dry feeding and slurry feeding of the batch and other additives. Ten kW of power is available for Joule heating through the stirrer/chamber circuit. In addition, there are external resistance heaters with a 7 kW power supply for startup and to maintain temperatures during tests.

Fig. 3

The stirrer is driven by a 325 W variable speed motor at speeds up to 1000 rpm. The impeller can be raised or lowered to optimize mixing and melting rates. There is a port for either a video camera or direct viewing. There is an additional port for the insertion of a thermocouple or other probe into the melt. The offgases flow into a sampling and treatment system; sampling is done downstream of the melter exit at the center of a 50.8 mm Schedule 40 pipe.

Offgas Treatment and Sampling - Each Joule-heated melter has its own offgas treatment system. For the Envitco melter, the offgas system consists of a quench chamber, steam atomizing scrubber, and packed tower for acid gas removal and neutralization. For the Stir-Melter, the offgas system is a spray quench/ejector venturi, jet sparging scrubber, and a counterflow packed tower for acid gas removal and neutralization. Each system has its own positive displacement blower.

The lab has complete sampling trains for the standard EPA Method 5 (particulate matter), Modified Method 5 (semivolatile organic compounds), Method 29 (multiple metals), and VOST (volatile organic compounds). In addition, size fractionated samples can be taken using a Pilat Mark 3 source test cascade impactor.

EPI DC Arc Melter - The EPI melter has a 0.33 m diameter x 0.53 m high (0.045 m3) graphite crucible contained inside a sealed, water-cooled, atmosphere-controlled vessel. A simplified schematic of the melter is shown in Fig. 4. The melter is capable of processing ~50 kg of material per charge and can be operated in either the cold- or hot-top mode. Power is supplied from a 100 kW DC supply through consumable graphite electrodes. The vessel cover contains three ports, located 120o apart, that can be used for addition of feed materials, thereby increasing the

melt capacity, or installation of a monitoring device such as a camera or pyrometer. While the current melter design allows only batch-type operation, it could be modified for continuous operation.

Fig. 4

Instrumentation - Through our research programs with WSRC, we have an automated data collection system for continuous sampling and recording major operating variables. For the Envitco melter, these include:

- electrode voltages and currents,
- melt chamber temperature,
- melter headspace (plenum) temperature,
- offgas flow rate and temperature,
- spray quench/ejector liquid flows and temperatures.

For the Stir Melter, these include:

- Joule and external heater voltages and currents,
- stirrer speed,
- stirrer motor current,
- melter chamber external temperature,
- drain tube temperature,
- melter headspace (plenum) temperature,
- offgas flow rate and temperature,
- spray quench/ejector venturi and packed column liquid flows and temperatures.

Deltech Melter - There is a separate laboratory in the L. G. Rich building for conducting crucible tests and glass analyses. This laboratory has a Deltech glass melting furnace model DT-31-12-RS. This is a bottom loading furnace with a temperature capability above 1700C, an optional stirring mechanism, and an element protection liner. In addition, there are a number of lower temperature furnaces for preheating, annealing, and devitrification studies.

Technical

In addition to the physical capabilities described above, the Vit Lab also has access to a number of skilled and experienced individuals. Thomas J. Overcamp, the principal investigator of the Vit Lab project, is a professor in the Environmental Systems Engineering department. Professor Overcamp's specialty in air pollution control has been very valuable to the vitrification project. The Environmental Systems Engineering department also has eight other faculty members involved in all aspects of environmental protection, remediation and compliance. The Vit Lab is staffed by seven full and part-time personnel. These include two experienced melter operators/technicians and two part-time administrative staff to handle and monitor purchasing, budgets, quality and other support issues. The remaining technical and management staff are responsible for designing, conducting and overseeing projects.

Education

The Vit Lab has played host to a number of graduate and undergraduate students, providing practical experience, research for graduate thesis topics, and financial support. Since beginning operation, thirteen Environmental Engineering graduate students and one Ceramics Engineering graduate student have been or are currently involved in vitrification-related projects. Nine of these students have left Clemson and two are now employed in the waste vitrification area.

EXPERIENCE

Compositional Studies

Durability studies of glass compositions that could be made from waste water treatment sludges located at Savannah River Site (SRS), Oak Ridge Reservation (ORR), Rocky Flats Plant (RFP), and Los Alamos National Laboratory (LANL) were conducted to determine whether they were candidates for vitrification. The range of the major oxide constituents estimated to be in the DOE site sludges are shown in Table I. By considering two levels of the major oxides, a space of 32 possible glass compositions was defined. To determine the ability of the glasses to effectively isolate hazardous metals, fixed-level oxide additions of 3 hazardous metals were made. Leaching studies found that at the higher ratio of glass formers (SiO_2, Al_2O_3 and B_2O_3) to glass modifiers (Na_2O, CaO, BaO, NiO, FeO and PbO), the most durable glasses were made, as measured by the sodium normalized release rate (NaNRR) and the 7-day Product Characteristic Test (PCT). It was also found in compositions with low former:modifier ratios that the relative quantities of soda and lime (two modifiers) affected the NaNRR. High Na_2O:CaO ratios further decreased the glass durability. At high former:modifier ratios the soda:lime ratio had very little effect. These studies provide a starting point in defining the additions that need to be made to various waste streams so that a durable glass will be formulated.

Table I

Surrogate Waste Processing Studies

Clemson University has conducted 18 major surrogate demonstration studies on vitrification of 1) five different wastewater treatment sludges from DOE facilities; 2) ion exchange resins in SRS simulated high-level waste slurry; and 3) high-sodium, low-level waste from Hanford Reservation. These tests have included production of sufficient glass to achieve steady-state operating conditions, durability testing of the waste glass, and measurement of the offgas emissions. A summary of these campaigns is given in Table II and discussed briefly below.

Table II

Results

Table III presents the compositions of the surrogate waste streams tested. The test results are summarized below.

Table III

M-Area - Borax, or sodiumboratedecahydrate, additions were made to provide borosilicate glass compositions with waste loadings of 70, 80, 90, and 95% for runs conducted in the Envitco melter, and 80 and 85% for runs in the Stir-Melter. These glasses processed well in both the Envitco and the Stir-Melter (2,3) and exceeded the requirements for TCLP and PCT. In general, it was found that the higher the waste loading, the more durable the glass.

WETF - A calcium aluminosilicate glass with 45% waste loading was made by adding perlite to the sludge surrogate for processing in the Envitco melter. Because of the Stir-Melter's limited temperature capability, two lower melting point glasses were selected for it. A borosilicate glass with 45% waste loading was prepared by adding 28% borax and 27% diatomaceous earth, and a soda-lime-silicate glass with ~35% waste loading was made by adding 42.3% precipitated silica, 13.0% lithium carbonate, and 9.4% sodium carbonate.

The glass compositions chosen for the Envitco melter were very difficult to process. Because of high viscosity and melting point, draining at the same rate as the melter was fed material was not possible due to glass build-up inside the drain tube. This composition was also found to

devitrify and form wollastonite (melting temperature >1800oC) during overnight and weekend hot holds. Despite these problems, nearly three melter volumes were made during the run.

The glasses chosen for the Stir-Melter were easier to process, exhibiting occasional blockages of the drain tube while producing over 5 melter volumes.

All the glasses made during this study exceeded the minimum values required for TCLP and PCT. In general however, the higher melting temperature calcium aluminosilicate glass was the most durable of the 3 compositions (2-4).

Rocky Flats Plant - A borosilicate glass with 50 % waste loading was made by adding 14 % Borax, 36 % Diatomaceous Earth, and 3 % aluminum oxide. The aluminum oxide was added to increase the viscosity of the glass to a level suitable for draining from the melter. In addition to the glass forming additives, activated carbon at 3 % was chosen as a feed additive to deter the formation of sulfate salts. The RFP sludge was fed to the melter as a slurry with 55 % solids content. This glass composition had a melting temperature of ~1150oC.

Processing of the RFP waste was very difficult. Offgasing and foaming, resulting from the high nitrate and high sulfate content of the waste, prevented continuous draining and feeding of the melter. Stable sulfate salts formed on the surface of the melt and hindered feed incorporation into the molten glass. The sulfate salts also caused blocking of the offgas treatment system, halting processing until the system could be cleared.

Despite devitrification tendencies due to high calcium content, the RFP waste glass passed TCLP and PCT requirements. Based on the Fe^{2+}/Fe^{3+} ratio, the waste glass produced was in a highly reduced state.

Volatilization of the plutonium surrogate and RCRA metals was low. Melter wear, especially for the electrodes, was high.

LANL Sludge - LANL TA-50 precipitate sludge contains precipitating agents and filter aids. A calcium aluminum silicate glass was made from this surrogate waste. Due to its high melting temperatures, there were some problems with devitrification in the melter. The product glass was very durable passing both the TCLP and PCT requirements.

High-Sodium Hanford - This is a high sodium, nitrate containing, low-level waste from the high-level waste vitrification process. A surrogate slurry was tested in the Envitco melter. The glass chosen was sodium aluminum-calcium silicate. Test using both slurry feed and dry feed after drying the sludge and glass forming materials was tested. The glasses produced were very durable and the offgas emissions of simulated radionuclides and heavy metals were very low.

Oak Ridge K-25 Plant B&C pond sludge - This waste is from a settling pond used for holding wastewater sludges from the K-25 plant. An alkali-lime-silicate glass with 50% waste loading was tested in the Envitco melter. Due to the low melting temperature, minimal processing problems were experienced with this waste. Even after extended hot hold periods in the melter, the glass did not devitrify. The durability tests have not been completed.

THE FUTURE

FY96 Planned Activities

Three major efforts in support of WSRC will be conducted at the Vitrification Laboratory during FY96. These are:

Support of the TVS - The TVS, or Transportable Vitrification System, is a 5 ton per day mixed waste vitrification facility built for DOE/WSRC by Envitco, Inc. The TVS was moved from Dreicor, Inc. in Irwin, TN on 15 flat bed trucks, was assembled and is temporarily sited on Clemson University property. The Vit Lab is supporting final modifications and shakedown surrogate runs of this facility by providing engineering and technical support, conducting surrogate trials in the EV-16 melter, and preparing the surrogate blends to allow processing 15 ton trial runs of glass through the TVS. A K-25, B and C pond sludge composition from Oak Ridge Reservation was successfully processed in the EV-16 melter during December of 1995, and CNF surrogate runs (in the EV-16 and in the TVS) are scheduled for early in 1996.

Landfill Stabilization - Processing trials studying the effects of various fluxing additions will be conducted in the DC arc graphite electrode melter.

Glove Box Offgas System - This project involves designing, building and testing a small off gas system intended to remove organics and Pu from mixed waste. The system will be installed on a small graphite electrode melter being build by the Idaho National Engineering Laboratory for WSRC. Clemson will test its offgas system design using the DC arc graphite electrode melter located in the Vit Lab.

Facility Expansion

A design, a construction financing plan and approvals to build have been obtained to expand the current vitrification laboratory by an additional 280 square meters. The enlarged laboratory will be enhanced with brick veneer, windows and landscaping and will accommodate new and larger melter systems or other waste processing equipment (6.1 m eave height). The expansion will also provide on-site storage space for materials and supplies, office space for the laboratory technicians and a designated feed preparation and mixing area.

Processing Diversification

The Vit Lab is currently negotiating with two equipment suppliers in response to their interest in establishing non-vitrification waste processing capability at Clemson University's Environmental Systems Engineering Department. The interest shown by these companies results from Clemson University's experience in waste processing, the existence of a dedicated pilot facility, and our ties with DOE through the Savannah River Site. Clemson University is interested in discussing working relationships with other equipment or systems manufacturers and waste treatment providers.

SUMMARY

Over the past 3 years, Clemson University's Vitrification Research Laboratory has been successful in 1) supporting DOE/WSRC in developing processing conditions for treatment of low-level mixed waste, 2) providing equipment manufacturers with the opportunity to better understand the capabilities and limitations of their equipment, and 3) providing educational and financial opportunities to students. As knowledge about the vitrification process is gained, attention turns to the treatment of actual wastes. either by DOE facilities like the TVS, or by commercial waste processors. Maturation and increased understanding of vitrification technology will deemphasize the need for the kind of work being done at the Vitrification Laboratory, requiring that the Vit Lab diversity and broaden its waste processing development activities. Discussions are now underway with several process developers, evaluators

and users to provide the desired broader base. If successful, this technologically broader laboratory will continue to provide valuable services to Government, Industry and Education.

ACKNOWLEDGEMENTS

The work discussed in this paper was conducted under contract No. DE-AC09-89SR18035 under the South Carolina Universities Research and Education Foundation Task Orders 95, 121 and 183 for the Department of Energy and the Westinghouse Savannah River Company's Savannah River Technical Center. The continued support and encouragement provided by WSRC is gratefully acknowledged.

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Co-chairs: Paul Macbeth, Dames & Moore

23-1

COLLIMATED IN-SITU GAMMA SPECTROMETRY: A NEW METHOD FOR FAST CLEARANCE MEASUREMENTS OF LARGE AREAS OR BUILDING STRUCTURES OF NUCLEAR FACILITIES UNDER DECOMMISSIONING*

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ABSTRACT

The requirements on a collimated in situ gamma spectrometer for the use in nuclear facilities under decommissioning is described. A prototype was developed and constructed. The equipment was used in seven facilities in Germany and France to check mass- or surface-specific activities on outdoor grounds and inside the buildings in restricted areas.
The results gained by in-situ gamma spectrometry are compared with "traditional" methods like sampling or measuring with contamination monitors. The advantages of this new technique are worked out. It could be shown that such a device is able to meet in the most cases the essential detection limits regarding national radiation protection ordinances or release criterias.

INTRODUCTION

The ideas in the late sixties which led to the use of bare spectrometric radiation detectors like NaI(Tl), Ge(Li) or nowadays high-purity germanium detectors (HPGe) in the field (in-situ gamma spectrometry) were to get easy and rapid information about the radiological state of outdoor grounds after nuclear weapon tests or to estimate dose rates created by natural radioactive nuclides in the soil.
In these cases, it was assumed, that there is no disturbance of the source-detector geometry for many hundred square meters around the detector. After the nuclear accident in Chernobyl in 1986 these

advantages focused strong scientific interest on the in-situ technique and in 1993 it was established in the German regulatory for emission surveillance after significant radioactive emissions (1).

MOTIVATION

In a late phase of the decommissioning of a nuclear power plant all components containing a significant inventory of radioactivity are removed, leaving large surfaces with often poorly known contamination levels. Especially for large areas e.g. outside the buildings "in situ gamma spectrometry may be the only method of achieving validation of the release criteria" (2). Taking in account only buildings of restricted areas in the next 50 years in Germany 7E3 Mg activated and 5E6 Mg contaminated concrete must be released from facilities under decommissioning.

Before the area inside the perimeter of the facility, including all concrete structures, can be treated as a building which can conventionally pulled down or used conventionally the remaining radioactivity must be determined in order to check the radiological relevance of the concerned part of the plant and to decide the possible path of material release. The basis of assessment maybe the so-called "10mSv-concept" (3).

COMPARISON WITH OTHER MEASUREMENT STRATEGIES

Today the mainly used strategies are analyzing samples, taken from the surface or measuring the surface activity via large proportional counters. In these cases we meet severe systematic problems:

How reliable are the results gained by taking samples and performing laboratory analysis (statistical significance of a number of samples)?

Drawing and analyzing samples leads for the single sample to more exact results than in-situ measurements do. The reasons are that in the latter case the source geometry is not known completely and the measuring times are normally much shorter because of the large number of measurements.

The main problem of the analysis of samples in the laboratory is the estimation of the representativity of a collection of samples. Comparing the results of in-situ measurements and laboratory analysis we noticed substantial differences which could be attributed to large inhomogeneities in the spatial distribution of activity. These were not resolved by sampling.

The importance of this problem can also be shown theoretically. In the publication of Ferguson (4) the number of samples are calculated to meet a defined statistical significance to find an existing contamination by sampling. As an example ~200 samples are necessary to find a circular contamination with a diameter of 35 cm on a relatively small area of 10 m2 with a safety of 95%.

Does a stable nuclide vector exist, in order to get reliable calibration factors between the count rate of a proportional counter and the surface activity? Are possible contaminations really surfacial or was e.g. a wall painted to fix old contaminations, so that quantitative values cannot be achieved by measuring the short ranged a- or b-particles?

A commercial contamination monitor meets a minimum detectable activity of 0,1 Bq/cm2 Co60 or Cs137 on a surface two- to ten-times faster than a collimated in-situ spectrometer. On the other hand shielding layers coming from decontamination coatings or migration of the radionuclides into the surface lead to relative small errors performing in-situ measurements whereas contamination monitors are not appropriate in the most cases to determine the remaining activity.

The actually discussed clearance levels for the unrestricted release of contaminated concrete or soils are so low, that even strong gamma emitters cannot be detected by measurements which are not nuclide specific.

Even a large surface contamination with 0,1 Bq/cm2 Co60 would create an additional dose rate smaller than 2 nGy/h. The detection of this contamination by measuring dose rates is impossible. In comparison an uncollimated in-situ measurement in a room will identify dose rates from not natural nuclides smaller than 0,06 nGy/h (!) in one hour measuring time.

The so-called "Freimeanlagen" used for clearance measurements of dismantled components or waste containing large plastic scintillation detectors in shielded boxes are normally not able to detect a few 10 Bq/kg of Co60 or Cs137 due to the variation of the natural background.

Errors arising from this sort of problems do not appear by measuring with a collimated high-purity germanium detector directly in the regions of interest ("in-situ"). There are only two problems left:

The vertical distribution of a contamination may be not known well enough.

This fact creates a uncertainty which can reach a factor of two. On the other hand this systematic error is solvable and furthermore small, compared to errors which can affect the quality of the results of traditional techniques.

A formal problem e.g. in the German radiation protection ordinance is, that the value of a surface contamination has to be achieved by a measurement averaging over 100 cm2.

It is obvious, that such an averaging area is not adequate for release measurements over thousands of square meters of probably contaminated walls, floors, roofs or soils. At the moment this subject is being discussed in Germany.

Apart from this the collimated in-situ gamma spectrometry allows to get very fast information about the radiological state of large areas and -if one takes in account the high quality of these information- for low costs.

DEVELOPMENT OF A COLLIMATED IN-SITU SPECTROMETER

It was obvious that using such a device for clearance measurements it had to be collimated to reduce the field of view. Reasons therefore are that

- possible contaminations must be locatable,
- there must be a well defined averaging area (e.g. some m), and
- contaminations outside the measured area must be effectively suppressed.

Selection of the Detector Type and Shape

The strong worsening effect to the minimal detection limits using a collimator required an efficient detector system. Because of the relative poor energy resolution of anorganic scintillation detectors they reach the same detection limit even for nuclides which dominate the spectrum in a measuring time about ten times higher than a germanium detector of "normal" size. Very small clearance levels or more difficult to measure nuclides in realistic nuclide vectors can only be detected with germanium detectors.

So we designed and constructed a prototype of a high-resolution in-situ spectrometer on the basis of 41%p-type HPGe-detector, surrounded with an optimized collimator made out of brass and a low natural activity lead-bismuth-tin-alloy.

The key aspect of the detector parameters is the detection efficiency, because it has a direct effect on the required measurement time and there is no provision for measurements in areas which exhibit a degree of contamination which would lead to dead-time problems (> 500 Bq/cm2). The use of a n-type detector would enable measurements of nuclides with low photon energies. As these detectors are more expensive and the measurement of low photon energies in old contaminations, and thus of the nuclides which have migrated into the matrix, is more difficult and leads to larger errors, we dispensed with this option and focused on achieving high detection efficiencies within a defined budget. An estimate of the measurement time and thus of personnel costs, on the one hand, and the higher acquisition costs for a more sensitive detector, on the other hand, shows that the use of a more sensitive detector is amortised after a measurement time of just a few weeks.

Fig. 1

The ideal form for the geometry of the detector crystal would be a relatively flat disc (L/D < 1, ratio of the length L and diameter D of the crystal), due to the fact that a preferential direction of sensitivity towards the front (or back) would thus be achieved in principle. However, extremely large disc shapes for use in detectors of the size described can only be obtained with difficulty because they would have to be manufactured from a very large germanium blank. For this reason, a virtually direction-independent detector (L/D 1) was acquired which also greatly simplifies the assessment of spectra recorded without collimation.

Detector System

In order to guarantee effective shielding against background radiation from the rear hemisphere, the detector end cap was set back 7 cm from the preamplifier housing in order to make room for a rear shield. With a capacity of 7.5 liters, the largest commercially available nitrogen tank for portable detectors was used. The holding time of the cooled system thus amounts to at least 4 days. The tank is designed for use in all spatial directions. The possible savings on weight via a smaller nitrogen tank are negligible in view of the collimator which weighs over 40 kg. The cable to the electronic measuring equipment was provided with a splash-proof Lemosa plug-and-socket connector.

Collimator

The materials used should have a sufficiently low natural radioactivity. The wall thicknesses must also guarantee sufficient lateral shielding even in the presence of high-energy radiation (e.g. Co60). This aspect is of particular importance, especially in the case of surface contaminations, due to the fact that contamination spread over a large area results in very large contributions from the large angles of incidence in relation to the surface normals. In this case, approx. 90% of the photons reach the collimator-detector system at an angle greater than 45. A minimum shield density of 50 to 60 g/cm2 for an angle of incidence parallel to the investigated surface can be seen as a compromise between portability and shielding effect.

The importance of material selection in the manufacture of the collimator is frequently overestimated. A collimator designed for Co 60 radiation could already be considered oversized for the radiation of the daughter of Cs 137 due to the energy-dependent attenuation. Thus, only high-energy radiation is of importance in further considerations. As a function of the wall thickness in g/cm, the attenuation is virtually independent of

the atomic number due to the fact that Compton scattering is nearly the only effect which occurs as interaction process at the relevant energies. Calculations show that the difference in the shielding effect between a steel and a tungsten collimator at the same mass per unit length amounts to only about 10%. However, this is compounded by additional effects stemming from the rotational symmetry of the problem; a linear increase in the density results in a more than indirectly proportional reduction in the collimator volume. This fact leads to a situation where our collimator has a maximum 15% poorer shielding-mass ratio than a collimator made of a tungsten alloy (e.g. Triamet). However, the costs of a tungsten collimator are several times higher than those of a "conventional" collimator.

An attenuation mass per area of ~53 g/cm2 was realized for radiation which reaches the collimator in an angle of incidence parallel to the investigated surface. Furthermore the detector is shielded for radiation from the rear hemisphere. Depending on the collimator used and the height of the detector above the surface our spectrometer averages over areas between 0.4 and more than 10 m2 in one measurement.Electronic equipment and Software

The electronic equipment consists of a battery-operated system with high voltage, the main amplifier and the ADC, and a Notebook for control and data storage. The electronic data are comparable with those of a laboratory measuring station. The electronic equipment weighs less than 6 kg and can be run in the battery mode for two to three hours. Any desired operating time is possible as an option with auxiliary batteries (2*12 V/16 Ah) or a mains supply (220 V).

Programs for automatic measuring sequences and for quality assurance are used (Genie PC / OS/2) in addition to conventional visualization and analysis software. The software is required for measurement cycles of up to 250 measurements per day. This means that the controlling of the spectrometer and the complex analysis algorithms run automatically in the background and the most important system parameters are simultaneously monitored during practical use. Standard graphics software (Stanford Graphics / Windows) is used for the visualization of the results of total-surface grid measurements in the form of a plot.

Measuring Apparatus

The measuring apparatus must be suitable for separately housing the relatively sensitive detector and the heavy collimator under exactly reproducible conditions. This also guarantees the maintenance of the calibration geometry during measurement. An apparatus for measuring vertical surfaces was developed for use inside buildings and as a transport device. A separate measuring setup was also to be developed for soils in relatively impassable outdoor areas. The two devices were designed in such a way that the same calibrations can be used.

The two devices can be set up and dismantled in a very short time using only a few tools. The set-up times, excluding the transport of the components to the measurement site, amount to roughly half an hour.

CALIBRATION

Collimated in-situ gamma spectrometry means measuring large volume sources of different chemical composition with any spatial distribution of the radionuclides. Experimental calibration proves to be difficult due to the fact that - understandably - calibration standards cannot be reproduced for all conceivable nuclide distributions which occur in in-situ gamma spectrometry. This situation is compounded by the fact that

these kinds of standards would also have to be very large and heavy. There are three possible solutions to this problem.

Experimental Calibration

Calibration curves can be calculated by recording a great number of individual spectra with point sources or small volume sources at different angles and depths and subsequently superimposing them as a function of the source geometry. This procedure can lead to errors, e.g. in the case of pure surface contaminations.

Numerical Method

It is also possible to use a purely numerical method for calibration by modelling the radiation source as well as the entire detector system in suitable program codes (e.g. Monte Carlo simulations). However, data obtained in this manner must be experimentally supported.

Standard Method

The calibration method suggested by Beck et al. 5 in 1972 is still the state-of-the-art today for the calibration of uncollimated systems. Given a correspondingly detailed analysis of the angular dependence term, this method can also be used to a limited degree for collimated systems. It combines experimental detector-specific data with numerical calculations. With the help of the so-called standard method and the corresponding published table values, an initial calibration can be determined relatively quickly and simply for uncollimated systems. However, the influence of the extreme angular dependence of a collimated spectrometer leads to extensive computational work for determining the calibration factors. The calibration factors used at this time are based on the standard method and were verified by purely experimental methods. By integrating the numerically determined calibration factors at a later date, a consistency test of the calibration can subsequently be conducted based on three completely independent methods. The following table summarizes several calibration factors for selected cases:

Table I

Several yield curves with calibration factors based on the standard method have already been verified experimentally. Further tests of the consistency of the various calibration methods are currently being conducted. According to our current experience, the two yield curves generally do not deviate from one another by more than 15%. Differences of more than 25% have not occurred up to this point. These tests make the individual calibrations very reliable, as the two methods are in no way related to one another and the occurrence of identical systematic errors is thus ruled out.

APPLICATION OF THE SPECTROMETER

Aside from the development of such a spectrometer we used our prototype inside the restricted area and outside the buildings of several nuclear power plants under decommissioning. Therefore we normally laid a grid of measuring points over the area, so that parts of the circular fields of view would overlap.

In the following nuclear facilities under decommissioning we used our spectrometer:

Outdoor grounds

(grassland and tracks): KKN (pressured tube reactor, D2O moderated CO2-cooled, decommissioned / Germany)

VAK (BWR / Germany)

NUKEM-A (fuel fabrication facility /

Germany)
WAK (fuel reprocessing plant / Germany)

Surfaces in restricted areas: KRB-A (BWR / Germany)
G3 (NUGG / France)
RAPSODIE (FBR / France)
WAK

Up to now with this campaigns new knowledge about the lateral distribution of contaminations, the variability of its composition or -by comparing with other measuring techniques- their systematic uncertainties could be quantified. We selected three examples:

Mapping of the short distance variability of the Chernobyl Cs137 fall-out by scanning 1500 m2 grassland completely with 3 m2 field of view in a single measurement

Fig. 2

With a personal expense of ~70 man-hours the above sketched area was measured completely. To take a stock of the radiological state of a large area in a such detailed matter can only be performed with this procedure. With the basis of such investigations specific rehabilitations can be made.

Investigation of the deviation in the measured surface activity via proportional counter and in-situ spectrometry for several surfaces with different histories and contamination scenarios.

At the moment four extensive floors were measured simultaneously with both methods. In all cases the in-situ spectrometry led to higher values because it "sees" also the deeper distributed activity. Depending on the history of the area the factor between the values can differ extremely between ~1,3 for airborne contamination on not decontaminated areas over factor 3-5 for decontaminated concrete surfaces to >30 for contaminations under coating (see below).

Investigation of a high contaminated, coated floor

Fig. 3

This example shows the advantage of in-situ measurements on three features:

1) The nuclide vector in this room is completely indefinite (logarithmic vertical scale !). Both the ratio and the extent of the contamination varies with orders of magnitudes. Taking samples makes no sense under this circumstances.

2) Due to the coating the measurements with contamination monitors lead to wrong information about localization and extent of contaminations.

3) Furthermore the variability of the nuclide vector makes a definition of calibration factors for contamination monitors impossible.

PERFORMANCE OF THE SPECTROMETER

The detection limit of a measuring device serves the assessment if this device is suitable for the required purpose. Here the values which can be reached depend strongly on the specific conditions (incl. the collimator used). The values in the table below were calculated from measurements in building free of contaminations and with a normal dose rate. The collimation was moderate, a measuring time of ten minutes and a horizontal homogeneous source distribution was assumed. An error probability of 10% was allowed. The basis of calculation is (6) and every value was calculated from more than three single measurements.

Table II

These values can rise on outdoor grounds and maximum collimation by a factor 3. But even then the actually discussed surface- and mass-specific clearance levels for the main nuclides Co60 and Cs137 can be achieved without problem.
Furthermore disturbing radiation from outside the interesting field of view is strongly suppressed. For low and medium energy gamma radiation (<700 keV) from nuclides distributed vertically and horizontally homogeneous in soil 90% of the total photon flux appear from an angle of 48 relative to the symmetric axis of the detector in the collimated case. Without a collimator this value would rise to 82, so it would be impossible to localize a source.
Fig. 4
Practical examples for minimal detectable activities are:
Under rigid conditions (maximum collimation, some migration of the nuclides in concrete) and normal dose rates due to natural nuclides detection limits of 800 Bq/m2 for Co60 and 1400 Bq/m2 for Cs137 in three minutes are reached.

A contamination with a nuclide vector containing only 20-30% of these nuclides can be detected according to the requirements of the German radiation protection ordinance.
For the relatively difficult to measure activity of U238 (via Pa234m) a detection limit of 150 Bq/kg is reached after 15 minutes.

Uranium can be measured on storage pits with any enrichment (f.e. (7)).

The observance of exemption values for deposition of waste contaminated with the more difficult to measure nuclides of the nuclear fuel cycle can be proved (f.e. (8)).
In the same time a surface contamination of 500 Bq/m2 U235 is detectable and can be discriminated from the natural Ra226 and U235 186 keV background in concrete containing 1 Bq/kg U235.
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NEWLY DEVELOPED DECONTAMINATION TECHNOLOGY FOR NON-INCINERABLE RADIOACTIVE WASTE BASED ON GASEOUS REACTION OF CARBONYLATION AND FLUORINATION

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ABSTRACT

New gas-phase decontamination technology based on gaseous reactions utilizing volatile properties of carbonyl compounds of radioactive transition elements and fluoric compounds of actinides was developed. To determine the feasibility of this new technology, reacted under high CO pressure (50-200atm) with heating (about 350C), non-radioactive (cobalt, chromium, nickel, rhenium, molybdenum, manganese, ruthenium, zinc) and radioactive nuclides(cobalt-60, nickel-63, ruthenium-103) transition elements were removed as gaseous forms. Experiments with uranium using fluoric gases were also done. Forms of these volatile compounds were predicted to be carbonyl compounds and fluoric compounds. In the case of radioactive nuclides existed in the hard oxide layer of stainless steel, by utilizing pretreatment with supercritical CO2+I2+H2O, the hard oxide layer was removed completely and the gaseous reaction was promoted; cobalt-60 was mostly removed by 200atm CO gas, and cobalt-60, ruthenium-103 and uranium were removed 30-35%, 49-57% and 55-60% respectively by 200atm CO+COF2 gases. From these experimental results using non-radioactive and radioactive nuclides, the feasibility of this new idea was determined. This decontamination technology based on gaseous reactions has the ability and possibility to decrease drastically the large volumes of non-incinerable radioactive wastes, which include ash, many kinds of used-metals, and equipment.

INTRODUCTION

Decontamination and volume reduction of the radioactive wastes generated from atomic power plants, nuclear fuel reprocessing plants and nuclear institutes are very difficult. Gas-phase decontamination technology using fluoric gases(dioxygen-difluoride O2F2, chlorine-trifluoride ClF3, krypton-difluoride KrF2) to convert to fluoric compounds(hexafluoride forms) of actinides(uranium, plutonium) have been already proposed and developed (1-6). But, gas-phase decontamination technology of radioactive transition elements, or transition elements and actinides(simultaneous decontamination) haven't been studied and proposed. The purpose of this study is to investigate the feasibility of simultaneous decontamination using gaseous reactions. Many radioactive transition elements(e.g. cobalt-60, cobalt-58, nickel-63, manganese-54, chromium-51, etc) as well as other neutron irradiated products and fission products(e.g. molybdenum, technetium, ruthenium, actinides: uranium/ neptunium/ plutonium) are troublesome in most nuclear sites or wastes. In the cases of transition elements and actinides, their carbonyl and fluoric compounds are volatile. The objective of gas-phase decontamination is to use CO gas for carbonylation and fluoric gases for fluorination to convert many nonvolatile radioactive transition elements and actinides to volatile chemical species.

It is important to develop the decontamination technology using gaseous reaction, because gas-phase decontamination has many merits.

PRINCIPLE OF DECONTAMINATION TECHNOLOGY

Carbonyl compounds of normal transition elements have volatile properties. And, fluoric compounds of actinides also have volatile properties. Table I and Table II show the thermal properties of typical carbonyl compounds and fluoric compounds of transition elements and actinides. From these thermal properties; low melting point and/or low boiling point, many carbonyl and fluoric compounds are gaseous, volatile or sublimate at room temperature. New decontamination technology reported in this treatise is based on and utilized these volatile properties.

Table I

General properties of carbonyl compounds are;

1) Molecular structure and electron dispositions are similar to inert gas's.
2) Gaseous, volatile or sublimate(low melting point and/or low boiling point).
3) Insoluble in aqueous solutions(water, acid and alkali). Soluble in organic solvents(benzene, ether, alcohol, etc.).
4) Easily flammable in air.
5) Thermal decomposition at high temperature. After thermal decomposition, carbonyl compound is decomposed to raw metal and CO gas.
6) High toxicity.

Table II

General properties of fluoric actinide compounds and specific transition elements are;

1) Gaseous or volatile(low melting point and/or low boiling point).
2) Vigorous reactivity with water. After reaction with and decomposition by water, hydrogen fluoride is produced.
3) High toxicity.

By the reaction of CO gas with transition elements on a material's surface, carbonyl compounds are produced. If any radioactive transition elements(e.g. cobalt-60, cobalt-58, nickel-63, manganese-54, chromium-51, molybdenum-99, technetium-99, ruthenium-106, etc.) exist on a material's surface, these transition elements are evaporated as gaseous forms. In the case of actinides, volatile fluoric compounds are produced by the reaction with fluoric gases(e.g. F2,HF,O2F2,ClF3,BrF5, etc.). The principles of this decontamination technology are shown in Fig. 1 and Fig. 2. And, chemical reactions used are indicated in chemical Eqs. (1)-(3).

Fig. 1

Fig. 2

Chemical Reactions of Carbonylation:

Eq. 1

Eq. 2

Eq. 3

Use of this principle based on carbonyl and fluoric gaseous reactions is a new concept as a decontamination technology for nuclear wastes. If gas-phase decontamination technology can be developed, it will be very practical and economically advantageous, because it is now very difficult to decontamine and treat the large volume of nuclear wastes; especially non-incinerable radioactive wastes.

EXPERIMENTAL

The experimental apparatus is shown in Fig. 3. It consisted of gas supply high pressure syringe pump(max. 240atm; made by ISCO Co./type 500D), reactor(od. 0.5", length 5"; made of stainless steel: SUS316L) with valve

and high pressure gauge(max. 500atm; made by KEYENCE Co./type AP16), oxidation furnace with heated wire-shaped CuO(500-550C) column flowing air for the treatment of poisonous exhaust gas, and a gas trap with activated carbon for back-up at the last position.
Fig. 3
To evaluate the removal efficiency of various nuclides by gaseous reaction, an inductive coupled plasma analyzer and radiation detectors (for alpha-, beta- and gamma-rays) were used to measure the amount of decreased nuclides on the mounting material. Initially, test samples were made of glass (borosilicate: tube and plate) and stainless steel (SUS304: plate) for mounting radioactive nuclides to insert into the reactor to prevent the contamination of the reactor by mounting directly and the effects of oxide layer on the reactor's surface. Known amounts of radioactive nuclides, diluted by nitric acid solution (0.01-0.1mole/l), were taken and placed horizontally onto the samples; the nuclides were mounted inside tube sample and out-side plate sample. The test samples with nuclides were heated in an oven to about 110-120C to dry. In the case of stainless steel, test samples of stainless steel were heated for one hour at 500C in air in an electric furnace. By this pretreatment, a reddish golden oxide layer was grown on the surface of stainless steel, including the radioactive nuclides in the oxide layer. After inserting this pretreated sample into the reactor, remaining air in the reactor was purged completely by argon gas. Next, reaction gas was charged into the reactor, and was highly pressurized by the syringe pump. In the case of charging fluoric gas, another tank was used to charge the reactor, because the syringe pump was not resistant to halogen gas corrosion. After charging the reaction gases, the reactor was heated by an electric furnace. Tests were done varying the heating temperature, time, pressure, mounting sample's materials(glass, stainless steel) and nuclides. In addition, different trapping methods for volatile compounds by liquid or solid removers were examined.
Experimental conditions were set up as follows;
1) Reaction temperature :room temperature-500C
2) Treatment time:5-60min.
3) Pressure :1-200atm.
4) Gases :CO,COF2,F2,CO2,Ar
(high purity, packed in metal cylinder)
5) Mounting materials of nuclides
: glass[borosilicate] tube id. 8mmx50mm(l),plate 10mmx50mmx1.2mm(t)
: stainless steel[SUS304] plate 10mmx50mmx1.2mm(t)
6) Nuclides :Co-60, Ni-63, Ru-103,
Uranium(UO22+) as nitrate
RESULT and DISCUSSION
At first, to determine the feasibility of this principle as a gas-phase decontamination technology, cold tests using non-radioactive nuclides were done. Elements tested were cobalt, chromium, nickel, rhenium, molybdenum, manganese, ruthenium, zinc. The results of the cold tests are shown in Table III. From these results, all elements used were removed more or less by CO gas treatment. The removal efficiency of chromium, nickel, rhenium and zinc is sensitive to CO pressure. By the results shown in Table III, it was verified that carbonylation has feasibility as a new decontamination technology for radioactive transition elements.
Table III

Next, hot tests using radioactive nuclides were done. Figure 4 shows the radioactive cobalt(Co-60: 2700Bq) removal efficiency by carbonylation under conditions of CO 200atm, reaction for 30 minutes, and reaction temperature heated stepwise up from room temperature to 400C using the same sample contaminated with cobalt-60. After stepwise treatment, the overall removal efficiency of cobalt-60 was 88-93% . In Fig. 4, the line graph is the cumulative removal efficiency and the bar graph is the removal efficiency of each temperature step.

Fig. 4

The radioactive nickel(Nickel-63: 130Bq) removal efficiency by carbonylation is shown in Table IV. From the results of nickel-63, it was clear that the removal efficiency of nickel-63 is high at high temperature and pressure. Specifically, at the treated condition of 350C and 50 atms CO gas for 30 min., a removal efficiency of greater than 98% was obtained.

Table IV

The radioactive ruthenium(Ruthenium-103: 2700-3200Bq) removal efficiency by carbonylation is shown in Fig. 5. Compared with the results in Fig. 4, ruthenium-103 was removed under lower temperature than cobalt-60.

Fig. 5

Next, by simultaneous carbonylation and fluorination, the co-decontamination of cobalt-60, ruthenium-103 and uranium was tried. The glass plate contaminated with known amounts of cobalt-60(540-600Bq), ruthenium-103(2100-2800Bq) and uranium (101dpm) was treated repeatedly twice for 10 minutes at 250C, pressurized total 200atm of mixed CO+COF2 gases. Measuring the glass sample after this treatment, cobalt-60, ruthenium-103 and uranium were removed 37%, 81% and 66%, respectively. Cobalt-60 was removed as carbonyl compound, uranium was as fluoric compound, but it is assumed that ruthenium removed by both chemical forms as carbonyl and fluoride.

All tests mentioned above were done using a glass sample. Next, using a stainless steel sample, the gas-phase decontamination tests were done. Stainless steel samples were pretreated with cobalt-60 (700-740Bq), ruthenium-103 (2800-3100Bq) and uranium(101dpm). As be expected by the direct carbonylation using CO gas only, the removal efficiency of cobalt-60 and ruthenium-103 were small. To remove the surface oxide layer for promoting the gaseous carbonylation and fluorination, the pretreatment using supercritical CO2, halogen (I2) and adsorbed water for etching reagents was examined. The stainless steel pretreated sample was inserted into the reactor, next the sample surface was wetted with water vapor by introducing hot water vapor from outside, then iodine vapor also was introduced at about 0.4-0.7 mmol. After the set up, the reactor was heated at 110C, then CO2 was introduced at 150 atm. Under these conditions of pressure and temperature, CO2 is supercritical. The reaction time was 20 minutes. After this treatment by CO2+I2+H2O, two tests were done. In the first test, 1atm of F2 gas was introduced in the reactor heated to 300C for fluorination. By this treatment, Cobalt-60 was not simultaneously removed, but ruthenium-103 and uranium were removed at 14% and 19-20%, respectively. Next, in the second test, 200atm of CO gas or mixed CO+COF2 gases were introduced for reactions of carbonylation and fluorination. By utilizing pretreatment at supercritical conditions, 95-99% of cobalt-60 was removed. It is assumed that the small amounts of cobalt-60 that remained were diffused into the substrate of stainless steel during the 500C pretreatment. By utilizing supercritical

pretreatment of CO2+I2+H2O, the carbonylation temperature of cobalt was lowered clearly from 300-400C as shown in the results in Fig. 4 to about 100C. The reason for promoting carbonylation is assumed to be that Cobalt-60 reacted with iodine converted to iodide(CoI2), then the reactions of CoI2+CO+Fe(SUS)>Co2(CO)8+FeI2 progressed on the stainless steel sample. Next, by the direct treatment of mixed CO+COF2 gases after the supercritical treatment, the removal efficiencies of cobalt-60, ruthenium-103 and uranium were 30-35%, 49-57% and 55-60%, respectively. These removal efficiencies from oxidized stainless steel were similar to the tests on the glass sample. By utilizing supercritical pretreatment, the feasibility to remove radioactive nuclides from a metal surface with hard oxide layer was determined. The results mentioned above for co-decontamination(Cobalt-60, Ruthenium-103 and Uranium) using a glass sample, and the pretreatment effect of supercritical CO2 using a stainless steel sample are shown in Table V.

Table V

Because the treatment of supercritical CO2+Hologen(I2)+H2O has the ability to remove a hard oxide layer of stainless steel completely, it is expected that this treatment method will be useful in many other fields as a metal surface treatment or other applications.

CONCLUSION

The newly developed concept of gas-phase decontamination, which is an advanced waste management technology, was proposed. The results of the feasibility tests on this new technology, under high CO pressure(50-200atm) with heating(about 350C) using non-radioactive (cobalt, chromium, nickel, rhenium, molybdenum, manganese, ruthenium, zinc) and radioactive nuclides(cobalt-60, nickel-63, ruthenium-103) of transition elements, indicate that all these elements and nuclides were removed as gaseous forms. In the cases of cobalt-60 and nickel-63, they were removed to 88-93% and above 98% by treatment under CO 200atm, max.400C(30min.) and under CO 50 atm, 350C(30min.), respectively. In the case of stainless steel with a hard oxide layer, it was very difficult to remove the radioactive nuclides in the hard oxide layer by gaseous reaction. But removal of the oxide layer and carbonylation were promoted by utilizing the pretreatment of supercritical CO2+I2+H2O. Cobalt-60 in the hard oxide layer of stainless steel was mostly removed. From the co-decontamination experimental results using mixed CO+COF2 gases pretreated with supercritical CO2+I2+H2O, the removal efficiencies of cobalt-60, ruthenium-103 and uranium were 30-35%, 49-57% and 55-60%, respectively. By the experimental results indicated above using non-radioactive and radioactive nuclides, the feasibility of this new idea was determined. In the future, gas-phase decontamination will be a practical technology by determining the optimum treatment conditions. This proposed decontamination technology based on gaseous reaction has the ability and possibility to decrease drastically the large volumes of non-incinerable radioactive wastes, which include burned-ash, many kinds of used-metals, and equipment.

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23-4

A NEW CUTTING AND PACKAGING TECHNOLOGY FOR LARGE ACTIVATED AND CONTAMINATED COMPONENTS FROM NUCLEAR POWER PLANTS

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ABSTRACT
In the past few years, large components from the primary cycle in a number of nuclear power plants have been exchanged.
These include steam dryers and steam separators with diameters and heights greater than 5 meters and with weights up to 50 tons. Since the construction of the components is in part very complex and they exhibit surface doses of several mSv, cutting with thermal equipment is very difficult and is associated with relatively high dose exposures for personnel. The complex design and the size of the parts make remote handling of the equipment practically impossible.
For these reasons, GNS had developed a concept in which the cutting is performed using a heavy-duty hydraulic scrap cutter, which is installed on an electrically driven caterpillar vehicle with an extending arm. Experiences with the cutting of similar, but unless complex components in a non-nuclear field of application are available and were taken into account in establishing the concept.
The cutting and packaging of a steam dryer and a steam separator are scheduled to take place in a German power plant; in the process the parts are to be packaged in final disposal containers and then brought to the final repository ERAM.
In the report, the concept for cutting and packaging shall be explained and the experience gained from the actual cutting in the nuclear power plant shall be discussed.

INTRODUCTION

In one German BWR-nuclear power plant, a steam dryer and a steam separator have been stored for about 20 years. These large components have not been cut up until now because, on the one hand, the dose rate at the components was too high, and on the other hand, a practical cutting technique was not available.
The present report begins with a description of the properties of the steam dryer and the steam separator as well as of the equipment planned to be used for cutting, and describes the sequence of cutting operations, the necessary auxiliary equipment and the estimated time required.
The cutting is planned to be done with a hydraulic scrap cutter, which is installed on an extending arm of an electrically-driven caterpillar. An essential requirement is the possibility of not having to perform the work by remote control.

Experience with the cutting of similar components, if not as complex, exists in the conventional field of application. Cutting trials as an approach of this cutting work have been performed.

DESCRIPTION OF THE COMPONENTS

Steam Dryer

The steam dryer has a diameter of approx. 5200 mm, an overall height of approx. 5500 mm and a overall weight of approx. 38 t. Its main components are pipes, water channels and conductive steel plates. The walls of the steel plates have a thickness of up to 10 mm.

The materials used are mainly high-alloyed steels with the designations 1.4550 and 1.4541.

More massive structures are contained in the supporting frame. It consists of a ring at the top (length of edge: 50 mm) and a ring at the bottom (length of edge: approx. 155 x 80 mm). In the vertical position, the bottom ring is at a height of approx. 1600 mm above the bottom, the top ring approx. 4200 mm. In the top ring, cross struts are still present (length of edge: 50 mm). The top and bottom rings are connected to each other by means of massive supports (length of edge: 50 mm).

The steam dryer rests on 4 legs, so that the lower pipe structures are not in contact with the floor.

Steam Separator

The steam separator has a diameter of approx. 5200 mm, an overall height of approx. 4400 mm and an overall mass of approx. 50 t. Its main components are the clamping ring, the bottom part and the separator part, that is to say the steel plates, the pipe-like structures and the channel profiles. The walls have a thickness of approx. 8-12 mm. More massive parts have also thicknesses up to approx. 60 mm. The materials used are X10 CrNiTi 189 and X10 CrNiNb 189.

The massive structures of the clamping ring are set on the floor. The rest of the steam separator with a maximum diameter of approx. 5200 mm rests thus on the clamping ring, whose diameter is approx. 4500 mm. The height of the clamping ring is approx. 450 mm. Cyclone structures are accommodated in the protruding segment. They are approx. 1500 mm high. Figure 1 shows the components to be cut.

Fig. 1

CONDITIONING CONCEPT

The conditioning of the components described in Section 2 is performed according to a principle which is successfully used by industrial facilities and installations for the conventional disposal as well as for the disposal of concrete structures.

For cutting, the equipment, which is described in detail in the following section, is installed on a manipulator vehicle. With the manipulator vehicle and the manipulator arms, the cutting equipment is positioned for cutting. For this purpose, it is planned to use two different types of cutter, which are dimensioned such that they can cut the metal structures. The cut-out parts can be picked up by the cutter and loaded into drums which have been prepared beforehand.

Cutting Equipment

As cutting equipment, a tank cutter well tried in the conventional industry and a scrap cutter are planned. The manufacturer's data about the dimensions and weights are summarized in Table I. Figure 2 shows the use of the cutter type UP 50 with scrap cutter mouth, and presents the difference between both cutter mouths which shall be used for the cutting of the components.

Fig. 2
Table I
Due to the difference of the structures (steel plates, massive bars) of the material to be cut in the components, the use of both cutter mouths presented is necessary (see Fig. 2). The change from the tank cutter mouth to the scrap cutter mouth takes approx. 1 hour according to the experiences gained in the conventional field. In order to switch the cutter mouths, the caterpillar sets the installed cutter mouth in a frame, the mouth is loosened and the extending arm of the caterpillar switches to the next cutter mouth. The new cutter mouth is installed according to the assembly instructions.

Manipulator

As the manipulator vehicle, a commercial crawler (caterpillar) is planned, on the arms of which the above cutting equipment is attached. With the crawler, the loading is evenly distributed on the ground. Since the vehicle, in the operation planned, can only be used in closed spaces, an electric motor is used instead of a combustion engine.

In order to ensure sufficient radiation protection for the handling personnel, a new handling cabin is installed.

The main dimensions and weights of the planned manipulator are presented in Table II.

Table II

The working range of the cutting equipment is presented in Fig. 3.

Fig. 3

The exact positioning of the cutting equipment is made possible by lifting and lowering the extending arms, with the possibility of turning the cutting equipment by 360 around the extending arm and with the right placement of the vehicle. When the scrap cutter mouth is used, the angle of the mouth during cutting is adjusted by the hydraulic cylinder so that the mouth side with the highest counterforce is stationary while the other mouth side closes the mouth opening, until the same work pressure is reached in both working cylinders.

Cutting and Packaging Concept

The cutting of large-volume components with the equipment planned here is state of the art in the non-nuclear field. The components planned now are different due to the high number of combinations of steel plates and massive supports and the geometric arrangement, which is often complicated. For this reason, a detailed cutting plan cannot be prepared at the moment. The execution of the cutting will more depend on the experience of the personnel on site and on the experiences successively gained.

The cutting is planned to be performed from the top downwards and from the outside towards the inside. All structures above the bottom support ring are cut. The only and necessary exceptions are the supporting bars between the bottom bearing ring and the yoke.

The replacement of the cutting equipment is done according to the cutting progress. For time reasons, as few replacements as possible shall be performed.

Placement of the Scrap into Drums

The parts cut out of the components with the tank cutting mouth have approximately the dimensions corresponding to the depth and width of the mouth. They are held in the mouth and are placed directly into a 200-l-drum ready for loading. Cut out sections which are longer can be pulled through the mouth without any external assistance and are slightly bent.

This "bent" strip can also be put in a prepared 200-l-drum using the tank-cutter.
During cutting with the scrap cutter mouth, only simple cuts can be made. This leads to the fact that some parts cut out remain on the components to be cut or fall onto the floor. The cuts are thus selected by the operator such that parts are produced which can be placed directly into a drum. These parts are picked up with the cutter and put into a ready drum. Larger parts are placed onto a specially prepared area and are further cut up.
Cut up scrap parts, which lie on the floor and can not be picked up by the cutter are pushed or pulled in to position from the edge of the steel plate with a manipulator or the cutter and from there picked up with a tool and placed into drums.
The mass of metal per 200 l-drum has been currently estimated to be approx. 50 - 200 kg, 100 kg on the average. For a total mass of the steam dryer of approx. 38 t, approx. 400 200-l-drums will be needed. Similar conditions apply for the steam separator.
Maintenance of the Equipment Used
The equipment planned has proven itself in conventional use. Maintenance and repair are only required after throughputs and lengths of time which are not expected for the present case.
The cutter with the differing mouths can be modified and maintained without any special expense or special equipment. Also, workers are trained for the project by the manufacturer/supplier in the scope of the preparatory functional testing.
The operating time of the moving mechanism for the further present cutting purpose is so small as compared with operating time in conventional areas, that extensive maintenance or repair work can be excluded. As a rule, all work can be performed by the power plant's own maintenance staff. The manufacturer or supplier is consulted if needed.
Radiation Protection
The radiation protection aspects of the conditioning campaign are as high a priority as the technical realization.
Through the use of heavy-duty equipment, for which no substantial repairs are expected, it is possible to cut and package the steel components rather quickly. Wherever possible in all working places, shielding is used, including in the shielded cabin which is installed on the manipulator vehicle. Conservative estimates of the individual and collective dose rate have been made and show that the work according to the concept described here is fully acceptable. The average individual dose for all persons involved with the cutting work is approx 3.6 mSv. The operators of the manipulator vehicle have the maximum individual dose of approx 9 mSv.
SUMMARY
The use of the cutting equipment presented in combination with the vehicle has been tried and tested for years and has proven itself in the conventional industry. This refers not only to the cutting concepts and the execution on site but also to all technical systems such as moving mechanisms, drive, hydraulic and electric supply.
Typical areas of use are the disassembly of chemical installations and refineries. Essential aspects are minimizing the aerosol load, avoiding risk of explosion, and the distinct acceleration of the cutting process compared to manual techniques such as flame-cutting.

Preliminary tests have been performed on various steel plates with a mobile tank cutter, which had been in service already for a long time. For this reason, there was too much leeway between the cutter and the holder (finger). Plates made of the material 1.4571 (with strength values similar to 1.4550) were used:
2 pcs. 4 x 500 x 500 mm
4 pcs. 10 x 500 x 500 mm
By combining various plates, material thicknesses of up to 30 mm were reached. Up to a thickness of 25 mm, all combinations were cut safely. When correctly adjusted, cutting of plates with 30 mm and more can be done satisfactorily.
In one further test with a UP 50 shears having a scrap cutter mouth, full-steel profiles with properties like 1.4550 were cut.
In order to demonstrate the cutability of the edge profiles on the steam dryer, cross-sections of 50 x 50 mm and 60 x 60 mm were used for the test. The profiles were able to be cut in a satisfactory manner.

23-5
A STUDY OF AIRBORNE RADIOACTIVITY GENERATED DURING THE PREPARATION OF CONCRETE WASTE FROM REACTORS UNDERGOING DECOMMISSIONING*
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ABSTRACT
The demolition of activated concrete waste from reactor biological shields undergoing decommissioning produces the potential for worker exposure to radioactive aerosols. The determination of appropriate derived air concentrations (DAC) for worker protection during routine and accidental situations depends on a knowledge of the particle size distribution of the aerosol. Measurements are described which were used to evaluate the mass median aerodynamic diameter (MMAD) and the activity median aerodynamic diameter (AMAD) for aerosols generated by cutting two types of activated reactor concrete under laboratory conditions using a sawing device and a coring tool. Samples were also taken during the actual demolition of the biological shield at the Experimental Boiling Water Reactor (EBWR). In the laboratory study, twelve sample collection experiments were performed as part of a 2x2 factorial statistically designed experiment with three replications. The results indicated that the type of cutting tool used significantly affected the MMAD and the AMAD with coring values being much larger than sawing values. The numerical value of the 60Co AMAD for sawing was found to be more than an order of magnitude greater than the ICRP recommended default AMAD of 1 mM. These results suggest that DAC values used at the worksite should be based on experimental data instead of regulatory assumptions, or that larger default values should be used in practice. The results also showed that particle size distribution was not significantly different for the two types of concrete studied. Also, the MMAD and the AMAD were found to be not statistically different for the various samples. In contrast to this, one of the field samples taken during the demolition of the concrete reactor shield by an air-hammer under wet conditions at the EBWR had an AMAD much smaller than the MMAD and also smaller than the 1 mM default value. One possible explanation is that 60Co is being leached

from the concrete dust by the water spray resulting in smaller aerosol droplets. Further research with other types of concrete and other cutting devices is needed, and the author is seeking such samples from other reactors being decommissioned.

INTRODUCTION

The demolition of reactors undergoing decommissioning usually involves the cutting of large pieces of activated concrete for shipment to waste disposal sites. Each of the cutting techniques produces a radioactive dust, and although efforts are made to mitigate the dust production, there still remains the possibility of inadvertent worker inhalation through accidents, respirator failure and enclosure failure. Very few studies have been performed to date on the specific radiological hazards of these airborne radioactive particulates. The purpose of this research was to characterize the particle size distribution of these concrete aerosols in order to provide data for the determination of appropriate derived air concentrations (DAC) for worker protection during routine and accidental situations. Specifically, determinations were made of mass median aerodynamic diameters (MMAD) and activity median aerodynamic diameters (AMAD) for aerosols generated from cutting activated reactor shielding concrete. In addition, factors which affect the parameters of the distributions were investigated. The factors included two types of concrete and two different cutting tools. Experimentally determined distribution parameters were compared to those assumed in regulations which limit the occupational exposure to radioactive aerosols. Mass distribution parameters were compared with their activity distribution counterparts. Qualitative comparisons were made between samples collected in the laboratory at Purdue and samples taken in the field at the Experimental Boiling Water Reactor (EBWR) at Argonne National Laboratory during actual reactor decommissioning procedures.

EXPERIMENTAL

For the laboratory study at Purdue, activated concrete blocks were obtained from the EBWR and the University of Washington Nuclear Reactor (UWNR). A diamond circular saw and a diamond coring tool were used to cut the activated concrete under dry conditions in a radiological glove box. An 8-stage cascade impactor was used to collect the aerosol samples. The air flow through the impactor was controlled by a critical orifice designed to maintain a 0.25 ft3 min-1 flow rate through the impactor. Masses collected on glass fiber substrates were measured with a microbalance, and 60Co activity was measured with a hyperpure germanium detector. Twelve sample collection experiments were performed as part of a 2x2 factorial statistically designed experiment with three replications. The cascade impactor was transported to the EBWR for use during the demolition of the biological shield with an air hammer. The air hammer moved in all directions, thereby dismantling the shield from the inside out. A water misting spray was used to reduce airborne aerosol concentrations. The cascade impactor was placed on a ledge of a penetration of the concrete shield approximately half-way up the shield and just inside a plastic curtain which had been erected to keep the dust out of the penetration.

RESULTS

The mass and activity data from each impactor stage used in the laboratory study were entered into a graphics software package which generated a log-probability plot along with a fitted least squares line. Figure 1 is a plot of the mass data from sample 11 as an example. The

equation of the line is given which yields an MMAD of 13.7 mM. Figure 2 is the activity plot of sample 11 which yielded an AMAD of 13.6 mM. Table I is a listing of the response variables for the 12 experiments. It can be seen that the 60Co AMAD values are more than an order of magnitude greater than the ICRP recommended default AMAD of 1 mM. Consequently, the usual practice of setting 60Co Annual Limit on Intake (ALI) and Derived Air Concentration (DAC) values based on an assumed AMAD of 1 mM would result in perhaps unnecessarily conservative regulations for dry cutting conditions. The results of an AMAD larger than the 1-m default value are consistent with observations presented in Dorrian and Bailey's recent review paper (1). The results also support the new recommendation of a 5-m default value given in ICRP Publication 66 (2).

Fig. 1

Fig. 2

Table I

The response variables are somewhat similar within each group of three replicates. This could be explained by the tight control on experimental techniques and methodologies used during the processing of a sample. It may also be attributed to the relative homogeneous nature of the concrete ingredients in those particular locations where cuts were made. The R2 values for all the samples are very close to unity. This indicates that the data follow a nearly perfect linear line. The aerosol distributions are, then, almost assuredly log normal in nature. This agrees with the ICRP functional form assumption of the aerosol distributions as used for setting ALI and DAC values.

The results of the statistical analysis on the two types of concrete and the two types of cutting tools are given in Table II. This portion of the statistical analysis shows that the aerosol particle size distributions are significantly different for the two types of cutting tool. Tests on every response variable showed this finding. It is important to note that the results of the statistical tests for MMAD and the mass fraction less than 10 mM agree with each other. The same is true for the agreement between AMAD and the activity fraction less than 10 mM. The reason for its importance is that the MMAD and the AMAD values for the coring samples were determined by extrapolating the log-probability plot fitted line. Whenever extrapolations of experimental data are performed the results become less credible. The mass and activity fractions less than 10 mM, however, were determined without extrapolation. The agreement of the statistical tests between these response variables indicates that extrapolation did not adversely affect the analysis.

Table II

The distributions were not significantly different for the two types of concrete. This finding is somewhat surprising given that the mixture components of the concrete are quite different. It indicates that cutting tool rather than type of concrete has a strong influence on the resultant aerosol distributions. Health physicists and program managers may wish to incorporate the range of particles produced by different cutting techniques in their decision making process for choosing particular techniques for demolition projects. An interaction between cutting tool and concrete type was found to be not significant for each response variable. Therefore, the effect of cutting tool on the resultant aerosol distributions was not influenced by the type of concrete being cut.

The second portion of the statistical analysis was to determine if the mass distribution response variables differed significantly from their

associated 60Co response variables. Table III lists the results of this test. The results of this analysis have shown that the mass response variables do not significantly differ from their 60Co activity response variable counterparts. This interesting finding may suggest that the particular levels where the cuts were made had uniform activation levels.

Table III

Table IV lists the response variables for two samples taken at the EBWR during actual demolition of the activated concrete bioshield wall. Comparisons with the laboratory study at Purdue are quite tentative because the samples were taken under quite different conditions. Nevertheless, a few observations can be made. Although the EBWR samples were taken under wet operating conditions, the MMAD were again much larger than 1 mM. The absolute MMAD values of the EBWR field study where demolition was done with a Brokk air hammer, were quite similar to the MMAD of particles produced by sawing in the laboratory study at Purdue.

Table IV

The 60Co activity measurements of the EBWR samples produced some unexpected results. While it was shown earlier that there was no statistical difference between the MMAD and AMAD of the Purdue samples cut under dry conditions, such is not the case for those collected at the EBWR. Sample 2 produced the most striking results in this respect. Although its MMAD value is much larger than 1 mM, its 60Co AMAD value is much smaller. This finding suggests that while the use of water does not have a great affect on the mass distribution of the particles produced by the cutting, it does change the radioactivity distribution to a large extent. A visual inspection of the after filter (AF) substrate revealed a conspicuous dark water stain. This substrate was measured as having over 60% of the total 60Co activity that was collected on all of the impactor stages. A possible explanation for the very small AMAD value could be that the 60Co was leached from the particles due to the use of the water misting spray. The 60Co may then have been attached to water droplets so small that they were not caught by the impactor stages until they reached the after filter. Under this operating condition, setting 60Co ALI and DAC values based on the ICRP recommended AMAD value of 1 mM may be too liberal.

The use of water during demolition operations is routine. Therefore, it appears incumbent upon health physicists to determine actual AMAD values during working conditions and calculate ALI and DAC values accordingly. This should be done instead of relying upon the ICRP recommended AMAD value of 1 mM. This applies for both dry and wet demolition conditions. The results of this research indicate that, under both dry and wet conditions, the usual assumption of a 1 mM AMAD would result in inaccurate worker internal dosimetry determinations. In addition, the water used in the misting spray or for cooling the cutting tools will be contaminated and should be collected and analyzed for radioactive concentrations. These results also indicate the importance of knowing the lung solubility classification and the leaching characteristics of radioactive concrete aerosol particles.

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AN ALTERNATIVE USE OF SURFACE CONTAMINATION LIMITS IN BUILDING DEMOLITION

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ABSTRACT

The current surface contamination limits used for release of facilities for unrestricted use are applied to decommissioning projects to verify that the levels of removable contamination or residual total (fixed and removable) contamination that could pose health risks to the general public are acceptably low. The acceptable amounts of residual radioactivity remaining should be taken in context of how the structure or surface is to be used in the future. Risks associated with various levels of residual surface contamination are correlated to accepted risk based standards. One of these risks is potential exposure to radioactive material in the industrial rubble resulting from structure or surface demolition and disposal. However, if the demolished material is used as dispersion media, this exposure risk is reduced and the projects costs are lowered by eliminating the need for costly surface decontamination. This paper will discuss the use of this methodology to release a structure.

DECISION PROCESS

The following is a list of items that must be considered in order to use surface contamination limits in building demolition.

1. Determine source term
Total radioactivity involved per radionuclide
Determine solubility class of radionuclides
2. Determine any hazardous waste concerns
RCRA or TSCA waste streams
3. Determine volume of rubble to be generated
4. Determine disposal options
Recycle
Road bed material
Erosion control
Construction fill
Ferrous metal recycling
Other metal recycling
Wood chips for fuel
Disposal
Industrial landfill
Sanitary landfill
5. Obtain regulatory approval

Perform pathway analysis and risk assessment
6. Perform soil sampling for release of structure footprint

DOSE CALCULATIONS

Internal exposure is the predominant exposure pathway during building demolition because surface contamination has a high potential for becoming airborne. Strict radiological controls are required at this phase to ensure the health and safety of the workers, the public, and the environment. Appropriate monitoring, including personnel air samples are required. At this point, occupational radiation exposure through the inhalation pathway is the primary area of concern.

After demolition, the primary exposure pathway is through external gamma irradiation from the deposition of the material. This exposure may come from submersion in a contaminated atmospheric cloud, immersion in contaminated water, or exposure to contamination on or in the ground. The Regulatory agencies may request a pathway analysis that determines the dose to members of the public from pathways of concern like food, fish, meat, water, and air. Dose can be determined by using the following equation:

Eq. 1

If the resultant public dose is less than 0.1 mSv/year (10 mrem/year) to any individual, the material may be disposed of in the manners proposed. The resultant public dose may be negotiable with the applicable regulatory body.

DISCUSSION

A determination should be made on the average contamination levels on or in the walls, floor, and ceiling. For simplicity, the building is assumed to be empty. Then, the density and thickness of the material is used to calculate the activity per gram. If, items like sheet rock and wood are crushed and mixed with the concrete and rubble during the demolition process the AVERAGE density may be used for the material where the contamination will be dispersed. This results in a surface activity level coefficient.

Eq. 2

Typical surface contamination limits are expressed in dpm/100 cm2, for example, 1000 dpm/100 cm2. If we divide this by 2.22 dpm/pCi we can obtain the pCi/100 cm2, for the above example we get 450 pCi/100 cm2. Taking the surface activity level coefficient and dividing it by the thickness of the material gives the activity per gram dispersed through the material as it is broken up during demolition.

Eq. 3

Estimations of additional material not accounted for in the contaminations is added to the activity concentration by dividing the estimated total activity by the sum of the total contaminated grams plus the total non-contaminated grams.

CONCLUSION

Eq. 4

By uniformly dispersing surface contamination during the building demolition process, the average activity per gram of the material will be well below any regulatory guideline. This process allows the material to be recycled for beneficial use without any harm to the health and safety of the workers, public, or the environment.

DECOMMISSIONING AND CLOSURE OF A MANUFACTURING PROCESS THAT GENERATED LOW LEVEL MIXED WASTES

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ABSTRACT

This paper describes the strategy that was developed to perform facility decommissioning and license termination for a polymer manufacturing facility where uranyl acetylacetonate (UAA) and uranyl dinitrate were used as catalyst components for several blends of polymeric compounds. Elimination of this product line by the manufacturer resulted in the need to prepare a facility decommissioning and license termination plan acceptable to the New York State Department of Labor (NYSDOL) as the licensing agency, implementation of the plan consistent with New York State Department of Environmental Conservation (NYSDEC) regulations, and preparation of a facility closure and license termination report.

Upon elimination of the product line, excess UAA and UNH product intermediates and low level radioactive waste such as QA/QC samples required disposal as part of NYSDOL licensing requirements. Because the intermediates would be considered a low level radioactive waste, and the UNH possibly a mixed waste due to an oxidizing characteristic and, it was decided to reprocess the intermediates to a final product prior to disposal. The situation was further complicated by the fact that no disposal facility would readily accept the intermediates, the hazardous waste characteristics of the final product were unknown prior to reprocessing, regulatory approval by both the NYSDOL and NYSDEC to initiate reprocessing was required, and the Barnwell, SC disposal facility (the most likely receiving facility) was to close in the near future.

This paper describes the steps in developing the overall decommissioning strategy to successfully terminate the manufacturer's license. The most challenging aspect of the decommissioning procedure was to find and implement an acceptable regulatory strategy which would not require a Resource Conservation and Recovery Act (RCRA) treatment permit and still meet the time frame associated with closure of the Barnwell disposal facility that has since reopened.

INTRODUCTION

A manufacturing facility located in New York State utilized radioactive compounds as components for polymeric materials produced from 1987 to 1993 for use by the U.S. Department of Defense. A radioactive materials license was issued to the manufacturer to receive, possess, use, and transfer radioactive materials pursuant to the State of New York Industrial Code Rule 38, Ionizing Radiation Protection (12 NYCRR Part 38). Radioactive materials designated under the license were natural or depleted uranium compounds, uranyl acetylacetonate (UAA) and uranyl dinitrate hydrate (UNH), in a solid or liquid form in a quantity not to exceed 11 millicuries at any one time.

GENERAL PROCESS DESCRIPTION

UAA and UNH are low activity (46 microcuries per kilogram) materials. Formulation of polymer products required approximately 136 kilograms of UNH or UAA during a typical production year. During the manufacturing process, UAA and UNH were blended into an intermediate product from which a Part B product was made. A Part A product was also produced, but did not contain any radioactive materials. Parts A and B were mixed together

by the consumer to make a final product. It is important to note that the polymer product required UAA and UNH chemical properties and not their radioactive properties. Batch operations to produce UAA and UNH containing products required approximately two days of operation and were conducted approximately four times per year.

Boxed, one gallon metal cans of UAA and UNH were delivered to the facility's receiving dock, where the Radiation Safety Officer (RSO) documented shipment receipt, and personally transported the containers to the radioactive materials storage cabinet. Each one gallon can contained 3.63 kilograms of UAA or UNH in a plastic bag, within the can. The cans were not opened at any time during delivery to the storage cabinet.

When dispensed, one can of UAA or UNH was transported from the radioactive materials storage cabinet to a scale glove box, located in a mixing room. The can was placed in the glove box, the can and plastic liner were opened, and UAA or UNH powder was placed in a paper cup on the scale. When the desired weight of UAA or UNH powder was obtained, any remaining powder in the can was resealed in the plastic liner, the metal cover replaced, and the can returned to the radioactive materials storage cabinet. When empty, the can along with the plastic liner were placed in a 55 gallon drum for disposal. The scale glove box contained a local air exhaust duct to capture dust generated by the weighing operation.

The paper cup containing UAA or UNH was carried from the scale glove box to a mixing vessel, where the contents were poured into the vessel through an addition hatch. A local air exhaust duct was located at the addition hatch to capture fugitive dust. The mixing vessel contained organic solvents that dissolved the UAA or UNH. During mixing, a vacuum pump was used to degas this intermediate product.

When processing of the intermediate product in the mixing vessel was completed, it was drained from the bottom of the vessel and placed in another mixing vessel and blended with additional raw materials to make the Part B product. A vacuum pump was used to degas Part B during mixing. When mixing was completed, the Part B was dispensed from a bottom port of the mixing vessel and packaged into containers to be sent to the customer.

Samples of intermediate product and Part B were obtained from each batch produced for Quality Assurance/Quality Control (QA/QC) purposes. Samples of intermediate were filtered prior to dispensing to assure that no solids were present. QA/QC samples were transported to the QA/QC laboratory in covered containers. Filters and paper cups that contacted UAA or UNH containing material were placed in a 55-gallon drum for disposal. Gloves and paper coveralls worn by employees working with UAA or UNH were also placed in a 55-gallon drum.

Intermediate product and Part B samples were subject to different test procedures upon receipt by the QA/QC laboratory. Intermediate product samples were logged in and an aliquot was weighed on an analytical balance. Afterwards, it was dried and degassed in an oven, and volatilized in a furnace for residual analysis. Part B samples were logged in, and an aliquot reacted with Part A, cured in an oven, and the reacted product tested for physical parameters. Intermediate product and Part B samples not analyzed were stored in a holding area.

Unused intermediate product, Part B, and materials that came in contact with UAA or UNH were placed in a 55-gallon drum. The drum inventory at the conclusion of operations involving UAA and UNH consisted of twenty three 55-gallon drums. The drums were located in the shipping area and

consisted of 6 drums of dry solids, 6 drums of intermediate product, and 11 drums of Part B product. Product manufacturing components and laboratory apparatus associated with QA/QC continued to be used for non-radioactive product production. Figure 1 illustrates the handling steps associated with UAA and UNH in the manufacturing process.

Fig. 1

SITE RADIATION CONTROL

Existing procedures to control radioactive material contamination consisted of wipe sample counting and gamma readings following the use of UAA or UNH. If radioactive material was detected, the corresponding area was cleaned and resurveyed to assure the area was free of contamination. Wipe sample counting was conducted by facility personnel using a Ludlum Model 43-10 alpha sample counter attached to a Model 2000 scaler. Gamma measurements were collected with a Ludlum Model 3 survey meter.

Most surface contamination measured during surveys conducted by the manufacturer was less than 20 disintegrations per minute (dpm)/100 cm2 with the highest surface contamination at 1,000 dpm/100 cm2. Gamma levels detected were less than 0.5 milliroentqens (mR)/hr.

Based on existing records, survey information, and control procedures implemented during UAA and UNH use, wide-spread radiological surface contamination was not expected during decommissioning.

DECOMMISSIONING PLAN

Radiation survey methods submitted to NYSDOL and conducted at areas where loose UAA or UNH were handled or processed consisted of a real-time radiation survey with an alpha probe and collection and analysis of wipe samples for designated areas. Wipe samples were collected using paper backed smears by wiping a 100 cm2 surface area, and placing the sample in a coin envelope. Wipes were counted on site for alpha radiation with a Ludlum Model 43-10 alpha sample counter attached to a Model 2000 scaler to measure dpm/100 cm2.

Wipe samples were also collected where loose UAA or UNH were handled or processed. These areas consisted of the scale glove box, mixing vessel room floor, and associated mixers. In addition, the interior surface of the air exhaust system and the vacuum lines associated with mixing vessels were sampled. One wipe sample of 100 cm2 was collected per three square feet of surface area exposed to UAA or UNH dust. Floor areas were sampled at materials use locations. The air exhaust system was wipe sampled at the openings of the exhaust duct for the scale glove box. Wipe samples were collected at the exhaust clean-out door, blades of the exhaust fan, and the exhaust duct cap. Two wipe samples were collected from overhead rafters in the mixing room and two wipe samples were collected from the wall adjacent to the scale glove box.

An alpha and gamma survey was conducted in areas where UAA and UNH were used and also where wipe samples were not collected. Survey instrumentation consisted of an Eberline Model ASP-1 survey meter and to an HP-260 pancake probe.The area survey was conducted at the following locations:

- in and around the radioactive materials storage cabinet
- below the two mixing vessels
- the floor area next to the walls of the first floor mixing room
- the water drains in the first floor mixing room
- bench tops and hood interiors in the QA/QC laboratory
- interiors of the furnaces and ovens in the QA/QC laboratory
- shelves and floor area of the sample storage area

floor surface of the waste drum storage area.

A guideline to trigger additional wipe sample collection and measurement for alpha radiation activity was detection of gamma radiation in excess of 0.25 mR/hr at 1 centimeter from the surface.

The site decommissioning plan contained a decontamination plan that would be activated if surface wipe radiation contamination measurement exceeded 33 dpm/100 cm2. A clean surface was defined as a wipe sample measurement less than 33 dpm/100 cm2 and survey probe measurement less than 0.25 mR/hr at one centimeter from the surface. Also, five percent of surface wipes were to be quality assurance tested with a Canberra Model 2404 proportional counter to detect alpha particle radiation to a level less than 1 dpm.

Proposed personal protective equipment consisted of a dust mask (Moldex 3400 or equivalent), rubber gloves, and Tyvek coverall with shoe covering. Prior to leaving the decontamination area, workers were to be whole body surveyed with an alpha scintillation detector and/or beta-gamma detector. The worker whole body survey area was scheduled for a low radiation background area. Radiation contamination identified on workers was to be removed using soap and water located in the decontamination area.

When required, worker training was to be provided to workers conducting decontamination activities as specified in 12 NYCRR Part 38 and 29 CFR Part 1910.1200. Training included identification of the hazards associated with UAA and UNH and instruction in the use of personal protective equipment (PPE) required for these activities. Decontamination wipe methods and disposal of wipe cloths and PPE were to be discussed as well as personnel radiation monitoring and bioassay methods.

CONTAMINATED MATERIAL DISPOSAL

Termination of the NYSDOL radioactive material license required surface work areas to not exceed those levels identified in the decommissioning plan and the removal of all radioactive material associated with the license from the facility.

Containers located at the facility consisted of 6 drums of dry solids such as paper cups, Tyvec suits, and gloves; 6 drums of liquid intermediate product; and 11 drums of Part B product. The 6 drums of dry solids were characterized as a low level radioactive waste and did not exhibit a mixed waste characteristic.

However, based on material safety data sheet for UNH, the 6 drums of intermediate product and 11 drums of Part B product contained a quantity of UAA and UNH sufficient to classify them as a RCRA mixed waste if they were not reprocessed. The solution disposal of the 17 drums containing intermediate and Part B product was to remove the oxidizing characteristic from the material. Liquids required solidification to meet disposal requirements of the potential receiving facility, Barnwell, SC. Regulations and therefore a method to eliminate the oxidizer component and solidify the waste for disposal as a radioactive material were investigated.

Both the New York State Department of Environmental Conservation and NYSDOL were notified and concurred with the plan to solidify these liquid wastes for transportation to the Barnwell, SC disposal site.

The manufacturer indicated that the 11 drums of Part B product could be mixed with materials chemically similar to the Part A component originally required to be reacted with Part B to create the final polymer product. This process was anticipated to result in a hardened polymer

that would react the nitrate component of the UNH and therefore would remove the oxidizer characteristic of the waste. Stoichiometric addition of Part A component to the Part B waste was conducted by the manufacturer and allowed to cure in the waste drums.

The 6 drums of intermediate product required additional processing prior to reacting with a Part A component to eliminate the oxidizer characteristic. Records of waste drum contents were reviewed, addition materials were identified and stoichiometrically added to the intermediate product, resulting in polymerization that met solidification requirements. The additional polymerization material increased the total waste volume from 17 drums of liquid to 27 drums of solidified product. Samples of the solidified products were analyzed for radioactivity and RCRA hazardous waste characteristics. The solidified product did not exhibit a RCRA hazardous waste characteristic; therefore, the manufacturer was able to transport the solidified waste drums to Barnwell, SC for disposal as a low level radioactive waste.

CONCLUSION

Survey activities consisted of surface wipe samples and survey probe measurements at all locations associated with UAA and UNH material use. No areas were identified that exceeded the contamination criteria approved by NYSDOL in the decommissioning plan.

Removal of these radioactive drummed materials from the manufacturing site and the absence of surface contamination within the facility resulted in the NYSDOL granting decommissioning status to the facility and termination of the NYSDOL radioactive material license.

The successful conclusion of reducing mixed waste to a LLRW and thereby devising a disposal option was accomplished by advising the manufacturer to take the chemical process to completion and coordination with the involved regulatory agencies as well as the potential disposal facility throughout each step of strategy development and plan implementation.

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SODA BLASTING DEMONSTRATION FOR REMOVAL AND TREATMENT OF RADIOACTIVE, ORGANOCHLORINE, AND HEAVY METAL CONTAMINANTS FROM SOLID SURFACES: A DEMONSTRATION AT THE OAK RIDGE K-25 SITE

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ABSTRACT

In 1993, O'Brien & Gere Engineers, Inc. was selected under a Program Research and Development Announcement solicitation of the Morgantown Energy Technology Center of the U.S. Department of Energy (DOE) to develop soda blasting technology for removal of radioactive contamination

from surfaces. The project included demonstration of the technology in decontamination and decommissioning (D&D) operations at a nuclear facility. The contract award and management was through Lockheed Martin Energy Systems of Oak Ridge, Tennessee under the direction of DOS's Oak Ridge Operation Office (ORO). Work was performed at ORO's K-25 plant site and O'Brien & Gere Facilities in Oak Ridge.

O'Brien & Gere developed a scouring system that removes hazardous and fixed radioactive surface contamination and minimizes residual waste. It uses an abrasive sodium bicarbonate medium that is projected with great force at contaminated surfaces. It mechanically removes surface contamination while leaving the surface intact. Blasting residuals are treated using physical/chemical processes.

Bench- and pilot-scale testing of the soda-blasting system was conducted between December 1993 and September 1994 on surfaces contaminated with uranium, technetium, heavy metals, and PCBs. Areas of concrete and metal were blasted. Residuals were dissolved in tap water and treated for radioactive, hazardous, and organochlorine constituents. The treatment comprised pH adjustment, aeration, solids settling, filtration, carbon adsorption, and ion exchange. It produced treated water and residual solid waste.

These tests demonstrated that the system is capable of removing greater than 95% of radioactive and PCB surface contamination to below DOE's unrestricted use release limits. Aqueous radionuclides, heavy metals, and PCBs were below DOE and USEPA treatment objectives for residuals after treatment. Waste residuals volume was decreased by 70%.

Preliminary analysis suggests that this treatment system provides significant waste volume reduction and is more economical than available surface decontamination technologies. Full-scale system capital and operating costs are under development.

The DOE's emphasis in this demonstration procedure has been on comparing the economics and efficiency of the technique against other available and developing technologies.

INTRODUCTION

Increasing waste disposal costs and decreasing storage facility capacity are prompting the DOE and commercial utilities to explore new waste minimizing decommissioning and decontamination (D&D) techniques. Current D&D activities are generally labor intensive, use chemical reagents that are difficult to treat, and may expose workers to hazardous chemicals. Therefore, new technologies are desired that minimize waste, allow much of the decommissioned materials to be reused rather than disposed as waste, and produce wastes that will meet disposal criteria.

To support this D&D objective, the O'Brien & Gere Companies tested a decontamination system on concrete and steel surfaces contaminated with radioactive (238U and 99Tc) and hazardous (PCBs and lead) waste in Oak Ridge, TN. The principal objectives of this on-site soda blasting demonstration project were to evaluate the effectiveness of decontamination by blasting with sodium bicarbonate and to minimize waste volume by dissolving and treating blasting residuals through a wastewater treatment system.

BACKGROUND

Testing of a soda blasting system was conducted at DOE's K-25 former gaseous diffusion plant, Building K-29, in Oak Ridge, Tennessee between December 1993 and September 1994. This gaseous diffusion plant separated 235U from uranium ore for use in atomic weapons and commercial reactors.

The radioactive contamination on surfaces located in K-29 was considered fixed contamination, which is not easily removed by casual contact. Contaminants present on surfaces tested comprise 238U, 99Tc, lead, and PCBs. Uranium was present on concrete and metal surfaces from process operation and equipment releases during enrichment. 238U is an alpha-emitter with a half-life of 7.05 X 108 years. 99Tc is a byproduct of 235U fission in nuclear reactors. 99Tc entered the K-25 process as a volatile impurity in recycled uranium (1). 99Tc is a beta-emitter that has a half-life of 2.12 X 105 years. Lead was present in coatings found on the concrete and metal surfaces at K-29; and PCBs were the result of a heat transfer fluid conduit leak.

For this testing, the following cleanup objectives were established: the total beta/gamma activity and the total alpha activity of the material each were to be less than 5000 disintegrations per minute/100 cm2 (dpm). These criteria were obtained from DOE Order 5400.5 for unrestricted use. Waste volume reduction (soda blasting residuals treatment) objectives were developed as benchmarks to evaluate on-site testing results. These treatment objectives were developed based on DOE 5400.5 standards and federal limits under the requirement of the Safe Drinking Water Act (SDWA). Aqueous phase treatment objectives for total U, 99Tc, PCB, and lead were 12 mg/l, 100 pCi/L, 0.5 mg/l, and 15 mg/l, respectively.

SODA BLASTING METHODOLOGY

Soda blasting utilizes sodium bicarbonate media to physically remove contaminants from surfaces. Compressed air propels sodium bicarbonate at surfaces, which removes contaminants and surficial coatings. The physical and chemical characteristics (non-destructive, non-toxic, and water soluble) of sodium bicarbonate render it a desirable blasting media. The crystalline structure of sodium bicarbonate is aggressive enough to remove contaminants and coatings from metal and concrete surfaces while not degrading blasted surfaces. Sodium bicarbonate media will not introduce additional hazardous or toxic chemicals during operation. Waste volume may be minimized by sodium bicarbonate dissolution and contaminant removal.

Areas of concrete floors and columns and steel and aluminum surfaces were selected, based on radiological surveying and PCB wipe testing to evaluate the soda blasting process. Blasting tests were verified by post-blast analysis using direct reading instruments and smear and wipe samples.

On-site testing evaluated six operating variables: air pressure, water pressure, nozzle orifice diameter, nozzle orifice design, media type, and media flow rate. Ten 1-m2 concrete surface grids were used to evaluate these operating parameters. Blasting parameters were evaluated on concrete surfaces, because typically, if removal objectives are achieved on concrete, they will be equal or better on less porous surfaces such as metal. Once blasting parameters were determined, three concrete grids and several metal object surfaces were blasted to evaluate decontamination efficiency.

Subsequent to each blasting test, a tap water wash was utilized to remove blast residuals from the tested surfaces. Each test surface was triple rinsed using between 2 and 15 gal of water for each test grid. Between each rinse, blasting residuals slurry (water and spent sodium bicarbonate) was collected using a small wet/dry vacuum. The blasting residuals slurry was transferred to 55 gal drums in preparation for blasting residuals treatment.

WASTE VOLUME REDUCTION METHODOLOGY

The aqueous phase solubility of sodium bicarbonate provides a mechanism for separating blast media from contaminants following blasting. Sodium bicarbonate has a solubility of 96 g/l at 20oC (2). By mixing water with spent blasting media, sodium bicarbonate dissolves which physically separates contaminants from the blast media. This reduces waste volume. Soda blasting residuals solution contains elevated concentrations of dissolved sodium bicarbonate. Therefore carbonate complexes control the stability of uranium and other metals in solution. The dissolved solids also interfere with ion exchange processes. Treatment processes developed for this system were based on physical/chemical mechanisms controlling the fate of contaminants in the presence of a high ionic strength wastewater system. The treatment system comprised pH adjustment, chemical precipitation, solids removal, carbon adsorption, and ion exchange. Figure 1 presents a schematic of the blasting residuals waste volume reduction system. Based on bench and pilot-scale treatability testing, physical/chemical unit operations were selected. Treatment of each of the four contaminants was conducted as follows:

Fig. 1

Uranium, in an elevated carbonate system, will form carbonate complexes such as UO2(CO3)6-8 and UO2(CO3)3-4 (3). However, at low carbonate concentration and high pH, uranyl hydroxides (i.e., (UO2)3(OH)5+) form. The solubility of these hydroxide species varies with pH. At an optimum pH, the hydroxide formation is insoluble and can be chemically precipitated. Conversion of uranyl carbonate to uranium hydroxide is difficult in concentrated bicarbonate solutions.

To achieve effective uranium removal, carbonate was removed. Carbonate removal was accomplished in this study by depressing solution pH, to transform inorganic carbon as carbonate (CO32-) and bicarbonate (HCO3-), to carbonic acid (H2CO3). The reduction of solution pH converted inorganic carbon to dissolved carbon dioxide, which was lost to the atmosphere. This loss of inorganic carbon allowed the formation and subsequent precipitation of uranium hydroxide complexes at alkaline pH. To accomplish this conversion during on-site testing, approximately 30 gal of 70% nitric acid was added to acidify 450 gal of blasting residual from an initial pH value of 9.35 S.U. to a pH value below 5 S.U. Subsequent to acid addition, the blasting residual was aerated for two hours to remove aqueous phase carbon dioxide.

Similar to uranium, at elevated pH values lead forms insoluble metal hydroxide complexes. As with uranium, inorganic carbon removal enhances the conversion of soluble metal carbonate complexes to insoluble metal hydroxide complexes that facilitate lead removal by chemical precipitation. The solution pH was elevated to alkaline conditions (pH=11.5 to 12 S.U.) with sodium hydroxide to facilitate the formation of insoluble uranium and lead hydroxides. Blasting residual was flash mixed (10 minutes), flocculated (30 minutes), and then gravity settled. Subsequent to settling, clarified supernatant was gravity drained to the clarifier holding tank.

The clarified supernatant was pumped from the clarifier holding tank, at 3 to 5 gal/min, through two cartridge filters in series (10 mm and 5 mm pore size, respectively) to remove unsettled particulates, and a 15-gal granular activated carbon (GAC) upflow pressure vessel to remove dissolved PCBs. GAC has a high adsorption affinity for large, non-polar

molecules such as PCBs. Subsequent to U, lead, and PCB removal, the solution was pH adjusted from 11.5 to 6.3-6.8 S.U. using 70% nitric acid. 99Tc removal from neutralized (6.3 to 6.8 S.U.) water was accomplished using ion exchange. In waters in contact with the atmosphere, 99Tc is expected to exist in a +7 oxidation state as the highly soluble pertechnetate anion (TcO_4^-) (4,5). Ion exchange replaces the presaturant ion (typically OH^- or Cl^-) with TcO_4^- on the resin surface. Bostick and coworkers reported successful 99Tc removal using both strong base and cross-linked polyvinylpyridine (PVP) resins (5).

Two commercially available ion exchange resins (DOWEX 21KXLT and Reillex HP) were evaluated for 99Tc removal. A 2 in. diameter, 26 in. deep bed of each resin was utilized. During testing, 50 gal of pretreated blasting residual solution were processed through each column. Liquid residual flow rates were maintained at 1 L/min. Column effluents were sampled at 30-min intervals. Samples were analyzed for technetium and total dissolved solids.

TESTING RESULTS

Surface Decontamination

Table I presents pre- and post- blasting results for mean alpha and mean beta levels on concrete and metal surfaces. Initial contamination levels on the surfaces ranged from 5,000 to 75,000 dpm beta/gamma and 1 to 400 dpm alpha. Removal averaged between 95% and 100% beta/gamma and non-quantifiable to 100% alpha, for surfaces tested using selected blasting parameters. The non-quantifiable percent removals for alpha resulted from initial readings which approached background levels. In each test, the post blast alpha readings were below the release limit of 5000 dpm.

Waste Volume Reduction

The waste volume reduction system demonstrated effective removal of uranium, and heavy metals through chemical precipitation. Table I includes pre- and post- wastewater treatment analytical results of blasting residuals. The system effectively removed >97% of uranium and >99% of lead and PCBs. Ion exchange column testing results demonstrated technetium removal to below the 100 pCi/L treatment objective for both resins. The most effective resin ran for 160 minutes at a flow rate of 1 L/min before exhibiting effluent concentrations that exceeded the treatment objective of 100 pCi/L. Although this resin capacity for technetium removal in a high dissolved solids matrix may be less than its capacity in other less competitive applications, results indicate that it is technically feasible.

CONCLUSIONS

Soda blasting removes fixed radioactive and hazardous surface contamination, while leaving the surface intact. Blasting residuals are dissolved and effectively treated using physical/chemical processes. This system produced treated water meeting stringent water quality criteria and residual solid waste requiring off-site management.

Testing results demonstrated that this soda blasting/waste residuals treatment system provided a 70% reduction in waste volume as compared to blasting without treatment. The system is capable of removing greater than 95% of radioactive contamination, achieving DOE's unrestricted use release limits; U, 99Tc, lead, and PCBs were below DOE and USEPA treatment objectives after blasting residuals treatment. Testing results also suggest that this system is more economical than surface decontamination techniques that are commercially available. Estimated full-scale system capital and operating costs (including waste disposal)

are approximately $7.00 to $ 8.00/ft2 of surface area for 230,000 ft2 of concrete surface area (6). Unit costs may be less for metal surfaces due to lower solids generation rates.

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DECONTAMINATION AND DEMOLITION OF A FORMER PLUTONIUM PROCESSING FACILITY'S PROCESS EXHAUST SYSTEM, FIRESCREEN, AND FILTER PLENUM BUILDINGS

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ABSTRACT

The Los Alamos National Laboratory (LANL) Decommissioning Project has decontaminated, demolished, and decommissioned a process exhaust system, two filter plenum buildings, and a firescreen plenum structure at Technical Area 21 (TA-21). The project began in August 1995 and was completed in January 1996. These high-efficiency particulate air (HEPA) filter plenums and associated ventilation ductwork provided process exhaust to fume hoods and glove boxes in TA-21 Buildings 2 through 5 when these buildings were active plutonium and uranium processing and research facilities. This paper summarizes the history of TA-21 plutonium and uranium processing and research activities and provides a detailed discussion of integrated work process controls, characterize-as-you-go methodology, unique engineering controls, decontamination techniques, demolition methodology, waste minimization, and volume reduction. Also presented in detail are the challenges facing the LANL Decommissioning Project to safely and economically decontaminate and demolish surplus facilities and the unique solutions to tough problems. This paper also shows the effectiveness of the integrated work package concept to control work through all phases.

Keywords: plutonium, work package, health and safety, characterization, filter plenum, engineering controls, decontamination, demolition, transuranic waste, waste minimization, and volume reduction.

INTRODUCTION

Many of the challenges of the TA-21 Filter Building Decommissioning Project are not unique to LANL, and their solutions can be applied to other decommissioning projects and programs elsewhere. The TA-21 Filter Building Decommissioning Project presented safety, personnel exposure, and contamination control challenges that required extra care to ensure that rigorous radiation protection practices were followed by project personnel. The project goals were as follows: 1) the removal of as much plutonium holdup as possible through decontamination and component removal to downgrade from a Category 3 Nuclear Facility rating to a Radiological Facility rating; 2) the removal of all process exhaust systems including 1500 linear ft of ductwork, glove boxes, and hoods from Buildings 3 and 4 North to the firescreen; 3) the decontamination and demolition of the firescreen; 4) the decontamination and removal of the filter plenum and glove boxes from the Rotary Filter Plenum Building (Building 146); 5) the removal and disposal of the HEPA filter bank from the Main Filter House (Building 324); 6) the demolition of the stack; 7) the free release of all remaining building walls, ceilings, and cement slab foundations; and finally 8) classification of most of the radioactively contaminated demolition debris as low-level radioactive waste (LLRW) rather than transuranic waste through decontamination. Because of the existing 239Pu holdup (approximately 1 mCi/ft) the process exhaust system, which includes the firescreen and filter plenums, was regarded as a Category 3 Nuclear Facility. Paramount to the success of the project, the downgrading from the Category 3 Nuclear Facility rating to a Radiological Facility rating was needed at the beginning of the project. This downgrading was accomplished through the initial elimination of 75 to 80 percent of the plutonium holdup through decontamination and component removal (firescreens) from the Firescreen Building (Building 329) and the decontamination of the main filter plenum in Building 146. The decontamination objective was to reduce the plutonium source term below a certain level and not to free release the structures. The subsequent downgrading from a Category 3 Nuclear Facility to a Radiological Facility eliminated much of the initial engineering work (that is, Engineering Analysis, Title I and II Engineering Project Plans) required for a Category 3 Nuclear Facility.

Decontamination methodology is discussed in detail including decontamination equipment, decontamination techniques, decontamination effectiveness, solid and liquid radioactive waste generation, waste minimization techniques, and waste volume reduction.

History of TA-21

DP West began operations in September 1945. Its main purpose was to provide the capability to produce metal and alloys of plutonium from the nitrate solution feedstock provided by other production facilities. This process involved several acid dissolution and chemical precipitation steps to separate the plutonium and other valuable actinides from the feedstocks. A major research objective at DP West was the development of new purification techniques that would increase the efficiency of the separation processes. These separation techniques used a wide range of chemicals from the periodic table. In conjunction with improving purification techniques in the main process lines, research was conducted

into reprocessing the waste produced to further enhance recovery. In addition, other operations, such as nuclear fuel reprocessing, were performed occasionally at DP West. Activities unrelated to plutonium processing also occurred at DP West (Fig. 1).

Fig. 1

The main plutonium purification processes were contained in Buildings 2, 3, 4, and 5 and later in Building 150. Uranium and plutonium metal produced in these buildings was secured and stored in Building 21, the old vault. Research into methods of recovering additional plutonium from waste streams was conducted in Building 33. Additional research on the properties and uses of plutonium was conducted at Building 210, the plutonium research building.

In 1977 a transfer of work to the new plutonium facility at TA-55 began, and much of the DP West complex was vacated. At the time, cleanup of the old process lines was initiated. This cleanup included removing contaminated equipment and material from Buildings 2, 5, and 150 and from parts of Buildings 3 and 4. The buildings were then remodeled for use by other groups at LANL.

Filter Buildings

The filter buildings provided process exhaust to Buildings 2, 3, 4, 5, and 21 at TA-21. The process exhaust filter system consisted of the following: the Firescreen Building (Building 329); the Rotary Filter Plenum Building (Building 146); the Main Filter House (Building 324), and the Main Stack.

Ductwork exited Buildings 3 and 4 North and ran along elevated stanchions until it reached the firescreen. The exhaust stream entered this structure, which was an elevated, sheet metal enclosed building containing screen filters and washdown equipment. A transparent glass line exited the sheet metal enclosure and discharged into a liquid waste transfer line, which ran to the on-site liquid waste treatment plant. The exhaust then entered Building 146, a concrete block building that housed a large, circular HEPA filter array and a glove box assembly for changing out the filters (Fig. 2). The HEPA filter array consisted of an octagonal filter bank containing eight sets of three filters housed in a drum. The drum assembly rotated so that new filter faces could be presented to the airstream, thus reducing by a factor of eight the downtime needed for change out. The exhaust stream then entered Building 324, the filter house, which was added to the flow path in 1973. It contained 20 HEPA filters in parallel. Exhaust was then released through the stack at the north end of the building.

Fig. 2

Decommissioning of the filter buildings involved the removal of hoods, glove boxes, and interior process exhaust ductwork from Buildings 3 and 4 North; the elevated ductwork that ran into Building 146; the HEPA filters and glove box and drum assemblies in Building 146; the firescreen, all ductwork, and the stack in Building 146; the HEPA filters in Building 324; and all ductwork and the stack in Building 324. Both buildings were then demolished.

LANL was responsible for overall project management, health physics, environmental compliance, criticality engineering, and waste management. Subcontractor oversight in the areas of engineering and health and safety also were performed by LANL. Dismantlement and demolition activities were performed by the on-site maintenance subcontractor, Johnson Controls

World Services, Inc., who also provided industrial hygiene services and was instrumental in developing work packages.

INTEGRATED WORK PROCESS CONTROLS

A key element to the success of the project was the application of the integrated work process control called the work package. Work packages typically included a specific task work procedure, a Task Hazard Analysis (THA), a Radiological Work Permit (RWP), and an ALARA Job Review, if required.

A THA was developed for each specific task and was an assessment of all nonradiological workplace hazards. The THA along with the RWP was the basis for developing work procedures and documenting the need for special permits and controls. The THA was signed by each employee who worked on the task, including supervisory personnel, and generally included the following:

- general information including historical sampling data related to the task;
- task description including procedures required to minimize hazards;
- descriptions of specific hazards;
- hazard control measures including personal protective equipment (PPE), permits, and training;
- any special decontamination procedures not covered by the RWP (for example, chemical decontamination); and
- spill prevention, containment, and response and/or accident mitigation.

As part of the work package, all decommissioning work that had a potential for personnel internal or external radiation exposure and/or contamination spread required an RWP. The RWP placed controls on personnel entry into controlled and radiological areas. The RWP identified the specific work activity, evaluated potential radiological exposure conditions, and established appropriate levels of radiological control technician job coverage, monitoring instructions, action levels and hold points, PPE, radiological controls for demolition, and dosimetry assignment for entry.

Work packages were typically developed within days of the actual work by the site superintendent, construction supervisor, lead radiological control technician, and other key health and safety personnel. This process provided project personnel a usable work plan, which included a detailed task procedure, a work evolution hazard assessment, personnel protection based on the hazard assessment (confined space permits, burning/welding permits, engineering controls, respiratory protection, PPE, and dosimetry), and all contamination controls. One key benefit from this approach is that the work package was developed in real time. Therefore, recent and pertinent survey data, lessons learned, and personnel experience obtained from preceding job evolutions were continuously incorporated into new work packages.

CHARACTERIZE-AS-YOU-GO METHODOLOGY

Characterization of the entire facility was not conducted. Instead, LANL uses a characterize-as-you-go methodology for decommissioning projects. Rather than extensively characterizing the entire project, enough data are collected early in the project through surveys, historical documentation search, and interviews conducted with individuals who have historical knowledge of the site. Types of important information include the specific processes conducted at the site, chemicals and radionuclides used in the various processes, and locations of any spills and releases. Detailed work procedures are developed as the work progresses, and

additional information is collected as necessary. This process avoids efforts that can be rendered useless by newly discovered problems, but it requires flexibility in scheduling and completing activities. This section discusses the application of this approach to the filter buildings.

Important to the characterize-as-you-go methodology is the detailed project characterization directory developed and revised as the project progresses. The characterization directory is a living document that includes digital photographs of key areas, rooms and system components to be decontaminated and/or decommissioned, diagrams, any historical information on the key system or component, survey data points, and any other pertinent information. This directory is updated continuously as information is made available. Key uses of the directory are to write the work packages, conduct prejob briefings, and orient new project personnel.

Engineering data requirements consisted of utility and structural information. Specifically, the locations of all utilities and any necessary reroutes must be identified. Structurally, the characterization effort had to ascertain whether the Building 146 drum assembly would rotate. The drum had not been turned since the 1970s, and seal integrity and the opposite filter banks were items of concern. Existing drawings were collected for reference and were annotated to identify the as-left facility condition. Historical records were reviewed to identify any abandoned utilities and any facility modifications that could affect decommissioning.

Knowledge regarding types and quantities of contaminants is essential for decommissioning operations and waste handling. Radioactive waste may be either low-level or transuranic, whereas chemical contamination may result in hazardous or mixed waste. Potential contaminants were identified from the remedial action work plan, operating summaries, decommissioning summaries, and historical interviews. Radionuclides of concern were 235U, 238U, 238Pu, 239Pu, 99Tc, 241Am, 243Am, 237Np, 232Th, and 231Pa. Chemical concerns included asbestos (146 HEPA filters), metals (146 HEPA filters), polychlorinated biphenyls (PCBs) (lighting ballasts), perchlorates (ductwork, 146 filters), and picric acid.

Because the data address waste management and safety concerns, exact readings were not as important as bounding readings. The data should identify thresholds for waste categories or PPE requirements.

ENGINEERING CONTROLS

Unique engineering controls developed for the project were modifications to the process exhaust system and the compartmentalized support tent with its attached "body glove" glove bag. Other engineering controls used during the project included standard glove bags, HEPA filtration methodology (both portable HEPA filtration units and the reliance on the existing process exhaust system), and strippable coatings.

Before beginning any major decommissioning activities, modifications to the process exhaust were necessary primarily because of considerable system negative pressure. Before any modifications, the system negative pressure was approximately 3 in. of water, too much to allow its use during decommissioning. Dampening was possible by cutting a 36-in. hole in the process exhaust downstream from the two stages of system HEPA filters directly below where it entered the stack. A cylindrical 38-in. long by 36-in. diameter sheet metal piece was then welded to the hole with a circular plate attached to provide the dampening (Fig. 3). System

negative pressure was adjustable from 0.2 to 3 in. of water with this modification. The negative pressure was adjusted to suit the task being performed.

Fig. 3

Because of significant plutonium holdup in the entire process exhaust system, the reliance on engineering controls to reduce this hazard was a LANL Health Physics group requirement. Data made available during the initial characterization of the firescreen, verified through surveys and air sampling, indicated the average surface plutonium contamination at >4.0E+06 dpm/100 cm2 removable and airborne contamination levels up to 1500 derived air concentration (DAC) -hours. One significant engineering control developed specifically for the project and used with great success was the body glove. The body glove with its attached support tent (Fig. 4) provided maximum contamination control and worker protection. The support tent was compartmentalized for maximum contamination control in the event of a body glove failure. All negative ventilation was provided by the existing process exhaust with portable HEPA units attached to the support tent as backups. The body glove is essentially a glove bag that personnel enter to perform work; whereas, a normal glove bag surrounds a highly contaminated item within the bag, and personnel work from the outside. Before erecting the body glove, all necessary tools and equipment for a particular task were introduced into the firescreen. Then the body glove was inserted directly into the firescreen, unfolded, and supported by a rigid metal internal frame. Work was performed inside the bag using a series of gloves positioned on the sides and top of the body glove.

In highly contaminated areas, such as the firescreen and main filter plenum, the body glove isolated workers from both seriously high surface and airborne contamination (Fig. 4). Airborne contamination levels were reduced from the initial 1500 DAC-hours to <1 DAC-hour, which allowed most work to be performed using supplied-air respirators that were required in the event of a body glove failure.

Standard glove bags were used throughout the project. All demolition and size reduction of overhead process exhaust ductwork was done using glove bags, a skill developed during the demolition of Buildings 3 and 4 South. When the interior process exhaust system was removed, Buildings 3 and 4 North were active facilities, and extensive use of glove bags prevented release of radioactive contamination and avoided costly cleanup efforts.

Fig. 4

DECONTAMINATION AND DEMOLITION METHODOLOGY

The objective of the TA-21 Filter Building Decommissioning Project was to reduce the plutonium contamination on surfaces below transuranic levels. If possible, metal surfaces were to be decontaminated further to meet Science and Ecology Group (SEG) waste classification guidelines to enable the metal to be recycled at their facility in Oak Ridge, Tennessee. SEG is a large recycler for radioactive-contaminated metal that deals mainly with the commercial sector. It has been used by LANL for less than one year. It was possible to recycle all plenum walls and ceilings, but floor surfaces were sent to LANL's LLRW landfill at TA-54. Project surface contamination acceptance criteria for LLRW and transuranic waste and SEG waste acceptance criteria are found in Table I. Ninety percent of all radioactive waste for the project was characterized as LLRW. Twenty percent of this material was shipped to SEG. Equipment was either decontaminated in situ or brought to the project decontamination area, an

old hot cell in Building 4 North. Sheet plastic was fastened to the floor, walls, and ceiling with duct tape, and two 1800 cfm HEPA-filtered negative air units were added to mitigate large amounts of surface and airborne plutonium contamination. All decontamination and size reduction of material was performed in glove bags and under negative pressure. Airborne contamination levels reached approximately 2000 DAC-hours during a certain decontamination operation that resulted from a glove bag failure. Typical airborne radionuclide concentrations inside the hot cell ranged from 3 to 50 DAC-hrs. All work was performed in air purified respirators.

Table I

The project relied on the following five proven methods of decontamination: vacuuming, wiping, scrubbing, using strippable coatings, and shot blasting. Vacuuming, wiping, scrubbing, and strippable coatings were primarily used to decontaminate the firescreen plenum, the main filter plenum, glove boxes, and ductwork. Shot blasting was used to decontaminate concrete surfaces especially the concrete slab in Building 146. Vacuuming, wiping, and scrubbing were used to remove radioactive dust and particles from plenum surfaces. Vacuuming was performed using HEPA-filtered vacuum cleaners. Surfaces were then wiped/scrubbed with a damp rag and an industrial all-purpose cleaner. Rags were discarded as radioactive waste.

After surfaces were vacuumed and wiped down, strippable coatings were applied. Decontamination factors ranged from 10 to 100 depending on the presence of grease or oily residue on surfaces. The use of strippable coatings involves the application of a polymer mixture, either by a paint roller or airless sprayer, to a contaminated surface. Both application methods were used in this project. As the polymer hardens, the contaminants are entrained into the material. The coating is then pealed off, containerized, and disposed of. This technique is best suited for floors, walls, and ceilings because of their easy accessibility. Strippable coatings were also used with limited success on internal glove box and ductwork surfaces.

Shot blasting was used on the concrete slab on Buildings 146 and 324 after all equipment was removed from the buildings and the ceilings and walls were removed. Most of both building structures were free released and sent to a local sanitary landfill for disposal. Shot blasting is an airless method that strips, cleans, and etches the surface simultaneously. The technique is virtually dust free; therefore, shot blasting of the concrete slabs was conducted without using respirators. Portable shot blasting units move along the surface as the abrasive is fed into the center of a completely enclosed centrifugal blast wheel. As the wheel spins, the abrasives are hurled from the blades, blasting the surface. The abrasive and removed debris are bounced back to a separation system that recycles the abrasives and sends the contaminates to a dust collector.

Demolition methodologies followed current, accepted industry practices. The general decommissioning sequence consisted of 1) HEPA filter removal from the main rotary plenum and from Building 324, 2) main filter plenum removal from Building 146, 3) hood and glove box removal, 4) exhaust system removal, 5) utility piping removal, 6) final system disconnects (that is, electrical and fire protection), and 7) a final status survey of both buildings to determine their suitability for free release. After additional spot decontamination of masonry block wall surfaces, the

buildings were demolished using a trackhoe. Finally, both buildings'concrete slabs were decontaminated by shot blast, surveyed for free release, and then removed using a trackhoe.

WASTE MINIMIZATION AND VOLUME REDUCTION

LANL and Department of Energy policy precludes the free release of any material with detectable activity above background levels, even when the surface contamination is below release guidelines. Although some materials have been released to a municipal landfill following demonstration of no detectable activity, waste minimization activities primarily emphasize volume reduction through on-site waste compaction and recycling of contaminated scrap metal. Concrete was cleaned using a shot vacuum system, and the remaining slabs will be crushed and used as on-site fill. Through recycling, steel decontamination, and concrete crushing, LLRW from decommissioning was reduced compared with previous decommissioning projects.

Soil remediation was coordinated with LANL's remedial action project. Sampling and other activities also were coordinated to ensure data applicability and cost effectiveness.

A significant amount of data currently exist for this project.1 This information was obtained during a LANL-wide project to quantify special nuclear material holdup in ventilation systems. These data indicated that sizable portions of the process exhaust would be classified as transuranic waste. Ductwork was decontaminated during decommissioning to minimize the volume of transuranic waste. Accordingly, during decommissioning the removed ductwork and decontamination waste were characterized for waste disposal purposes. This approach also recognizes the difficulty and expense of sampling exhaust systems before removal. Likewise, HEPA filter sampling was best left until actual removal, at which time the filters were cored and samples were obtained more easily. Additional data were collected to measure radioactivity in systems not addressed during previous holdup measurement campaigns. Measurements were made using nondestructive assay methods with sodium iodide and germanium detectors. Items likely to be free of contamination were surveyed to verify that no unexpected radioactivity was present. Appropriate engineering controls were used during decommissioning to protect uncontaminated materials.

Except for one small spot of contamination on the floor of Building 146, no historical releases occurred within either Building 146 or 324. The walls and floors were surveyed before demolition and were decontaminated if contamination above detectable limits was indicated. The long-range alpha detector, an experimental system developed at LANL,2 and conventional gas-proportional instruments were used to systematically survey the structures to verify that the material was uncontaminated.

Facility processes did not involve hazardous wastes listed under RCRA. The RCRA facility investigation work plan identified metals as a potential contaminant of concern, so the Building 146 filters were sampled for metals. Sampling for metals, like the surveys for radioactive constituents mentioned above, were performed when the filter was removed. Building 146 was sampled for perchlorates. This sampling was repeated after the drum had been turned. Historical records suggested that picric acid was used for some experiments. Building 146 was tested for picric acid before and after turning the drum, and the result was negative. During disassembly, duct systems were routinely tested for perchlorates and were all found to be negative.

The HEPA filters contained asbestos, and the roofs of both buildings were thought to contain nonfriable asbestos-contaminated material. All roofing material was tested for asbestos. Lighting systems were inspected for PCBs during disassembly, and fluorescent bulbs were handled as hazardous waste.

LESSONS LEARNED

An important lesson learned is that the observational approach is very effective from both cost and schedule perspectives. By minimizing characterization activities, initial expenses and time to completion are reduced. Moreover, involving the people who will be doing the physical work during the planning stage simplifies the techniques used and guarantees the feasibility of the chosen techniques. Perchlorate and other unusual chemical contaminants (such as picrates) may be hazards in old chemical processing facilities and should be sampled for.

Finally, an extremely important lesson learned is that a small, autonomous project team, capable of internal decision-making, is essential for staying on track. The customer must be part of the team.

ACKNOWLEDGMENTS

This work is funded by the U.S. Department of Energy and is being carried out under the LANL Environmental Restoration Project.

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DEVELOPMENT OF A MODEL DECOMMISSIONING/INTERIM MEASURES PLAN FOR AN INDUSTRIAL FACILITY: A CASE STUDY

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ABSTRACT

Many industrial facilities are faced with remediation of contaminated environmental media. For these facilities, workplans may be required by more than one regulatory agency prior to beginning clean-up activities. For sites regulated by multiple agencies, the development of multiple documents to satisfy all regulatory authorities may be a lengthy, expensive, and difficult process.

A nuclear materials fabrication facility located in East Tennessee is regulated by, among other agencies, the Environmental Protection Agency (EPA), the Nuclear Regulatory Commission (NRC), the State of Tennessee Division of Solid Waste Management (DSWM), and the State of Tennessee Division of Radiological Health (DRH). As presented in this case study, a single remediation plan was developed to satisfy the requirements of each regulatory agency. An overview of the development of the multi-purpose

remediation plan, a discussion of the benefits and limitations of the multi-purpose approach, and applicability of the multi-purpose approach to other industrial facilities is presented.

INTRODUCTION

As required by the Nuclear Regulatory Commission (NRC), Nuclear Fuel Services, Inc. (NFS), a nuclear materials fabrication facility located in East Tennessee, is currently conducting decommissioning activities. These activities are also regulated by the Environmental Protection Agency (EPA) and the State of Tennessee. Traditionally, prior to performing remediation activities, a facility would be challenged with the development of multiple remediation workplans to satisfy requirements of all respective regulatory authorities. This case-study describes an alternative approach in which NFS successfully developed a single, multi-purpose environmental remediation plan which encompassed the requirements of the EPA, the NRC, and the State of Tennessee.

Background

NFS, which began its Tennessee operations in 1957, processes radioactive materials under a License from the NRC. The company's first contracts were to manufacture a variety of nuclear materials including uranium and thorium metals and various compounds and alloys of these materials. Prior to 1966, land disposal of low-level radioactive and solid waste materials occurred in a section of the NFS property. Land disposal of these materials was an accepted disposal method at that time. The disposed material was ultimately covered, and the area became commonly referred to as "Pond 4." In 1990, Pond 4 was identified as a Solid Waste Management Unit (SWMU) by the EPA. Prior to NFS remediating Pond 4, EPA, NRC, and State of Tennessee regulatory authorities requested a remediation plan for the area. Regulatory review and approval of the Pond 4 remediation plan was requested by the agencies before NFS began remediation activities.

REMEDIATION PLAN DEVELOPMENT REQUIREMENTS

Facilities operating under State and EPA permits and State and/or NRC licenses may need the approval of each agency prior to remediating contamination attributable to solid and/or hazardous waste and radiological materials. Decommissioning for NRC regulated activities must be performed in accordance with an approved decommissioning plan. Similarly, environmental remediation activities that are (1) required by EPA due to the presence of contamination determined to be potentially dangerous to public health and/or the environment; or, (2) initiated by the permitted facility, may be performed in accordance with an approved interim measures plan.

NRC requirements for the development of decommissioning plans are included in Title 10 of the Code of Federal Regulations (CFR) and in an NRC draft regulatory guide. The CFR regulations applicable to NFS are located in Parts 30, 40 and 70. These parts are entitled "Rules of General Applicability to Domestic Licensing of Byproduct Material," "Domestic Licensing of Source Material," and "Domestic Licensing of Special Nuclear Material," respectively (1).

The applicable regulations contained in 10 CFR addressing facility decommissioning and the development of a decommissioning plan are summarized in a regulatory guide developed by the NRC entitled "Standard Format and Content of Decommissioning Plans for Licensees under 10 CFR Parts 30, 40, and 70" (2). This document provides a suggested format and describes the suggested content of a decommissioning plan. Specific

requirements for each section of a decommissioning plan are identified in the regulatory guide and are as follows: 1) general information; 2) description of decommissioning activities; 3) description of methods used for protection of occupational and public health and safety; 4) final radiation survey; 5) decommissioning funding; and, 6) facility and radioactive material security and safeguards.

EPA requirements for the development of an interim measures plan are contained in Title 40 of the CFR and in an EPA developed guidance document. The applicable CFR Regulations are located in Part 264, which is entitled "Standards for Owners and Operators of Hazardous Waste Treatment, Storage, and Disposal Facilities" (3). Subpart F of 40 CFR, Part 264, which is entitled "Releases from Solid Waste Management Units," contains information related to interim measures requirements.

The regulations contained in 40 CFR relative to facility interim measures and the development of an interim measures plan are included in an EPA guidance document entitled "RCRA Corrective Action Interim Measures Guidance" (4). A facilities' HSWA permit also contains requirements for development of the interim measures plan. Although the EPA guidance document (4) and the NFS HSWA permit include information relative to the required content of an interim measures plan, information relative to a suggested format is not provided.

A third EPA document, entitled "EPA Handbook: Stabilization Technologies for RCRA Corrective Actions," (5) was recommended by the EPA as a reference related to interim measures. This document does not contain requirements for the development of an interim measures plan. It does, however, contain information relative to interim measures technologies recognized by the agency.

REMEDIATION PLAN DEVELOPMENT STRATEGY

The NFS strategy for development of the remediation workplan for Pond 4 consisted of eight main steps as depicted in Fig. 1. These steps were 1) identification of regulatory requirements; 2) preliminary outline development; 3) presentation to regulatory officials; 4) outline revision; 5) assembly of multidisciplinary project team; 6) remediation plan development; 7) plan review/revision; and, 8) regulatory approval.

A primary objective of the NFS strategy was to optimize communication with regulatory officials both in the initial planning stages and throughout plan development. Regulatory contacts were initially briefed regarding the potential for the development of a single remediation workplan. At this point, the State officials indicated agreement with the NFS approach and advised following federal EPA/NRC remediation plan development requirements. EPA and NRC officials agreed with the preliminary approach presented by NFS.

Once development of the decommissioning/interim measures outline was completed by NFS, formal meetings were conducted individually with EPA and NRC contacts. The outline was approved by the NRC without revision and minor revision was requested by the EPA. The completed preliminary outline for the Pond 4 Decommissioning/Interim Measures Plan is presented as Fig. 2.

A secondary objective of the NFS strategy was to utilize a multidisciplinary project team approach. The multidisciplinary approach was selected by NFS management due to the technical nature of the subject material required for development of the Decommissioning/ Interim Measures Plan. The project team assembled by NFS consisted of a Hydrogeologist, a Health Physicist, a Waste Management Specialist, an

Engineer, and an Environmental Scientist. The Environmental Scientist was assigned responsibility for coordinating the team and development of the plan. A project schedule was developed, using Timeline project management software (Fig. 3), which allowed for 60 working days (i.e., approximately three months) to develop the Decommissioning/Interim Measures Plan for submittal to the EPA, the NRC, and the State of Tennessee.

Fig. 1

Fig. 2

Fig. 3

Once the initial draft of the document was completed, the review process began. The plan was scheduled to receive three "in-house" reviews by NFS: peer review, project management review, and senior management review and approval. These reviews were scheduled to ensure the document was complete, accurate, and of the quality suitable for submittal to the regulatory agencies for review and approval. Document revision was the responsibility of the Environmental Scientist; however, some revisions required the collaborative efforts of the project team. Resolution of some technical issues required meetings between the project team and management personnel. These meetings informed the managers of the technical issues and involved management in the resolutions. Once approved by the senior managers, the document was ready for submittal to regulatory agencies.

Regulatory submittal of the document by NFS was approximately three weeks behind the project schedule due to additional time required for in-house document revision. The document was approved by NFS senior management and was submitted to the NRC, the EPA, and the State of Tennessee for review and approval on December 7, 1993.

The NFS Pond 4 Decommissioning/Interim Measures Plan was jointly approved by the EPA and the State of Tennessee Office of Solid Waste Management on March 22, 1994. Minor comments to the plan were received from the NRC and the State of Tennessee Division of Radiological Health in early May, 1994. The NRC and State comments indicated the remediation strategy and approach proposed by NFS were acceptable; however, minor revision to the plan was required.

Prior to revising the plan, NFS verbally communicated the planned revisions to the NRC and to the State of Tennessee. The plan was subsequently revised by NFS and Revision 1 of the plan was submitted to the EPA, the NRC, and the State of Tennessee on June 16, 1994. The plan revisions primarily consisted of providing clarification of radiological health and safety issues. Approval of the plan by the NRC and the State was received on June 23, 1994.

DISCUSSION

The strategy implemented by NFS for the development of a single Decommissioning/ Interim Measures Plan for Pond 4 was successful as demonstrated by approval of the plan by all respective regulatory agencies, on schedule and within budget. The single document approach facilitated the regulatory review process, lessened the duplication of effort required in the development of two or more separate but similar plans, and was more time and cost efficient. Timely approval of the plan may not have been a reality had NFS not maintained open and frequent communication with the regulatory authorities and if the agencies had each been reviewing different plans.

Development of the plan addressed complex technical issues including engineering, health physics, hydrogeology, waste management, and

environmental science. Utilization of a multidisciplinary project team approach ensured the technical depth and accuracy of the document, thus providing the essential information for regulatory review and approval. Open communication was always maintained between NFS and the regulatory authorities. This communication began in the initial planning stages and continued throughout the completion of the plan. Involving the agencies in the initial planning stages helped to ensure that their approval would be received prior to expending the time and effort required to develop the comprehensive plan. Development of the single plan facilitated the open communication between NFS and the respective agencies as well as streamlined the regulatory review and approval process.

While the multi-purpose approach was successful for NFS, the potential for changes in various agency regulatory requirements during plan development are of concern due to the dynamic nature of the regulatory system. New regulations are constantly being proposed and promulgated by the EPA and the NRC. Changes in regulations within a particular agency, when encountered during the plan development or agency review and approval stages, may delay the approval of the plan, thereby delaying the start-up of remediation work. However, maintaining frequent and open communication with the regulatory authorities may enable the regulated party to be informed of potential changes in regulations before they occur.

Currently, approximately 250 industrial facilities exist, which like NFS, are regulated by the NRC, the EPA, and the State. As demonstrated in this case study, the "multi-purpose" approach utilized by NFS may also be an effective strategy for other facilities planning environmental remediation activities which require approval or concurrence by the EPA, the NRC, and the State. Similarly, it may also be prudent for facilities to apply the multi-purpose strategy to other overlapping EPA/NRC regulated activities; an example would be the development of a single workplan for an EPA RCRA Facility Investigation/NRC Site Characterization.

Note: The opinions expressed by the authors of this paper are their own and do not necessarily reflect the opinions or position of Nuclear Fuel Services, Inc.

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EQUIPMENT DECONTAMINATION THE BENEFITS OF REUSE AND AVOIDED DISPOSAL USING THE TECHXTRACT PROCESS

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ABSTRACT

The need for and benefits of effective and economical decontamination technologies for nuclear facility decommissioning are substantial and without question. Decontamination and decommissioning (D&D) programs in the U.S. Department of Energy (DOE), nuclear utility industry, and international markets are growing rapidly. With this growth, the acceptability and costs of traditional D&D approaches, which rely largely on demolition and disposal, are being increasingly called into question. Disposal in particular is a major complicating factor as disposal costs are rising, and the long-term availability/public acceptability of disposal capacity is a significant uncertainty for the industry.

Thus, the stage is set for innovative technologies which can decontaminate structures and equipment to below regulatory levels, thereby avoiding large-scale bulk disposal. Such technologies must be effective in the removal of a variety of contaminants from different types of materials and substrates. They must also offer total economics (considering process costs, disposal, reuse, future liabilities, and other factors) which are attractive versus demolition and disposal.

The TECHXTRACT process is a chemical technology for the extraction of radionuclides, heavy metals, PCBs, and hazardous organics from solid materials such as concrete and metal. Specifically designed chemical blends are used to penetrate below the surface and remove contaminants which have leached into the substrate. The technology has very high decontamination rates (e.g., 90-99% per cycle). In addition to decontamination performance, economic analysis indicates that TECHXTRACT can save 25-75% of total project cost versus traditional methods. It is a full-scale, commercialized technology which has been used in the DOE complex, electric utility, and private industry. This paper discusses the technology in more detail and offers a case history from a recent full scale decontamination project for a private industry manufacturer.

DESCRIPTION OF THE TECHXTRACT TECHNOLOGY

The TECHXTRACT technology is a patented, sequential chemical extraction process for the removal of radionuclides, PCBs, and other hazardous organic and inorganic substances from solid materials such as concrete, brick, steel, and exotic metals. The technology uses advanced chemical formulations and carefully engineered applications to achieve significant penetration and removal of these contaminants from below the surface of these materials.

The chemistry is based on several hypotheses relating to contaminant migration and removal. The first hypothesis is that contaminants migrate along the grain boundaries and into pores and microscopic voids (Kirkendall effect) of a material, even for seemingly non-porous media. Mobility of the contaminants, time, and secondary forces often drive these contaminants to deeper levels in the substrate. Furthermore, contaminants tend to become chemically or electrostatically bonded to the substrate. In many cases, the time between the contamination event and decontamination efforts will allow the contaminant migration pathways to become partially closed by organic and inorganic impurities.

All of these factors point to the need for a sophisticated chemical system which:

Reopens the pores and capillary pathways to the maximum possible extent (electrocapillary effect),

Penetrates into the pores as deeply as possible,

Breaks the electrostatic and chemical bonds which hold the contaminants in place,

Complexes or sequesters the contaminants to prevent recontamination,

Activates the capillary rise effect (Young and Laplace equation).

The TECHXTRACT chemical formulas are designed to address each of these complex needs, using over 25 components to incorporate dissolution, oxidation, reduction, hydrolysis, decomposition, wetting, complexation, microencapsulation, and flotation chemistry principles. The chemistry further compensates for situations in which the contamination is a mixture of pure elements, oxides, and related compounds with varying solubility indices. The spent chemical solutions, without contaminants, do not contain any TCLP constituents and have been disposed of by incineration, solidification (and land disposal), and discharge to liquid effluent treatment systems.

The technology incorporates a tailored process for applying and removing each of the chemicals in the right sequence and combinations to achieve optimal results. In most projects, three different chemical formulas are used. Chemicals are applied in low volumes, usually as a spray, to minimize consumption and secondary waste volume. After being applied, the chemicals are scrubbed into the contaminated surfaces, left to dwell for a defined time, and rinsed and removed. The application and removal of all three formulas constitutes one complete process cycle, and typically requires one day (24 hours). Sampling and/or surveys can be performed at the end of any cycle, and often shows reductions of 90% or more per cycle.

The TECHXTRACT process has been found to be most applicable in remediation or decontamination projects when one or more of the following conditions apply:

The acceptable level for any residual contaminant is very low (e.g., free release at 5,000 dpm/100 cm2 or lower),

Simple surface cleaning is ineffective,

Disposal is undesirable, either because the volume and resulting disposal and replacement costs are too high, or due to resource recycle or waste minimization objectives,

Significant safety concerns - such as flammability, corrosivity, creation of airborne contaminant particles, fugitive emissions or generation of toxic fumes and/or explosive gases - are raised,

Decontamination is to be performed on surfaces that are not flat and horizontal, such as equipment, walls, ceilings, structural beams, and internal piping,

For radioactively-contaminated metals, the technology is particularly applicable in situations involving equipment which is needed for ongoing operations or for return to a vendor, when there is a high salvage ("as is") value, when the scrap value is much higher if unrestricted release is achieved, or when other factors make metal melting (and restricted use) impracticable or unacceptable. For radioactively-contaminated concrete, it offers the potential to avoid the high costs associated with demolition, repair, replacement, and disposal of large waste volumes.

CASE HISTORY - EQUIPMENT DECONTAMINATION PROJECT

EET and SEG were recently contracted to perform a turnkey radiological decontamination project for a private industry manufacturer. The customer's plant produces filaments for light bulbs. In the past, the raw materials for their process had included small amounts of radioactive

thorium. Routine operations had resulted in widespread contamination, including the major manufacturing equipment, ancillary piping, and the floors and walls of the surrounding rooms.

Due to a change in operations, radioactive materials were no longer being handled at the facility. The facility's objective was to eliminate all radioactive contamination, with minimal impact to ongoing operations. The area of greatest concern were the large production machines which house the treating bottles and which are central to the manufacturing process. These machines had the highest levels of contamination, were critical to ongoing operations, and would be extremely expensive to replace and to dispose of as low level radioactive waste.

The major portion of the decontamination activities were scheduled to be performed during a two-week plant turnaround in the July, 1995. A successful demonstration of the TECHXTRACT technology was performed on-site by EET in May, 1995. This demonstration proved the effectiveness of the technology on typical machinery components. It allowed the plant to include full decontamination in its base line decontamination plans and to avoid ordering replacement equipment, much of which would have required long lead times and custom fabrication.

The production machines and treating bottle racks are complex pieces of equipment. Each treating bottle rack is approximately ten (10) feet long and fifteen (15) feet high and contains four (4) treating bottles. Each treating bottle is approximately eight (8) feet high, with an aluminum base and a double-walled copper bell that fits over the metal bars as they are being treated. This presented a number of difficult to access areas for decontamination.

Radiation levels on the equipment varied greatly. Initial levels as high as 200,000 dpm/100 cm2 (fixed alpha) were observed, with typical levels around 5,000 dpm/100 cm2 (fixed alpha). Elevated levels were also observed on the internal surfaces of the equipment and other components. The plant's decommissioning plan, which had been filed with and accepted by the state, called for radiation levels on equipment to be reduced to less than 500 dpm/100 cm2 of fixed alpha radiation.

During the project, EET personnel performed decontamination of the equipment using the TECHXTRACT process. SEG provided health physics support and project management. SEG also performed decontamination of other areas using traditional techniques. The machines were partially disassembled to allow access to all contaminated surfaces.

Several different decontamination approaches were used during the project. Small components were dipped and scrubbed in a series of TECHXTRACT chemical baths. Other disassembled items which were too large for the baths were cleaned by hand. In some cases, the radioactive matter was trapped in layers of water scale. This was dissolved using one of the chemical blends. The main machine structure was decontaminated in place, with the upper sections reached from scaffolding. Work was done in two, twelve-hour shifts to meet the plant's turnaround schedule.

EET and SEG encountered a number of unexpected conditions during the first few days of the project. Radiation levels were much higher and contamination was much more widespread than expected. There was also a significant amount of radioactive hold-up in the joints between components and other difficult to access areas. This resulted in more cleaning and more surface area for decontamination than expected. The hard water scale was another unique challenge of the project.

Despite these challenges, all of the equipment was decontaminated to the required levels for free release 500 dpm/100 cm2 (fixed alpha) under the plant's decommissioning plan. No components required disposal and replacement, and all of the precision machined parts remained within tolerance after chemical decontamination. The overall decontamination schedule was delayed, due to the circumstances described above, but this portion of the project was completed in approximately six weeks, and the first two machines were returned to the customer within their original time frame. As a result, the plant was able to resume production with minimal impact on its operations due the decontamination project.

CONCLUSION

The results from this project clearly show the ability of the TECHXTRACT technology to perform effective decontamination and extraction of radioactive contaminants. Other private industry and DOE projects have confirmed these capabilities for the decontamination of concrete, metal, equipment, and other items. The effectiveness of the technology lies in its ability to penetrate into the substrate so that contaminants such as radionuclides and PCBs can be solubilized and extracted. This technology offers significant benefits in the reuse of previously contaminated buildings and equipment, waste reduction, and avoided disposal cost. EET continues to perform research to improve the TECHXTRACT process and broaden its applications.

REFERENCES

1. The TECHXTRACT technology is patented under U.S. Patent Number 5,421,906. Other U.S. and international patents are pending.

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MEASUREMENT TECHNIQUES AND RESULTS, DATA ANALYSIS METHODS, AND MODELING USED FOR D&D AND WASTE CHARACTERIZATION OF ORNL, OLD HYDROFRACTURE FACILITY

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ABSTRACT

A series of site characterization measurements were performed in 1994 in preparation for the decontamination and decommissioning (D&D) of the Old Hydrofracture Facility (OHF) at Oak Ridge National Laboratory. The OHF was one of four sites in Melton Valley used in the development and full-scale application of the hydrofracture process, a unique disposal method that involved injecting radioactive liquid low-level waste mixed with grout and additives into a deep, low-permeability shale formation. The OHF facility did not appear to have been decontaminated before it was

abandoned; most of the equipment and piping used in the cells and control room are still present.

The objective of the site characterization was to determine the nature and extent of radioactive and hazardous materials and other industrial hazards in and around the structures. This information will be used in subsequent planning to develop a detailed approach for dismantling and disposing of the structures: 1) to evaluate and design the most cost-effective D&D approach; 2) to determine the level and type of protection necessary for D&D workers; and 3) to estimate the types and volumes of wastes generated during D&D activities and support decisions on waste disposal. The D&D characterization scope included the entire OHF structure, including the foundation and equipment or materials within the structure. To estimate potential worker exposure from the soil during D&D, the characterization scope also included the soils underneath and surrounding the building to a distance of 5 ft from the structure.

This paper will discuss measurement results, data analysis methods used to quantify various types of contaminants, and modeling used to estimate the relative isotopic distribution of the contaminants. Results from different types of measurements along with modeling results will be used to infer the spatial distribution and correlation of the contaminants and radiation fields they generate. The contribution of loose and fixed contaminants to the total general area radiation fields will also be estimated. The measurement methods included gross alpha, beta, and gamma surveys; directional gamma surveys; gamma spectroscopy; concrete coring; photography; and collection of soil and miscellaneous samples.

INTRODUCTION

Portions of the Old Hydrofracture Facility (OHF) at Oak Ridge National Laboratory (ORNL) were characterized in 1994. Characterization activities centered on various aboveground structures at OHF, equipment outside and inside those structures, and the soils immediately surrounding some of those structures.

The site characterization was performed under the ORNL Remedial Investigation/Feasibility Study (RI/FS) Project and made extensive use of the existing programmatic infrastructure. Permanent records of the investigation (including logbooks, photographs, laboratory analytical results, and engineering calculations) are maintained as RI/FS Project records.

Site characterization activities followed the Site Characterization Plan for the Old Hydrofracture Facility at Oak Ridge National Laboratory, Oak Ridge, Tennessee (1).

LOCATION

OHF, also known as HF-3, is approximately 1 mile southwest of the main ORNL complex in Melton Valley, near the intersection of OHF/WAG 5 Road and Intermediate Pond Access (Fig. 1). OHF was one of four sites in Melton Valley used in the development and full-scale application of hydrofracture operations. OHF covers approximately an acre; its boundaries coincide approximately with ORNL grid coordinates N17100, E28500 and N17300, E28700.

Fig. 1

HISTORICAL MISSION

OHF was constructed in 1963 to allow experimentation and operations with an integrated solids storage, handling, mixing, and grout injection facility. The facility was shut down in 1980 and transferred to ORNL's Surveillance and Maintenance Program.

The hydrofracture process was a unique disposal method that involved injecting waste materials mixed with grout and additives under pumping pressures of 2000 psi or greater into a deep, low-permeability shale formation (Fig. 2). The injected slurry spread along fractures and bedding planes for hundreds of feet from the injection points, forming thin grout sheets (often less than 1/8 in. thick). The grout ostensibly solidified and immobilized the liquid wastes.

Fig. 2

The facility was used for 7 experimental injection campaigns in 1964 and 1965 and 18 operational campaigns from 1966 to 1979. The experimental campaigns injected grout plus radioactive tracers; the operational campaigns injected group plus approximately 969,000 gal of liquid low-level waste (LLLW). The experimental injections were at an average depth of 945 ft; operational injections were at an average depth of 792 ft (2).

OBJECTIVES AND SCOPE

The objective of the field investigation was to provide information necessary for

- engineering evaluation and planning of decontamination and decommissioning (D&D) approaches,
- planning for protection of D&D workers, and
- estimating waste volumes from D&D activities.

The site characterization focused principally on OHF components for which the ORNL D&D Program is responsible: Building 7852, the four bulk solids bins, water tank T-5, pump P-3, and the pump house (excluding the valve pit, which is under the control of ORNL Waste Operations) (Fig. 3). For the purpose of characterization only, this report also addresses two items assigned to the ORNL Remedial Action Program:

- the injection wellhead in Building 7852, and
- the soil underneath and surrounding the D&D structures (e.g., Building 7852 and the pump house) within 5 ft.

Fig. 3

Characterization consisted primarily of inspections, radiological measurements, and radiological and chemical sampling and analysis. Inspections determine general facility conditions, as-built information, and specialized information (such as structural evaluations). Radiological sampling and measurements define the quantity and distribution of radioactive contaminants; this information is used to calibrate a dose model of the facility and estimate the total activity, in curies, of each major radioactive isotope. The radiological information from sample analyses is used to refine the radiological model of the facility, and the radionuclide and hazardous chemical analyses are used for waste management planning. This paper presents data from the field investigation and laboratory analyses in the form of summary tables of radiological and chemical contaminant concentrations, and a waste volume estimate.

Field investigations of ancillary aboveground external piping with 5 ft of the D&D structures was limited to visual identification and cross-referencing to existing drawings; no excavation was performed to locate or characterize underground piping or drains or investigate underground tanks and piping.

POTENTIAL CONTAMINANTS IN INJECTED WASTE

Huang et al. (3) estimated that the average concentration of radionuclides in the grout mixture prior to injection was approximately

0.26 mCi/mL or less for beta/gamma emitting radionuclides and 10 nCi/g or less for transuranic (TRU) alpha emitting radionuclides.

The radioactive tracers injected with the grout during the experimental campaigns was gold-198 (30 Ci), cerium-144 (4100 Ci), cesium-137 (5200 Ci), ruthenium-106 (40 Ci), strontium-90 (1400 Ci), and cobalt-60 (20 Ci) (4). The LLLW injected during the operational period contained approximately 604,000 Ci of cesium-137, 38,600 Ci of strontium-90, 233 Ci of Curium-244, and 5.8 Ci of TRU other than curium (5). Thus, the two principal contaminants of concern are cesium-137 and strontium-90.

Additional information on potential contaminants can be obtained by referring to analytical results of the contents of waste tanks T-1, T-2, T-3, T-4, and T-9, which were sampled during 1988 (6,7). The tanks were used to store LLLW until it was ready to be blended with grout, and they still contain significant quantities of liquid and sludge waste. In addition to cesium-137 and strontium-90, other beta/gamma emitters found in the tank contents were cobalt-60, europium-152/154/155, carbon-14, and tritium. Alpha emitters included uranium-233, plutonium-238/239, curium-244, americium-241, thorium-232/238, and californium-252.

Based on analysis of the water and sediment in the OHF impoundment and waste pit T-4 as reported by Huang et al. (3), beta/gamma emitters included cesium-137, strontium-90, cobalt-60, europium-154, and cesium-134; alpha emitters included curium-244, plutonium-238/239, americium-241, and uranium-235/238.

An overview of the potential Resource Conservation and Recovery Act (RCRA) status of the OHF tank contents is provided by Autrey et al. (6). None of the waste tanks contained a RCRA ignitable waste or were classified as RCRA corrosive. The pH of the tank liquids was basic (pH of 8.8 and higher). Toxicity characteristic leaching procedure (TCLP) tests were not performed, but Autrey et al. indicate that potential inorganics of concern include chromium, lead, and mercury based on elevated total concentrations, particularly in the sludges. In general, the tank contents contained little organic matter. However, volatile organic compounds of concern (TCLP constituents detected in tank samples) consisted primarily of solvents; RCRA-listed semivolatile organic constituents consisted primarily of various phthalates and ploynuclear aromatic hydrocarbons.

De Laguna et al. (8) indicate that the approximate chemical content of the waste solution (prior to mixing with solids) included NaOH [0.05 molar solution (M)], NaNO3(0.8M), NH4)2SO4(0.15 M), Al2(SO4)3(0.05 M), NaCl(0.05 M) and NaCO3(0.05 M).

CHARACTERIZATION APPROACH

Inspections

Activities included photography, measurement of as-built dimensions, inspection by a structural engineer, and inspection by an industrial hygiene specialist for lead paint and asbestos. Extensive photography was performed to document the current condition of the facilities and to help plan D&D activities. The photographs will be particularly valuable because different contractors will do different phases of the work, and new D&D participants will be able to quickly familiarize themselves with the buildings with minimal personnel radiation exposures. Measurements of as-built dimensions was necessary in this case because no as-built drawings were available and modifications had been made to the structures during their operating lives; such measurements might not have been necessary in a well-documented facility. Inspections by specialists such

as a structural engineer and an industrial hygienist add insight to the conditions and hazards that may be present and help to plan for these conditions.

Radiological Measurements

Field measurements of the radiological conditions in these D&D buildings are of primary importance; measurements can be conveniently divided into "general area" and "location-specific." General area measurements included exposure rate surveys, directional gamma measurements, and gamma spectroscopy. The exposure rate surveys provide the general area exposure rates needed for ALARA planning and D&D task sequencing. Directional gamma measurements provide radiation profiles for modeling radiological sources within the buildings. Gamma spectroscopy provides isotopic information for radiological modeling and waste management planning. All of these general area measurements are amenable to both direct surveys and remote measurements where human access is impractical or not ALARA.

Location-specific measurements (field counts and smears to quantify loose contamination) were done on potentially contaminated structural surfaces within the buildings. A protocol was developed to help ensure quantitative results under field conditions. First, calibrated field instruments were source- and background-checked before each day's use. Second, at each selected location the measurements were as follows:

Using a 10 cm x 10 cm square template, the location was outlined and numbered.

Using a 0.68-cm standoff spacer, an alpha measurement consisting of three integrated counts was conducted.

Using a 10-cm standoff spacer, a beta/gamma (open window) measurement consisting of three integrated counts was conducted.

Using a 10-cm standoff spacer, a gamma (closed window) measurement consisting of three integrated counts was conducted.

A smear sample was collected inside the 10 cm x 10 cm outline.

The location was photographed.

Third, the field instruments were source- and background-checked at the end of each day's use. Primary instruments for these measurements were the Eberline HP-270 beta/gamma detector and the Eberline AC-3 alpha detector, both with the Eberline EPS-2 counter. The smear samples collected were to be analyzed for gross beta/gamma, gross alpha, and gamma isotopes (spectroscopy); strontium-90 analysis was to be conducted if gross beta levels were high and could not be accounted for by gamma spectroscopy results.

Sampling

The objectives of sampling were to 1) identify radioisotopes present, including certain TRU isotopes; 2) determine depth of penetration of radionuclides into concrete surfaces; and 3) screen for the presence of hazardous chemicals. Samples would consist primary of concrete cores and subfoundation soil samples, but provisions were also made for collection of miscellaneous "opportunity samples" that might be identified in the field. Because of the small size of these facilities and the high cost of laboratory analyses, sampling activities were limited to just a few in each room or cell. Laboratory analyses would include a full suite of radionuclides and hazardous chemicals. Analysis of radionuclide penetration into concrete surfaces was done by taking core samples of the concrete; these core samples were analyzed by slit-scanning with a high-purity germanium gamma spectroscopy system. Slit-scanning involves shielding the detector so that only a 1/8-in. or 1/4-in. slice of core is

measured. Each 1/8- or 1/4-in. increment is measured down the length of the core to develop a contamination profile.

Summary of Approach

A site characterization approach should be based on the data needed to plan and execute D&D of the facility. Table I summarizes the data needs, uses, and collection methods for characterization of OHF.

Table I

SUMMARY OF FINDINGS

The facility did not appear to have been decontaminated before it was abandoned; most of the equipment and piping used in the cells and control room are still present. Exceptions are the high-pressure pump and diesel engine used for grout injection, which have been removed from the pump cell and engine pad, respectively. The cells contain other items such as ladders, drums, cleaning equipment, ropes and cable, pins and bolts. The integrity of the OHF structures is adequate (i.e., the structures will remain structurally intact) for safe decontamination or demolition. All rooms and cells are contaminated with hotspots. The pumphouse, control room, and cell areas are all contaminated, with most of the contamination fixed to the surface of the room/cell walls and floors. Exceptions are the loose grout on the floor of the mixing cell and grout in piping. The exteriors of water tank T-5, pump P-3, and the four bulk storage bins were surveyed and found to be free of loose surface (smearable) contamination.

The concrete foundation slab of the room/cells ranges from 4 to 9 in. thick. The floor slabs in the mixing and pump cells consist of two distinct layers: a top layer (approximately 3 in. and 1 in. thick, respectively) and a bottom layer (approximately 6 and 7 in. thick, respectively). Slit scanning of the concrete cores indicates that most of the measured gamma activity is within the first few inches of the surface. The activity along the length of the core is nonuniform; the activity is generally high at surfaces, decreases to near background level, and increases slightly again near the bottom where the core contacts underlying soil. The two-piece core from the mixing cell behaves differently than the others; the top portion has high activity at the surface, decreases to near background level, and increases at the interface by a factor of 20. The bottom section behaves the same as the other cores.

General area measurements and concrete and soil sample gamma spectroscopy results indicate that the primary gamma emitting isotope present is cesium-137/barium-137m. Other radiological contaminants include strontium-90/yttrium-90, cobalt-60, uranium, thorium, and plutonium, although in very small amounts. Comparison of direct measurement and smear results indicates that most of the contamination is fixed on surfaces.

The general area average exposure rates (closed window) in these areas range from 3 to 60 mR/h. Alpha activity is generally higher in the cells than in the control room, engine pad, and pump house. The average alpha activity ranges from approximately 59 to 4500 dpm/100 cm2; smear results range from approximately 1 to 800 dpm/100 cm2. In addition, there are several hotspots in the mixing and pump cells and the pump house. The elevated exposure rates in the mixing cell are on the bottom of the mixing tank and the suction lines used for transporting group to the pump cell (120-160 mR/h and 270-400 mrad/h) and in the floor drain (300 mrad/h and 150 mR/h). The elevated contact exposure rates were on the two grout

suction lines from the mixing cell (800 mrad/h and 120 mR/h and 3000 mrad/h and 300 mR/h) and the pipe suspended (hanging vertically) from the ceiling (1000 mrad/h and 800 mR/h; these exposure rates were measured before the pipe was shielded with lead blanket and lead sheet). A few areas on the pump house concrete pad under the lead shielding exhibit high exposure rates (maximum 2500 mrad/h and 1000 mR/h) approximately 6 in. from the floor.

Location-specific direct beta/gamma average readings about 10 cm from the surface ranged from approximately 5 to 200 mR/h (closed window) and 6 to 300 mrad/h (open window). The open window measurements were approximately 10 to 30% higher than the closed window measurements because of the beta field.

The total activity (i.e., curie content) of the radionuclides in the structural concrete and group deposits was calculated using the concrete and grout volume estimates, the location-specific measurements readings, and the radionuclide concentrations reported by the analytical laboratory for the concrete cores and grout samples. The calculations assumed a density of approximately 2.35 g/cm3 for both the concrete and grout. Concentrations of radionuclides in the floor slabs and grout deposits were assumed to be equal to the concentrations discovered in the concrete floor core and the grout samples, respectively. No samples were taken from the concrete block walls, and the radionuclide concentrations in the walls were inferred from results of location-specific radiological surveys rather than direct laboratory analysis. To be specific, the radionuclide concentrations in the walls were assumed to be represented by the floor core concentrations multiplied by the ratio of the average of the location-specific radiological survey readings on the walls to the average readings on the floor. The alpha survey results were used to obtain the wall-to-floor ratio for the alpha emitters, and the beta/gamma surveys (closed window) were used to obtain the ratio of the beta/gamma emitters.

As shown in Table II, the total activity for various radionuclides was calculated separately for the floors, walls, and grout deposits, and then summed to obtain a total curie estimate. For most of the radionuclides, the total activity in the grout deposits is significantly greater than that in the floors or walls.

Table II

The waste disposal volume estimates include Building 7852, the pump house, the bulk storage bins and associated equipment, tank T-5, and pump P-3, plus the foundations of those structures and the equipment or materials within Building 7852 and the pump house. They do not include the soils underneath and surrounding the structures or the ancillary external piping or drains leading to or from the structures. Remediation of the valve pit, the injection wellhead, the soils, and ancillary external piping is not currently considered a part of D&D implementation. Miscellaneous items such as furniture, tools, spare parts, electrical conduit, minor piping, unattached piping, trash, and investigation-derived waste (e.g., protective clothing and equipment used during D&D implementation) are not included in the volume estimate. The disposal volume was estimated on the basis of as-built conditions determined through field investigation and Energy Systems design drawings.

Volume estimates were performed for concrete, steel, and lead materials. The concrete volume estimates for the structures include the walls, floor slabs, and foundations/footings. The steel volume estimates include the

building roofs, the bulk storage bins and associated equipment, pumps, vessels/tanks, platforms and supports, ducting, and major piping inside the structures. The lead volume estimate includes lead shielding in the pump house, lead plates on the mixing cell roof, and (assumed) lead bricks in a 55-gal drum in the control room. The building material volumes were multiplied by the following swell factors to establish a disposal volume: concrete 1.25, steel 1.35, and lead 1.35.

One set of waste volume estimates was prepared assuming complete dismantlement of the D&D structures and demolition and removal of all the above- and below-ground materials. No partial dismantlement options were considered. Table III shows disposal volume estimates as a function of waste category, D&D structure, and construction material. A significant fraction of the disposal volume is due to the four bins and tank T-5; the table uses the bin and tank volumes prior to their being sectioned or cut up.

Table III

Concrete rubble is estimated to contribute approximately 1/4 of the volume of waste generated during OHF D&D. The bulk storage bins, tank T-5, and other scrap metal contribute most of the remaining fraction of disposal volume; however, the metal contribution is based on the assumption that the bins and tank are unsectioned or uncut. The waste volume from tank T-5 and the bins can be reduced by orders of magnitude if cut into smaller pieces.

None of the concrete cores or soil samples contain base/neutral/acid-extractables, pesticides, or inorganics that exceed the RCRA toxicity characteristic equivalent limits for solids, nor do any contain polychlorinated biphenyls that exceed the limit imposed by the Toxic Substances Control Act. Two potential RCRA constituent metals (chromium and lead) were identified in the paint chip samples. A TCLP test should be performed on the rubblized waste before disposal to determine whether the chromium and lead are RCRA constitutes; however, the paint is not expected to fail the TCLP test because of its relatively low leachability. Two samples were collected from the bins for asbestos analysis; test results were negative. The lead (shielding) plates in the room/cells are considered low-level radioactive mixed waste.

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DEVELOPMENT OF REMOTE CONTROLLED PLANT DATA ACQUISITION SYSTEM FOR DECOMMISSIONING OF NUCLEAR FACILITIES

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ABSTRACT
JAERI has been developing the remote controlled plant data acquisition system as a part of the technology improvement program for decommissioning of nuclear facilities under a contract with the Science and Technology Agency of Japan. The outline of the system development and some results is described in poster session.
Decommissioning procedure planning of nuclear facilities has to be based on the latest and exact plant data. But sometimes the latest data could not be constructed from the design drawings if long years had been passed after the construction and some modifications had been done after the start of operation. So the data acquisition works of the location and the dimension of the facilities are indispensable for the decommissioning. Also the development of remote controlled plant data acquisition system is necessary for the decommissioning procedure planning of the high radiation area because of the difficulty of the accessibility to the area.
JAERI has developed the remote controlled plant data acquisition system which use the small mobile robot equipped with ITV and laser pointer from 1992. This system is planned to apply to the decommissioning engineering of the JAERI's reprocessing test facility (JRTF).
Remote plant data acquisition system consists of three sections, mobile section, vision section and control section. Mobile section has two main drive wheels at the center of the body, two pairs of legs with 2 joints and 1 assist wheel at the front side and the rear side. By the movements of these parts, the robot is able to traverse the flat face and slope, and to cross the piping. Vision section is composed of the TV camera for normal view, the TV camera for close-up view, the laser pointer and the light, and be able to collect the pictures of the object and the data of its location and distance. Control section is composed of the TV monitor, the computer (EWS) for the control of mobile section and the computer for

data processing. The function of the control section is the operation of the mobile section and vision section, control of the movements and inspection.
JAERI has already made the system and performed the mobile function tests and the data acquisition function test. In the tests, JAERI have collected the fundamental data of the system.
JAERI will perform the measuring function test using the mockup model of the actual plant facilities. The data update function by comparison of the actual plant view from the ITV camera and the 3-dimensional CAD view based on the old design drawings will be added to the system. Also, the improvement of the 3D-CAD and evaluation the performance, the efficiency and the restriction of system will be done and the system will be developed to apply the actual decommissioning design works of the JRTF.

INTRODUCTION

Decommissioning procedure planning based on the latest plant data of the decommissioning objects is necessary on the decommissioning of nuclear facilities. But sometimes the latest data could not be constructed from the design drawings if long years had been passed after the construction and some modifications had been done after the start of operation. So the data acquisition works of the location and the dimension of the facilities are inevitable for the decommissioning. Also the development of remote controlled plant data acquisition system is necessary for the decommissioning procedure planning of the high radiation area because of the difficulty of the accessibility to the area.
JAERI has been developing the remote controlled plant data acquisition system (RDAS) to collect the layout information and dimensions of decommissioning objects under contract of the Science and Technology Agency. The production of the system has been finished and the function test has been performed. Outline of the design consideration and performance of the system are introduced. And the test plan for 3-dimensional measurement of the mock-up model are described here.

DEVELOPMENT OF RDAS IN JRTF

In the JAERI's reprocessing test facility (JRTF), the latest data has to be updated for the decommission engineering because of the several modifications after the construction. So the development of remote controlled plant data acquisition system is necessary for the decommissioning procedure planning.
JAERI has developed the remote controlled plant data acquisition method by which use the RDAS system small mobile robot equipped with ITV and laser pointer. The plant data will be updated by the comparisons between RDAS actual plant data and the 3D-CAD data of the old drawings and the decommission engineering will be performed by these updated data. The flow of the system is shown in Fig. 1.
RDAS system has been made in 1992 and 1993. The mobile function tests and the data acquisition function test has been performed in 1994.
Fig. 1

DESIGN CONSIDERATION OF RDAS

These conditions have been considered on the design of the RDAS system from the environmental conditions of JRTF.
1) Considering reduction of weight and simplification of the device from the viewpoint of mobility and transportation, these items are adopted in the system design.
- use of aluminum alloy in main structures
- external power supply with remote cable

- use of optical-fiber cable for the signal cable
- reduction of the transmission device by built-in driving motor in each wheel and joint

2) Driving mechanism consists of 4 legs and 2 driving wheels to realize the following driving capability:

- climbing step and slope
- crossing over the trench and piping
- driving and rotation in flat face

Design condition of the RDAS has been shown in Table I.

Table I

CONSTITUTION OF RDAS

RDAS consists of mobile section (see Fig. 2) and control section. Mobile section consists of driving mechanism and visual attachments. Driving mechanism has the two pair of legs with assist wheel in front side and rear side. Each leg has 2 joints (knee & hip). 2 driving wheels are equipped at the center of the body of driving mechanism. Visual attachments for plant data acquisition consist of ITV cameras for normal view and for close-up view, laser pointer and light. Vision attachments are able to rotate to any direction by the pan and tilt mechanism.

Fig. 2

Control section (see Fig. 3) consists of control panel, computer and monitor. This section is used for the control of mobile section, operation of the driving mechanism and the visual attachments, setting of the mobile condition, surveillance and measuring of distance and self position.

Fig. 3

In the control section, the measurement of the relative distance function and the measurement of the self-position function has been equipped. The absolute coordinate value of self-position of RDAS has been calculated from the relative distance between the self position and base points (more than 3 points) whose absolute coordinate values was precisely known. Flow of the measurement of the position is shown in Fig. 4.

Fig. 4

FUNCTION TEST OF RDAS

After the end of the assembling of RDAS, the following items have been performed as a function test of RDAS in 1994.

1) Driving forward, driving backward and change of the direction on flat face
2) Climbing the slope and step
3) Crossing over the trench and piping
4) Driving forward, driving backward and change of the direction on the girder
5) Measurement of the distance between the object and RDAS
6) Measurement of the absolute coordinate value of the object and RDAS

Design conditions has been confirmed by the test results.

FUTURE WORKS

The Combination test with 3D-CAD system will be performed in 1995. The main item of the combination test is as follows:

1) Comparison test of ITV view and 3D-CAD view
2) CAD data modification test by the result of RDAS measurement.

After these function tests, JAERI will evaluate the performance, the efficiency and the restriction of system and This system will be developed to apply the actual decommissioning works of the JRTF.

23-16

THE APPLICATION OF RECUPERATIVE FLAMELESS THERMAL OXIDATION IN THE TREATMENT OF VOLATILE ORGANIC COMPOUND VAPORS AND THE REMOVAL OF ASSOCIATED HYDROGEN CHLORIDE SCRUBBER SYSTEMS THROUGH RISK ASSESSMENT ANALYSIS

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ABSTRACT

From 1966 to 1970, approximately 88,400 gallons of organic waste from the Rocky Flats Plant were disposed in the Subsurface Disposal Area of the Idaho National Engineering Laboratory's Radioactive Waste Management Complex. As a result of the subsequent breach of this containerized waste, volatile organic compound vapors have migrated to the vadose zone in and around the Radioactive Waste Management Complex. A Record of Decision was signed on December 2, 1994, which specified vapor vacuum extraction with treatment as the selected remediation technology. The technology selected to treat the extracted volatile organic compound vapors was Recuperative Flameless Thermal Oxidation. This technology uses an inert ceramic matrix which acts as both a sink and a source for heat energy to bring the extracted volatile organic compounds to their oxidation temperature, thereby achieving a destruction efficiency of approximately 99.99%. This technology has numerous advantages, including a vapor destruction process which requires no regeneration or disposal of the matrix material and minimal by-product production. Typical volatile organic compound extraction treatment systems use hydrogen chloride gas scrubbers to lower hydrogen chloride emissions to acceptable levels. Risk assessment methods were used to show that risks from inhalation of untreated hydrogen chloride gas to workers and the public were within acceptable limits. This analysis facilitated negotiations with government regulators to eliminate hydrogen chloride scrubber systems, saving money and considerably simplifying treatment operations.

INEL BACKGROUND

The Idaho National Engineering Laboratory (INEL), formerly the National Reactor Testing Station (NRTS), encompasses 890 square miles and is located in southeast Idaho (Fig. 1). In 1949, the United states Atomic Energy Commission, now the Department of Energy (DOE), established the NRTS as a site for building and testing nuclear facilities. Waste generated during these activities was disposed within the boundaries of the site.

At present, the INEL supports many engineering and operations efforts of DOE and other Federal agencies, including nuclear safety, research, reactor development, reactor operations and training and waste management technology development. The DOE Idaho Field Office (DOE-ID), which is responsible for the INEL, designates authority to operate the INEL to contractors. The primary contractor for the DOE-ID at the INEL is Lockheed Idaho Technologies Company (LITCO), which provides management and operation services to the majority of INEL facilities. The remedial

design/remedial action contractor for LITCO at the INEL is Parsons Engineering Science, Inc.

OBJECTIVE

The objective of this paper is to explain the use of Recuperative Flameless Thermal Oxidation (RFTO) in the treatment of organic vapors and to discuss the advantages and disadvantages of RFTO with respect to more traditional vapor treatment technologies for remediation of Operable Unit (OU) 7-08 at the INEL. In addition, the non-traditional application of risk assessment methods used to negotiate the elimination of hydrogen chloride (HCl) gas scrubbers normally used with the RFTO system will be discussed.

Fig. 1

PROJECT BACKGROUND

The Radioactive Waste Management Complex (RWMC), located in the southwest corner of the INEL (Fig. 1), was established in the early 1950s as a disposal site for solid, low-level waste generated by INEL operations. Within the RWMC is the Subsurface Disposal Area (SDA), where radioactive and organic waste is managed via interim storage operations, waste certification, technology development and preparation of the waste for future shipment to the Waste Isolation Pilot Plant in New Mexico.

From 1966 to 1970, approximately 88,400 gallons of organic waste from the Rocky Flats Plant were disposed of in the SDA. The organic waste included 24,000 gallons of carbon tetrachloride, 25,000 gallons of other volatile chlorinated hydrocarbons, machine oil and calcium silicate absorbent, used to reduce free liquids prior to shipping the waste to the INEL. The containers of volatile organic compounds (VOC) were typically dropped or dumped from the sides of pits and trenches in the SDA and covered with soil to varying depths.

As a result of the burial and subsequent breach of these containerized organic wastes, VOCs have migrated to the vadose zone beneath and within the immediate vicinity of the SDA in the RWMC. The vadose zone is the area between the ground surface and the Snake River Plain Aquifer. The organic contamination in the vadose zone (OCVZ) at the RWMC has been designated as OU 7-08. The remediation of OU 7-08, as part of the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) process, does not include the waste buried in the pits and trenches in the SDA. Remediation of these wastes will be considered under a separate action.

The OU 7-08 remedial action has been implemented in accordance with the signed Record of Decision (ROD) (1) for OCVZ dated December 2, 1994. The selected remediation technology identified in the ROD is vapor vacuum extraction with treatment (VVET), which involves the extraction of the most significant concentrations of VOCs from the vadose zone beneath and within the vicinity of the SDA, and VOC destruction through treatment at the surface.

Extraction and destruction of the VOCs will reduce risks to human health and the environment associated with the organic contaminants present in the vadose zone and ensure that contaminant concentrations that may migrate from the vadose zone to the Snake River Plain Aquifer do not exceed state and Federal maximum contaminant levels.

During development of the ROD, a treatability study (2) was conducted to determine the feasibility of VVET in remediating OU 7-08. For this study, carbon adsorption was the selected treatment technology. The results of

the treatability study indicated that the geology of the subsurface beneath the SDA is conducive to vapor vacuum extraction.

However, while carbon adsorption effectively contained the extracted VOCs, its regeneration and off site disposal posed significant logistic and regulatory problems. Therefore, catalytic oxidation, or an equally-effective VOC destruction technology, was specified in the ROD and incorporated into the Request for Proposal for OU 7-08. The treatment technology proposed by the successful bidder, RFTO (3), is discussed below.

RECUPERATIVE FLAMELESS THERMAL OXIDATION TECHNOLOGY

The RFTO units (Fig. 2) are designed to extract and thermally destroy VOC-laden vapor from the vadose zone. The extraction and treatment process consists of several sub-systems.

The overall process is divided into three basic operations: pretreatment, thermal oxidation, and post-treatment of the oxidizer exhaust gas. A discussion of the general specifications and performance of major process equipment follows.

Pretreatment

The function of the pretreatment equipment is to collect the fume into a header using a vacuum blower, inject air and/or supplemental fuel, and thoroughly mix the fume/air/fuel mixture.

Fume is withdrawn from the well head under vacuum and conducted through heat traced and insulated piping to the header on the process skid.

In the main header, supplemental air is introduced into the line by vacuum. The fume then enters a vacuum blower capable of producing the vacuum required at the well head and the pressure required to push the fume through the process system. Supplemental fuel is injected through a sparger in the line (as needed) to maintain the oxidation temperature set point.

The fume stream and fuel are then mixed in a static in-line mixer which ensures the air and fume are adequately mixed before entering the oxidizer. Once air and/or fuel has been mixed with fume, the combined stream enters the thermal oxidizer. Temperature, pressure and flow are monitored and controlled in the main fume header.

Fig. 2

Thermal Oxidation

The thermal oxidizer consists of four major components: 1) a preheater used to bring the oxidizer to operating temperature, 2) a metal shell or containment vessel, 3) the refractory lining, and 4) the "matrix." The function of the matrix is to contain and control the oxidation reaction. Basic control is achieved by balancing the mass velocity in an upward direction with reaction velocity in a downward direction to maintain the reaction zone within a fixed location in the reactor. Since total flow to the reactor is controlled, the only variable is reaction velocity, which is a function of temperature. If the fume is lean, supplemental fuel is added to the fume through the sparger; if the fume is rich, the temperature indicating controller (TIC) will reduce or eliminate fuel addition.

During startup, the main fume line is isolated from the thermal oxidizer. Combustion air and fuel gas are admitted to the oxidizer preheater pilot using a burner management system. The upper section of the oxidizer is then heated to establish an appropriate temperature profile.

The metal shell of the oxidizer provides containment of the process gases. The refractory lining acts as an insulating medium to minimize

heat loss and to prevent the metal shell from reaching the high oxidation temperatures found in the matrix.

The matrix consists of an inert material selected for its thermal and flow distribution properties. The matrix is divided into two zones, the mixing zone and the oxidation or reaction zone. Ceramic balls and saddles of various sizes are selected to provide good mixing and distribution of the fume in the mixing zone and to provide both a sink and a source for heat energy in the reaction zone. During normal operation, fume, supplemental fuel and/or air are thoroughly mixed as they flow through the mixing zone. As the combined stream flows towards the reaction zone, it absorbs heat from the matrix. When the stream reaches the oxidation temperature, organic compounds oxidize to form carbon dioxide, HCl, unoxidized VOCs and water vapor, releasing heat that is reabsorbed by the matrix. Temperatures, pressures and flows are monitored and controlled in the thermal oxidizer.

Post-Treatment: Non-Traditional Use of Risk Assessment Methods

Typically, risk assessment activities under CERCLA are conducted to adequately and accurately characterize the potential risk from a given site (i.e. Baseline Risk Assessment). In addition, risk assessment methods are often utilized to evaluate the effectiveness and efficiency of one or more potential remediation options as part of the remedy selection process. In most cases, the application of further risk assessment methods following the remedy selection is minimal. However, the application of these methods to refine the selected remedy for OU 7-08, while non-traditional, has resulted in considerable savings in time and money while ensuring adequate protection of human health and the environment.

It became increasingly apparent through the planning and design of the selected remedy that HCl gas scrubber systems would be both financially and, more importantly, logistically burdensome. The realities of maintaining stocks of potentially dangerous chemicals, the logistics of frequent water supply deliveries, the regulatory and logistical issues associated with spent scrubber solution, etc., combined to make the prospect of scrubber system elimination attractive.

However, in order to facilitate the elimination of HCl scrubber systems, assurance was needed that treatment unit workers, co-located RWMC workers, and members of the public would not be exposed to unacceptable risks from the inhalation of HCl gas from OCVZ treatment unit exhaust. Calculations were performed to estimate HCl effluent concentrations based on anticipated VOC inlet concentrations and destruction removal efficiencies. These concentrations were then used as input parameters to an EPA-approved air quality model to evaluate annual average or chronic long-term breathing air concentration of HCl gas. Concentrations at both the maximally exposed worker and public receptor locations (Fig. 3) were used to calculate hazard quotients for receptors at the locations. In each case, the hazard quotient for the maximally exposed receptor was less than 1.

Based on these calculations, it was determined that the risks to the maximally exposed worker and public receptor were acceptable. Agreement was subsequently reached with governmental regulatory agencies to allow the elimination of HCl scrubber systems from OCVZ treatment units.

SUMMARY

The VVET system has been operating since January 11, 1996. The RFTO treatment units are performing as expected with few complications.

Preliminary off-gas monitoring results indicate that the assumptions made and calculations performed to support the deletion of the HCl scrubbers are valid. Off-gas monitoring will continue throughout the operation phase of the remedial action.
Fig. 3
REFERENCES
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23-17
REMEDIATING THE INEL'S BURIED MIXED WASTE TANKS
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ABSTRACT
The Idaho National Engineering Laboratory (INEL), formerly the National Reactor Testing Station (NRTS), encompasses 890 square miles and is located in southeast Idaho. In 1949, the United States Atomic Energy Commission, now the Department of Energy (DOE), established the NRTS as a site for the building and testing of nuclear facilities. Wastes generated during the building and testing of these nuclear facilities were disposed within the boundaries of the site. These mixed wastes, containing radionuclides and hazardous materials, were often stored in underground tanks for future disposal.
The INEL has 11 buried mixed waste storage tanks regulated under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) ranging in size from 400 to 50,000 gallons. These tanks are constructed of either stainless or carbon steel and are located at 3 distinct geographic locations across the INEL. These tanks have been grouped based on their similarities in an effort to save money and decrease the time required to complete the necessary remediation. Environmental Restoration and Technology Development personnel are teaming in an effort to address the remediation problem systematically.
INEL BACKGROUND
The Idaho National Engineering Laboratory (INEL), formerly the National Reactor Testing Station (NRTS), encompasses 890 square miles and is located in southeast Idaho (Fig.1). In 1949, the United States Atomic Energy Commission, now the department of Energy (DOE), established the NRTS as a site for the building and testing nuclear facilities. Wastes generated during the building and testing of these nuclear facilities were disposed within the boundaries of the site. These mixed wastes, containing radionuclides and hazardous materials, were often stored in underground storage tanks for future disposal. At present, the INEL supports engineering and operations efforts for the DOE and other Federal

agencies in areas of nuclear safety, research, reactor development, reactor operations and training and waste management technology development to name a few. The DOE Idaho Field Office (DOE-ID), having responsibility for the INEL, designates authority to operate the INEL to contractors. The primary contractor for the DOE-ID at the INEL is Lockheed Idaho Technologies Company (LITCO), which provides managing and operating services to the majority of INEL facilities. The remedial design/remedial action contractor for LITCO at the INEL is Parsons Engineering Science, Inc..

BURIED MIXED WASTE TANKS

The INEL has 11 buried mixed waste storage tanks regulated under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) ranging in size from 400 to 50,000 gallons. These tanks are constructed of either stainless or carbon steel and are located at 3 distinct geographic locations across the INEL. These tanks have been grouped based on their similarities in an effort to save money and decrease the time required to complete the necessary remediation. Environmental Restoration and Technology Development personnel are teaming in an effort to address the remediation problem systematically. The radionuclide contamination associated with the majority of the tanks is present at levels which will require remote handling techniques. In addition, the contents in at least one of the tanks meets the definition of transuranic waste (i.e. greater than 100 nanocuries/gram). Other contaminants of concern include volatile organics, heavy metals, and poly-chlorinated biphenyls.

The physical form of the waste contained in the tanks is a mixture of solid, sludge, miscellaneous debris and liquid. Studies were recently conducted to determine the approximate waste volume and radiological reading in each tank. Visual inspection was also performed using remote control techniques to determine relative tank integrity and impediments to tank pumping or mixing.

Work on the tank project is being performed in accordance with CERCLA. As such, project personnel must abide by regulations such as the CERCLA, Resource Conservation and Recovery Act, Toxic Substances Control Act, Clean Water Act, Clean Air Act, etc. In addition the tanks are a part of the INEL's Federal Facility Agreement and Consent Order where the DOE, Environmental Protection Agency and state of Idaho jointly work to solve remediation problems at the INEL. These three parties are kept apprised of the status of activities concerning the tank project and they will concur with any actions taken to remediate them.

The successful remediation of the INEL tanks will have benefits nationwide (and possibly worldwide) since many other areas with similar problems have not yet begun dealing with this challenge. The general strategy being applied to the tank project is to investigate the tanks in stages where each stage is completed at a specific decision point. The goal is to take advantage of the decision points to defensibly select a solution for each tank and implement that solution. A systems engineering approach is being used to reach a decision for each tank.

Fig. 1

Figure 2 depicts the decision process flow diagram.

This combination of contaminants, regulatory environment, physical location, and tank construction poses an interesting problem for the remediation engineer.

The remainder of this paper focusses on the remediation of one specific tank (referred to as "V-9"), the schedule, and finally conclusions reached to date.

The capacity of V-9, a sand filter tank, is 400 gallons; it is constructed of stainless steel and it has never been sampled. Based on historical records this tank was used for one day 40 years ago and then usage was discontinued because the sand filter clogged. Consequently project personnel are unaware of the level of contamination associated with the tank.

Tank V-9 is to be remediated through a cooperative arrangement between the DOE's Environmental Restoration (ER) Program and the Office of Technology Development (TD) with a carefully integrated scope, schedule and budget. ER and TD personnel have recognized the opportunity presented by the technical challenges of remediating mixed waste tanks and are using the expertise of each group to produce a technical and cost effective solution for Tank V-9. ER is contributing their expertise in preparing health and safety, CERCLA, and regulatory documentation and in coordinating and managing complex field operations and full scale systems. TD is contributing its expertise in field scale demonstrations of innovative techniques and in the application of commercial technology to new areas. This team approach provides the optimal solutions for the tank project and it will facilitate the transfer of the selected technologies to other sites.

Fig. 2

THE PROBLEM

The goal of the INEL buried mixed waste tank project is to answer the following questions and implement the results for each tank or set of tanks while keeping in mind all regulatory, technical, and management factors. Answering these questions provides the criteria which serve as the foundation upon which work is being conducted to remediate tank V-9 this year:

What is the preferred alternative for the tank using the CERCLA evaluation criteria?

Will the contents of the tank be removed?

If the contents are not removed, what are the requirements for monitoring and/or stabilization and how will they be implemented?

After removal, where and how will the contents of the tanks be treated and stored or disposed?

What requirements must be met for the contents to be stored or disposed?

What requirements must be met for the contents to be treated?

How can ER and TD team to solve the problem?

What technologies are available to minimize personnel exposure for all activities ranging from sampling to grouting to removal and disposal?

The INEL has begun the implementation phase of the tank project in order to answer these questions through the remediation of tank V-9.

THE PLAN

As previously stated this particular type of work has not yet been performed within the DOE system. Significant interest exists within the DOE as well as in the private sector to see how this project progresses. The tank project at the INEL will provide useful baseline information for mixed waste tank remediation world-wide. Key aspects and the associated schedule of the remediation plan for V-9 include:

Compilation of all existing data - complete

Characterization of tank contents to gather data to support risk assessments and treatability studies - summer, 1996

Grouting of the tank - summer, 1996

Monitoring of the grouted tank - summer, 1996 to spring, 1997

Preparation of documentation (sampling plan, health and safety plan, safety analysis documentation, environmental documentation, etc.) - spring, 1996 through spring, 1997

Removal, evaluation, and disposal of the tank and its contents (summer 1997)

In the case of V-9 the INEL buried mixed waste tank project is moving ahead with a treatment demonstration to grout the contents in a manner such that they can be easily retrieved, analyzed and retreated if necessary. The plan includes the sampling and grouting of the tank contents followed by monitoring using tracer tests for six (6) months. Removal of the tank and its contents is scheduled for the summer of 1997. During this removal, the grouted material will be sampled and analyzed to determine the success of the grouting demonstration. Final disposal of the waste form will also take place in 1997.

CONCLUSIONS

This paper discusses the work completed to date as well as future efforts (sampling, grouting, monitoring, removal, and disposal) for tank V-9. The results from this important activity will directly impact future decisions regarding grouting work at the INEL and elsewhere. Consequently, lessons learned from this project will be applicable to other government agencies as well as the private sector in remediating mixed waste tanks. Periodic status reports regarding the progress of this project are available from the authors. The results of this work will also be discussed in future publications.

23-18

SUCCESSFUL RETRIEVAL OF RADIOACTIVE WASTE BY BNFL AT SELLAFIELD IN THE UK

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ABSTRACT

British Nuclear Fuels Plc (BNFL) owns and operates both the UK's nuclear reprocessing facility at Sellafield in West Cumbria and the UK's principal low level waste (LLW) disposal site at Drigg, situated some 6km to the south east of Sellafield. Over a period of years routine nuclear reprocessing activities have given rise to both liquid and solid wastes of various types which must be retrieved in a controlled manner prior to their long term storage and disposal. As a result BNFL have an ongoing program for developing and implementing techniques for retrieving different waste types. This paper describes the successful implementation of three of the many retrieval projects currently being carried out by BNFL:

Fuel pool sludge retrieval

Fuel cladding swarf retrieval

The retrieval of dry solid transuranic contaminated waste

Although these activities relate to quite different applications they are all subject to common management controls which are aimed at achieving

cost effective solutions with high levels of safety to both personnel and the environment.

FUEL POOL SLUDGE RETRIEVAL

Background

The reprocessing operations carried out at BNFL's Sellafield plant involve extended immersion of spent nuclear fuel in water filled pools which are continuously purged with fresh water. This water passes through concrete settling tanks to remove traces of sludge and corrosion products. In the case of pool water from the Magnox (magnesium alloy full cladding) program a series of open settling tanks were used between 1960 and 1986, during which time approximately 170m3 of sludge accumulated. As part of the long term strategy for retrieval and storage of nuclear waste BNFL decided to retrieve this sludge for two main reasons. Firstly, the sludge itself gave rise to significant radiation levels around the tanks which severely limited working times in the area because of the problems of high dose uptake. Secondly, the tanks which were of an early single containment design were showing signs of ageing and it was decided to transfer the sludge to storage facilities built to current containment standards, pending its subsequent treatment then encapsulation in a concrete grout.

The Solution

After extensive design studies aimed at achieving the technical objectives whilst maintaining dose rates as low as reasonably practicable (ALARP) the concept of a remotely operated travelling unit was devised. Initially, a series of inactive commissioning trials were conducted on a mock up of the tanks. This step brought significant advantages to the project:

It permitted the optimization of operating and maintenance procedures before any exposure to the radiation environment.

Operating and maintenance personnel could be trained in the inactive environment. In this connection extensive photographic and video records were made for future reference.

Safety issues could be documented and the proposed procedures fully explained and justified to the regulatory bodies.

The actual machine installation is shown in Fig. 1.

Fig. 1

The retrieval machine comprised two desludging heads - a large one for the settling tanks and a smaller one for the inlet and suction chambers. These heads are suspended from a travelling bridge and crab unit which allows access to the full area of the tanks. This arrangement minimized lifts over the structure and reduced loading on the tanks. The machine was remotely operated from a control room, utilizing closed circuit television cameras. The sludge discharge and dilution water feed was achieved using double containment flexible connections to the pipebridge. The equipment resuspends the sludge using water jets in a controlled manner converting it to a slurry whilst minimizing disturbance at the tank surface and preventing spread of airborne contamination. The slurry is pumped via a pipebridge to intermediate settling tanks before transfer to the Site Ion Exchange Plant (SIXEP) and ultimately encapsulation in the Waste Encapsulation Plant (WEP).

Achievements to Date

Installation of the machine began in April 1992 and was completed by mid August 1992. Over the next two years desludging progressed at a rate better than the initial target figures as shown in Fig. 2, until minimal

residual sludge remains. Development and design work is now in hand to remove residual sludges and debris and to plan full decommissioning of the facility.

The success of the project was due mainly to teamwork and the use of a systems engineering approach. The video and photographic records were used to brief and train the workforce prior to undertaking critical tasks. The machine was installed with considerably reduced radiation exposure when compared with original predictions.

Fig. 2

FUEL CLADDING SWARF RETRIEVAL

Background

The decanning of spent Magnox fuel elements gives rise to intermediate level waste (ILW) in the form of irradiated magnesium alloy swarf. Over a period of years the swarf has been stored in a series of 22 water filled concrete silos or compartments at Sellafield as shown in Fig. 3.

There was a phased commissioning program in which the first six silos were commissioned in 1964 with the final silo brought into operation in 1983. The early silos contain a mixture of Magnox swarf, sludge and beta gamma waste. The contents of the last four silos are mainly Magnox swarf, with very little sludge.

Fig. 3

It is BNFL's intention to transfer all Magnox swarf to the Magnox Encapsulation Plant (MEP) which commenced operations in 1990. This plant encapsulates Magnox swarf in concrete grout in 500 liter stainless steel drums which are then transferred to a purpose built store prior to long term disposal.

The strategy for retrieval concentrated development effort on the equipment and operational requirements for retrieval of the swarf in compartments 19-22 which is suitable for direct encapsulation in MEP. The experience gained in this phase will then provide a valuable input to the design of the more complex equipment required for the mixed waste in the earlier compartments.

The Solution

The Swarf Retrieval Facility (SRF) designed for compartments 19-22 is shown in Fig. 4.

The machine is designed to interface with the standard silo opening and includes an hydraulically operated petal grab which is lowered via a hoist into the water covered swarf. When the full grab is raised a swarf bin is traversed to a position underneath it and the grab is opened to release the swarf into the bin. When the swarf bin is charged to a sufficient level the contents are monitored and raised via a gamma gate to a swarf bin transit flask which is used to transport the charge to the MEP facility.

Fig. 4

In developing this concept design and operational personnel worked as an integrated project team to ensure that safety and radiation issues adhered to the ALARP principle (As Low As Reasonably Practicable). This resulted in a number of notable features.

Because of the high levels of radiation and contamination the machine was designed to be as maintenance free as possible. It was constructed in modular form so that failed equipment could be easily removed and replaced.

A weight optimization study was carried out to minimize silo roof loadings whilst still maintaining a sufficient level of shielding to conform to modern plant standards.

A unique lifting beam was designed to achieve the required clearance when lifting transport flasks onto and off the machine.

The machine was designed so that equipment which protruded below its lower shielding could be retracted when the machine was moved. This had a twofold benefit of minimizing dose uptake and reducing the risk of collision and dropped objects.

Two independent seals provided an interface with the silo ventilation system. The integrity of the ventilation system was thus maintained by fulfilling the two main design criteria. The air gap between the water level and the silo roof was kept slightly less than atmospheric pressure to avoid spread of contamination to the operating environment and there was no undue build up of hydrogen.

Inactive testing was carried out off site at contractors works, saving radiation dose and providing invaluable training.

Achievements to Date

Because the machine had to be installed on an existing active plant much emphasis was placed on the need to achieve a "Right First Time" result by applying the principles of Total Quality Management (TQM) and teamwork. With this in mind a chargehole mock-up was built at the manufacturers. This allowed time to be spent proving the installation procedure, testing the machines durability and allowing modification while still in a clean environment. Procedures were developed and operations and maintenance personnel were trained in their use while the machine was still in an inactive situation. The benefits of this approach are demonstrated by the fact that actual dose uptakes during operation are significantly less than were originally predicted.

Due to these project management initiatives the SRF was delivered, installed and commissioned ahead of schedule and is currently exceeding its yearly productivity targets as shown in Fig. 5 overleaf. Original expectations were that it would empty a compartment in approximately 18 months. In practice it has already emptied one in a 14 month period which included an initial commissioning phase of one month and a one month division wide shutdown.

Fig. 5

As operations progress the project team is continuing to introduce improvements to allow for variations in the swarf quality. This experience is being fully documented and will be of benefit to other projects, particularly when the team faces the challenge imposed by the mixed wastes in the earlier compartments. This challenge includes construction of three larger, more complex Silo Emptying Plants (SEP I, SEP II and SEP III). Designs are well advanced and these machines will be manufactured and installed by 2001. A major downstream facility, Sellafield Drypac Plant, is being constructed on the same timescale to receive, treat and compact the waste prior to encapsulation.

DRY SOLID TRANSURANIC CONTAMINATED WASTE RETRIEVAL

Background

Transuranic (TRU) waste originating from the early UK defence program was placed for interim storage in existing structures, know as magazines at Drigg. A typical magazine is shown in Fig. 6. The waste was contained within steel drums of up to 205 liter capacity or in larger timber and plastic cuboid containers known as "crates".

Fig. 6

Design Philosophy

The general philosophy has been to provide semi-permanent structures (Magazine Modules) for the controlled removal of TRU Waste. These are modular mild steel fabrications bolted together with watertight sealed joints which are designed to interface with the entrance to each magazine in turn. The design has been executed on the basis that in-magazine retrieval operations will be carried out in protective clothing (supplied air suits) with provision incorporated to achieve high integrity containment at points of export from the magazine, such that all subsequent operations will not require extensive use of protective clothing.

Drum Retrieval

For the drums a procedure was developed to admit operators to each magazine in turn, maintaining magazine containment, so that each drum could then be checked, have containment improved by bagging or over-drumming as necessary, be checked free from contamination, have surface dose rate recorded and be assayed for fissile material content. In recognition that all of the drums would eventually require transport from the site they were all provided with additional containment, although inspection revealed that this was only an absolute requirement for less than 1% of the drums. Forced filtered extract from the magazine was provided inducing an air flow across the changeroom barrier in accordance with usual practice. Although there was a provision for operators to enter the magazines in pressurized suits supplied with air lines this facility was only used on an as required basis. Even in some cases where it was used it was only a precautionary measure because portable equipment used to measure alpha in air could not always discriminate adequately between plutonium and high levels of radon/thoron.

Fissile material assay was accomplished using a combination of direct weighing (from which density of the waste matrix was derived), low resolution gamma spectrometry (gives plutonium content result independent of burnup in reactor or age since separation, when energy band is carefully chosen from the measurement, and with an absorption correction based on matrix destiny) and passive neutron counting (for high density waste matrix). No reliance was placed on their earlier records or labels.

Retrieval facilities were moved from one magazine to another by contractors, each move taking less than a week to prepare hardstanding. Over the period 1976 to 1986 some 10,500 drums were retrieved from the magazines and those magazines that were then empty were decontaminated as necessary and returned to unrestricted use. Although the team of operators was not employed full time on this task direct labor and supervisory input to the project averaged three man years per year with additional maintenance, management and health physics input as required. For these personnel the group average radiation dose uptake did not exceed 2 mSV per annum during the drum retrieval project. The capital cost of the equipment employed was of the order US $200,000 (mid '70's money values), excluding the cost of an interim store which was constructed on the site as a buffer between the retrieval operation and subsequent off-site transport.

Crate Retrieval

Five of the ten magazine stores on the Drigg Site contain some 160 non-drummed packages still to be removed, and 120 drums which are presently inaccessible or too heavy for removal by manual means.

The items are to be retrieved, overcrated or re-drummed, monitored and transported (within appropriate containment) by rail and road links to Sellafield for subsequent storage and processing for disposal.
A new specially designed module is ready for use and regulatory permission is awaited to start operations, see Fig. 7. The crates will be retrieved, overcrated, monitored (to determine fissile material content and to demonstrate acceptable levels of radiation or external contamination) and transported to Sellafield for storage awaiting treatment.
Fig. 7
The facility will be relocated at intervals of approximately one year in order to carry out operations at the five magazines still containing TRU waste. The modules are steel fabrications bolted together, incorporating weathertight sealed joints capable of multiple dismantling and re-erection over a five-year period without loss of alignment. Each module is thermally insulated with floors suitably reinforced to sustain a ten tonne floor loading. The modules when assembled, provide for access of personnel in protective clothing and systems for the loading, assay and handling retrieved wastes. From knowledge of the relatively low prevailing dose rates within the magazines and that wastes being retrieved are TRU, the principal radiological hazard is considered to be the spread of contamination and release of airborne activity. A dedicated ventilation extract system is provided to draw clean air through the modules (low contamination) to areas of higher contamination.
Considerable attention has been paid to appropriate fissile content assay systems. Initial monitoring takes place immediately out of the magazine environment in the magazine extension. Crates and drums will subsequently be moved to a Central Monitoring Facility.
The central monitoring building, Fig. 7, contains the following items:

A crate monitor of sufficient size to enclose the largest overcrated TRU waste package and monitor each crate for up to 48 hours to obtain an accurate fissile material content.

A drum monitor capable of accepting up to 500 liter drums which uses total and coincidence neutron counting and gamma spectrometer techniques to determine the fissile content of all drummed waste.

An area of the monitoring building is partitioned so as to form the monitoring control room.

Adjacent to the central monitoring building is a storage compound for monitored, unmonitored and unused overcrates and drums. Within this area the monitored overcrates and drums will be loaded and secured inside ISO freight containers for transportation to Sellafield Site. Unloaded ISO freight containers are to be returned to Drigg for re-use.

Safety

It will be appreciated from the foregoing description that the retrieval facilities have taken account of the following key safety aspects:

Criticality: measurement of the contents of crates/drums as close to the point and time of retrieval as possible, following a careful initial move.

Plutonium ingestion: containment of magazines, contaminated areas and of the waste itself using physical barriers, air flows, developed transport containers, and suitable protective clothing for operators.

Fire: minimization of combustible material loading, including over-containment, with fire-detection and alarm systems.

External radiation: monitoring of items, use of suitable handling equipment with area gamma radiation measurements and alarm systems.

Industrial safety: suitable equipment provided for in-magazine and subsequent handling of heavy items.

CONCLUSIONS

The three projects outlined in this paper illustrate BNFL's structured approach to the retrieval of nuclear waste prior to long term storage and disposal. In all cases, the key to a successful conclusion was the development of a team philosophy with designers, operators and maintainers combining their expertise throughout project life. The use of initial commissioning trials on inactive mock-ups proved to be a very effective method of ensuring that key operational and safety targets were achieved.

The major success factors included:

Safe operation of the integrated Sellafield site including storage, reprocessing, retrieval and waste treatment facilities.

Concentration on strategy, waste categorization and front end design at the early stages of the projects.

Use of TQM and a team building approach to problem solving.

Investment in development and off site inactive commissioning facilities.

The experience gained so far has demonstrated that BNFL has the ability to undertake difficult and complex waste retrieval and management projects in the UK and elsewhere in the world.

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DETECTION OF ALPHA CONTAMINATION INSIDE PIPES

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ABSTRACT

Traditional alpha monitors sense the presence of alpha-emitting contaminants by directly detecting the alpha particles. Thus the sensitivity of these monitors is limited by the short range of alpha particles in air (typically less than 10 cm). Contamination inside pipes is a particular problem; if the pipe is too small to permit insertion of a monitor, there is no way to detect internal alpha emission with a traditional monitor. In contrast, monitors using the long-range alpha detection (LRAD) technique detect the ionized molecules produced by interactions of the alpha particles with ambient air. Such monitors are limited by the lifetime of the ions (the observed several-second lifetime allows transport of the ions over many meters or tens of meters) rather than the range of the alpha particle. The characteristics of LRAD monitoring are ideally suited to measurements inside pipes. All of the ions created by all of the contamination inside a pipe can be measured in a single detector located outside the pipe. Because air is the "probe", convoluted or twisted sections of pipe can be monitored almost as easily as straight sections. A collaborative exercise between LANL and BNFL Instruments Ltd has resulted in both laboratory field testing of an LRAD-based pipe monitoring system at LANL and BNFL's Sellafield reprocessing

facility in the UK. In this paper, we will report the first results of these tests.

INTRODUCTION

The decontamination and decommissioning (D&D) programs within the US DOE and BNFL have identified a continuing and significant need for development of appropriate measurement technologies for use on residual activity in process vessels, pipework, and structural material. BNFL Instruments Ltd, a wholly owned subsidiary of BNFL, has shown that it would be beneficial to measure alpha contamination within interconnecting pipework both before and after removal of contaminated material from the plant. Both planning and execution of decommissioning and subsequent waste disposal would be facilitated by this knowledge.

Traditional methods of monitoring the interior of pipework, especially for alpha-emitting contaminants, are difficult at best and impossible at worst. Many of the characteristics of the long-range alpha detection (LRAD) technique are ideally suited to measuring contamination inside enclosed volumes such as pipework. A collaborative exercise between LANL and BNFL Instruments Ltd has resulted in both laboratory and field testing of an LRAD-based pipe monitoring system at LANL and BNFL's Sellafield reprocessing facility in the UK.

TRADITIONAL PIPE MONITORS

Locating small quantities of alpha-emitting contaminants such as plutonium and americium inside small pipes is often complicated by the poor penetrating ability of the alpha particle. Probes based on a, b, g, and neutron detection have been employed with varying degrees of success. The alpha particles commonly generated by these contaminants cannot penetrate the walls of even a very-thin-walled pipe and, in fact, cannot penetrate more than a few centimeters of air. Thus direct alpha probes cannot be used outside a pipe and must be close to the contamination when deployed inside a pipe. A small traditional alpha detector can be inserted into a pipe to search for contamination. This technique works well with simple systems of large diameter pipes but becomes rapidly more difficult as the complexity of the pipework increases.

Beta particles are not produced by many of these contaminants and have poor penetrating ability as well (although not as poor as the alpha particle). Thus, except in special cases, beta particles are not useful probes for alpha-contaminated pipes.

Many (but not all) alpha-emitting contaminants also emit low-energy gammarays or X-rays or both as they decay. These gamma and X-rays can be difficult to detect outside the pipe when large sources are present and extremely difficult or impossible on small sources.

Finally, passive and active neutron measurements are very useful for determining the location of large amounts of contaminant within an accessible pipework system before operational materials are cleaned out of the system. These techniques have only limited usefulness in the low-level contamination and buried pipework scenarios common in facility D&D.

Thus, traditional measuring technologies can provide useful information on large amounts of alpha-emitting contaminants and contaminants located in simple or exposed pipe systems. However, many contaminated pipes do not fall into one of these categories. There are often a number of less-than-satisfactory alternatives including partial monitoring (of exposed surfaces), physical destruction of the pipe, complex internal monitoring systems, and handling and disposal of the pipe as contaminated waste, whether or not it is truly contaminated.

LRAD TECHNOLOGY

In contrast to traditional particle detectors, LRAD-based monitors detect the ionized air molecules generated by an alpha particle interacting with ambient air. A typical 5-MeV alpha particle will produce about 150,000 electron-ion pairs throughout its several-centimeter range in air. The electrons attach relatively quickly to neutral air molecules, producing heavy negative ions in equal numbers to the originally produced positive ions (1). The ions are transported to an electrically conductive detection grid, where the current produced by the ions can be measured with an electrometer or similar instrument. The observed current is proportional to the number of ions, which is in turn proportional to the amount of alpha contamination in the system. Although the principles of the LRAD technique have been discussed in detail elsewhere (2-4), we will discuss the basic principles, especially as related to pipe monitoring systems, in this paper.

The air ions can be transported to the detection grid by an electric field (electrostatic LRAD) or an air current (airflow LRAD). An electrostatic LRAD-based system utilizes a weak electric field (there is no gas gain in standard LRAD systems) to transport ions from a relatively flat surface to an ion collector plate. Airflow LRAD monitors (such as the pipe monitors discussed here) rely on a moving air current to transport the ions to a collector grid. An example of a generic airflow detector system is shown in Fig. 1.

Fig. 1

Monitoring systems based on the LRAD principle have several inherent advantages over other types of detector. These can be broken into three categories: 1) operational advantages, 2) engineering advantages, and 3) physics advantages.

Operational Advantages

Monitors based on LRAD technology operate in real time. The monitoring results are available to the workers in the field who can use it to decide about the safety of particular operation.

In Situ monitoring has been defined many ways. One useful definition is that the contaminated area is undisturbed and that no secondary waste is generated. Well-designed LRAD-based monitors satisfy these criteria and provide a combination of sensitivity and portability that is not available from other types of monitors.

All of the ions from a widely distributed area of contamination can be collected onto a single measurement grid. This gives the LRAD monitors the ability to monitor extended sources in a single measurement rather than requiring the summation of many smaller measurements.

Using ambient air as a detection and transport gas has additional advantages beyond those of a windowless detector. It is very easy for air to flow through a complex pipe network without getting caught on turns or constrictions. Thus, the air can penetrate complex pipe networks (except for blind limbs) more easily than traditional detectors.

Engineering Advantages

The LRAD detector system is very simple. There are no moving parts, no high-voltage power supplies or pulse electronics, and no special detector gases. As shown in Fig. 1, the only critical parts are the ion collector plate (or grid) and the dc electrometer required to measure the ion current.

This simple design implies the required field reliability. There are no fine wires, thin windows, gas handling systems, optical connections, or

complex electronics. This small number of parts leads to a long life expectancy, but, more importantly, LRAD-based detectors are rugged enough to stand up to repeated use in field conditions. The "sensitive" element in an LRAD-based detector is typically a thick piece of metal (generally aluminum or copper); temperature changes, scratches, or even deformation do not affect its detection ability.

Physics Advantages

As detailed in Ref. 2, LRAD detector systems are capable of measuring contamination levels of less than 100 disintegrations per minute (dpm) or less than 10 dpm/100 cm2. The length of time required to make these measurements is often limited by the counting statistics of the alpha particles rather than by the detector itself.

LRAD PIPE MONITORS

Both applications of LRAD-based monitors to D&D projects in general (5,6) and prototype pipe monitoring systems (7,8) based on the LRAD technique have been described in some detail previously. The data presented here were taken with the 10-cm-diameter pipe monitor described in Ref. 8 and with the more rugged and adaptable monitor illustrated in Fig. 2 and described below.

Fig. 2

The adaptable pipe monitor consists of two modules that can be attached to random sizes of pipe. Each of the modules has a 15-cm iris valve for connection to pipes between 1 and 12 cm in diameter. The other significant part of the input module is a 15-cm-diameter air filter for removing particulates from the ambient air. The mixing volume allows air to be drawn through the entire filter rather than just a smaller area at the center.

The second, or output, module again has an iris valve for connection to the pipe. An ion collection grid (similar to that shown in Fig. 1) is mounted inside the second mixing volume. The final component is a small fan oriented to draw air through the monitor. This system is identical in principle to the monitor described in Ref. 8 but is better suited to use with a variety of pipe diameters and wall thicknesses.

LABORATORY TESTS

Two important parameters that were not fully addressed in the work presented in Refs. 7 and 8 are the effects of air velocity and pipe diameter. Figure 3 illustrates the results obtained when the detector was connected to a 600-cm length of 10-cm-diameter pipe; the two curves represent the response to an alpha source which was placed either 60 or 540 cm away from the collection grid. The intrinsic sensitivity of an LRAD sensor is expressed as detected electrons per radioactive decay. Converting this sensitivity into measured quantities [the output of the LRAD is measured in femtoAmperes (fA) or picoAmperes (pA) and source strength is measured in dpm] reveals that a physical measure of the sensitivity of the detector to radiation is fA (or pA) per dpm.

Fig. 3

Both response curves exhibit a pronounced peak at about 100 cm/sec. The maximum sensitivity of this detector grid at 60 cm is about 0.11 fA/dpm and at 540 cm it is about 0.051 fA/dpm. The 540-cm curve is lower because some of the ions produced at 540 cm recombine before detection. We interpret the loss of efficiency at low velocity as loss of efficiency in transporting ions from the source to the detector. The loss at high velocity is probably due to "punch through," a phenomenon where some of the quickly moving ions pass through the holes in the detector without

being attracted to the grid. The grid is held at several hundred volts (see Fig. 1), which is insufficient to collect all fast-moving ions.

To examine the details of LRAD detector response at lower airflows, we made a series of measurements in the system illustrated in Fig. 2. A 15-cm LRAD grid was attached to a section of 5-cm-diameter pipe that was either 1.22 m (data shown in Fig. 4a) or 6.10 m (data shown in Fig. 4b) long. The air speed through the pipe was varied by adjusting the voltage applied to a variable speed fan.

Fig. 4

The response in a 1.22-m pipe has a broad plateau extending from about 70 cm/sec to above the measurement limit of 120 cm/sec. The plateau with a longer (6.10 m) pipe starts at a much higher airspeed (about 100 cm/sec).

FIELD TESTS

We have identified a number of field tests that will further validate both the concept and practical design of the LRAD-based adaptive pipe monitor. These can be broadly divided into three areas: sensitivity, geometry, and other effects.

Sensitivity

To assess the sensitivity and limit of detection of the LRAD monitoring system, it will be used to measure small alpha sources placed within sections of stainless steel pipe with dimensions typical of those commonly found in operating and decommissioned facilities on the Sellafield site.

Geometry

The sensitivity tests will be repeated using pipes of varying length, diameter and curvature and varying the airflow through the pipe. In addition, the alpha sources will be placed at a number of positions within the pipes.

Other Effects

The effects of environmental changes will be evaluated by repeating the sensitivity tests throughout an 8-hour period and logging the temperature and humidity during the same period. Additional measurements using external (to the pipe) gamma-emitting sources (typically 370 kBq) will determine the sensitivity of the LRAD monitor to background radiation interference.

CONCLUSIONS

We have reached a number of tentative conclusions from the laboratory tests. Many of these will be investigated further during the field testing phase.

Perforated plate detectors of the same diameter as the pipe, such as that illustrated in Fig. 1, have a relatively sharp efficiency peak as a function of airspeed. In a 10-cm diameter pipe, this peak is at about 100 cm/sec.

Increasing the diameter of the detection grid relative to the diameter of the pipe (as shown in Fig. 2) can widen this sensitivity peak.

Increasing the air flow through a pipe can partially compensate for increased distance to the contamination. In particular, the optimum airspeed for a short pipe is not necessarily optimum for longer pipes.

The results obtained in short sections of clean 5-cm and 10-cm pipe are significantly different; larger changes in the pipe will probably result in larger changes in optimum detector design.

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AN UNDERWATER SHEAR COMPACTOR

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ABSTRACT
This paper describes a concept of a specialized decommissioning tool designed to operate underwater and to reduce the volume of radioactive components by shearing and compacting.
The shear compactor was originally conceived to manage the size reduction of a variety of decommissioned stainless steel tubes stored within a reactor fuel cooling pond and which were consuming a substantial area of the pond. The main objective of this tool was to cut the long tubes into shorter lengths and to compact them into a flat rectangular form which could be stacked on the pond floor, thus saving valuable space.
The development program undertaken on this project investigated a wide range of factors which could contribute to an extended cutting blade performance, i.e.: materials of construction, cutting blade shape and cutting loads required, shock effects, etc. The second phase of this project was to review other aspects of the design, such as radiological protection, cutting blade replacement, general maintenance, pond installation and resultant wall loads, water hydraulics, collection of products of shearing/compacting operation, corrosion of the equipment,

control system, operational safety and the ability of the equipment to operate in dry environment.

The results of the development and study phase provided the confidence to justify proceeding to a fully detailed design, supported by all the necessary calculations. The paper summarizes the extended work program involved with this shear compactor tool.

TECHNICAL REQUIREMENTS

Although the volume reduction system was required to be designed to handle steel tubes, the equipment had to be sufficiently flexible in concept to be capable of treating other decommissioned materials.

The prime aim of the chosen equipment was to treat stainless steel tube, of two sizes, both of 48 mm external diameter with a wall thickness of either 3 mm or 4.5 mm. Any system developed was to be able to operate in a reactor fuel cooling/storage pond, filled with demineralized water. The depth of the operation was to be such that the pond water would provide the radiological protection for the workers located on the pond surround. The chosen system could, if required, utilize the pond manipulator for manoeuvering long tubes and the resultant compacted products under water. As the pond in which the equipment was to be operated could, at any time, be connected by a sluice door arrangement, to another pond housing a small research reactor, the chosen system was to be designed to avoid inducing any underwater shock to the reactor or the pond structure. Because the space allocated to the volume reduction system was restricted, the equipment had to be easily removed and parked when not in use. Excellent visibility in the pond had to be maintained at all times and as a result, any hydraulic systems employed had to be water powered. There were, therefore, a number of important reasons why the equipment and any support structure used were to be of light construction, they being: to reduce the reaction loads in the pond walls to an absolute minimum and to prevent overloading of the pond crane when removing any equipment from the pond.

CHOOSING THE SYSTEM

A wide variety of volume reduction processes were considered for this application including sawing, slitting, grinding, shearing, etc. but because it was the system which produced no, or minimal secondary waste products like swarf or chippings, the shearing process was selected. The shearing action did, however, need qualifying. The shearing blade shape would need careful design to avoid any jamming of one blade against another and to produce a significant volume reduction, the shear action would need to be allied with a compaction system to flatten the tubes between each cut.

As a result of all considerations the combined shear/compactor was chosen as the most likely candidate system for this application. At this point in the investigations, it was decided to attempt to combine the shearing with the compaction operation, all in one machine, if possible. Before proceeding with the detail design, however, it was necessary to undertake some development work.

RESEARCH AND DEVELOPMENT

Compaction Tests

The first objective was to determine what loads were required for compacting the 4.5 mm thick tube, the worst case to be considered. A small rig was designed and built, generally as shown in Fig. 1. The test rig consisted of a base carrying two vertical pillars, these pillars guiding a moveable top plate. Both base and top plate were arranged to

carry the various shear blades to be tested at a later date, but for the compacting tests the shear blade slots were blanked off.

Fig. 1

By positioning this rig on the bedplate of a 1000kN Amsler tensile test machine and using the machine in it's compressive condition, the top anvil of the Amsler could be brought to bear on the top plate of the test rig, the applied load being accurately recorded on the tensile test machine's control system. During these tests, the degree of compact was recorded for a given load and tube thickness, by measuring the distance between the base and top plate of the rig. It is interesting to note that even though the tubes were compacted to a theoretical dimension of 2 x thickness, the actual dimension after relaxation of the compaction load was always slightly more, the recovery being due to the natural hysterisis of the stainless steel material. A variety of different lengths of tube were compacted, the tests showing that a length of 200 mm would be the optimum for underwater handling by the pond manipulator.

A summary of the results of the tests to compact a 48 mm diameter x 4.5 mm wall stainless steel tube of 200 mm length are shown in Table I.

Table I

Shearing Tests

The most common form of shear blade is shown in Fig. 2a. The problem associated with this particular form is that of the offset load, tending to force the blades apart. If the blades are not sufficiently stiff, the condition shown in Fig. 2b. arises, resulting in either a broken blade or the tube becoming jammed between the blades. The last condition would be particularly disastrous on a remotely operated machine in radioactive conditions. If the blades were made stiff enough to prevent these conditions occurring, the resultant weight would be such that they would prove very difficult to manoeuvre with remote tools. The risk of one blade galling or self welding to its partner could also not be overlooked, especially in an unlubricated system which relies on two heavily loaded faces moving past each other with very little clearance. It was, therefore, decided that the most promising form of shear blade to test was that shown in Fig. 2c. It was hoped to design the blades to provide sufficient load to reach the maximum tensile stress of the tube material before the cutting edges touched each other. If this could be achieved in the tests, it would mean that a definite stop could be installed on the final design of the machine to prevent the edges contacting and thus give a longer cutting edge life and minimize the need to change them. This type of blade would also prove very difficult to jam. The resultants of the cutting load would be equal and opposite to each other, imparting no side loads on the machine, providing the blades are correctly aligned. It was decided to test a variety of blade materials, hardness required, included angles, degree of pre-compaction, etc.

A derivative of the double blade is shown in Fig 2d. This type of cutter utilizes one of the previously discussed pairs in conjunction with a flat plate and was considered to be of interest because it obviates the need for the accurate alignment of the double blade arrangement. This variation was tested to compare the cutting loads required of the two systems.

Fig. 2

In reviewing the results of all the shearing tests, it was decided to opt for the double blade system, with an included angle of 750, because of

low load requirements, tool life and general durability. The tests had also revealed that the high stress/ no blade contact theory was valid and identified the gap that was required between the blades to achieve the desired tensile stress. The principle of compacting, while shearing took place, ensured that the sample was not ejected when the stress produced the break in the material. For the same reason, no shock was evident, a distinct requirement of the pond location. A summary of the results of shearing are shown in Table II.

Table II

To summarize, a combined shearing / compaction operation on a 48 mm diameter tube with a wall thickness of 4.5 mm and a length of 200 mm, employing a double shear blade with an included angle of 750, required a load of 970 kN. This force resulted in a product measuring 200 mm long x 13 mm thick. The combined shearing/compacting results are shown in Table III.

Table III

THE SHEAR / COMPACTOR MACHINE

To ensure that no shearing / compacting loads were transferred into the pond wall structure, the machine was designed as a self consuming load system. However, before commencing the design work, a number of important decisions were made, particularly in the light of the development test program results. These were:

as a water hydraulic cylinder capable of an output force of 1000kN was commercially available and any larger unit would have to be specially constructed, it was decided to use that equipment. It was constructed in the correct grade of stainless steel and had a proven performance when operating on a demineralized water fluid.

as a force of only 1000kN was available, with little in reserve over the required 970kN developed during the test program, it was agreed to limit the degree of compaction to provide a finished thickness of approximately 17 mm against the proposed 13 mm, demanding a reduced compaction load of 680kN. This important decision meant that the designers had some 30% spare capacity in the available hydraulic cylinder with a corresponding penalty in the finished thickness of the compact. It was considered acceptable to pay this loss of reduction for the cost advantage gained from using proven, available equipment.

The initial design layouts indicated that the overall size of the machine would be determined by two basic factors :

the maximum movement required between the two cutting blades and

the diameter of the power unit used to produce the force of 1000 kN

The size of the hydraulic cylinder mounting flange determined the dimensions of the cylinder mounting block. See arrangement of machine in Fig. 3. Parallel to the cylinder block and joined to it by four large tie bars, is the reaction block. Sliding along the four tie bars and moved by the hydraulic cylinder piston rod, is the moving block. Positioned on both reaction and moving blocks are the cutting blade location fittings, designed to permit the changing of the cutting blades without having to remove the machine from the water. These location fittings are so designed that they ensure correct alignment of the cutting blades and the tube being treated, an important requirement revealed by the test program.

Fig. 3

To ensure that the cutting edges do not come into contact with each other, four reaction blocks are fitted to the corners of the blades. The

shape of the blade and the reaction blocks control both the cutting and degree of compaction obtained. Two cutting edges are fitted to each blade, the top horizontal set for cutting vertical tubes and the vertical set for the horizontal tube arrangement. Both sets of blade location fittings have been designed to bolt to the flat surfaces of the moving and fixed reaction blocks. Thus, with the 250 mm stroke available from the hydraulic cylinder and a change of location fitting and cutting blade, tubes up to 190 mm diameter can be dealt with if required.
Bridging the bottom two tie bars and positioned below the blade location fittings, is a fixed stop plate which is 200 mm from the horizontal cutting edges. To one side of this fixed stop plate are two location pins which locate the adjustable and removable horizontal stop. This stop can be adjusted to any distance up to 910 mm from the vertical cutting edges.
On the top surface of the reaction block is a single lifting lug for removing the machine from the support structure. Fitted either side of the cylinder block are the machine support plates, joined at the top by a cylindrical bar.
All materials specified for the machine is generally stainless steel. Where friction of moving couples was involved, the stainless steel for construction was changed to a type with a high molybdenum content, thus minimizing the risk of galling or self welding. As weight was a dominant factor, particularly as the capacity of the pond crane was limited, and the reactions in the pond structure should be as low as possible, considerable thought was given to providing a light but structurally stiff assembly.
The estimated weight of the machine, in air, is 2,040 kg.
SUPPORT STRUCTURE
The support structure for the machine is constructed of three basic components:
  the top section which straddles the pond surround and clamps to it. To prevent overloading the wall with this manually operated clamp, a collar is placed between the handwheel and the clamp shaft support
  built into the top section is an operator's platform, complete with guard rails, giving an uninterrupted view of the machine some 3 m below the water level. All materials used in the construction of the top section are of normal carbon steel suitably protected against corrosion by plastic coating.
  The lower machine support section is suspended from the platform assembly by a stainless steel, 3 tube configuration. The three tubes are cross braced at regular intervals by fabricated diaphragms, to reduce the strut length of the rear tubes which are subject to a compressive load. Each of these tubes was left open at each end to permit the entry of water, thus maintaining adequate shielding for the operating personnel.
The fabricated lower machine structure is also fabricated from stainless steel, suitably stiffened with welded ribs. To reduce the weight of the machine support, each of the heavy side plates are, in fact, formed from two light plates some 25 mm apart, joined on all edges by a bridging closure plate. To provide an amount of positive buoyancy and reduce the wall reactions, no water is allowed to enter the enclosed hollow side plates.
The shear / compactor machine locates in the tapered vertical slots and the overhung machine weight is reacted through the adjustable stops provided on the inside of the side plates. Attached to the lower support structure is a perforated aluminum alloy tray fitted with a collector box

for the compacted tubes. The collector box can be removed when full and the tray can be hinged downwards when the space is required in the pond. The estimated weight of the support structure, in air, is 910 kg. The arrangement of machine, support structure and pond wall reactions are shown in Fig. 4. The wall reactions were all calculated for the 'in air' condition. The true reactions when the equipment is suspended in water will be reduced due to a degree of buoyancy inherent in the system.

Fig. 4

HYDRAULIC SYSTEM

To keep the size of the shear / compactor to an absolute minimum, a pressure of 210 kg.cm-2 is used to obtain an output from the hydraulic cylinder in excess of 1000 kN. The return stroke of the cylinder delivers in excess of 690 kN.

A normal oil pump is used to pressurize an actuating cylinder which is mechanically linked to a smaller diameter injection cylinder. The injection cylinder is filled on its suction stroke with demineralized water. On the injection cylinders pressure stroke, the shear / compactor machine cylinder is pressurized and moved a small distance. By continuing to inject demineralized water into the machine cylinder, the complete stroke is obtained by means of a series of small stepped movements.

By using this form of circuit, only the injection cylinder, machine cylinder, non return valves and vent valves are filled with demineralized water, thus reducing the stainless steel hydraulic equipment to a minimum.

OPERATING PROCEDURE

With the support structure clamped to the pond wall, the shear / compactor machine can be lowered into the water, with hydraulic system pipes connected, until the machine's horizontal bar locates into the tapered slots in the support structure. By continuing to lower the machine it will rotate about the bar until the machine is horizontal and supported in a cantilevered attitude. The crane hook should then be disengaged from the machine lug. Removing the machine from the pond is the reverse of the above procedure.

The recommended procedure for operating the machine is to undertake the horizontal cutting of the long tubes using the guide to produce lengths of around 900 mm, the resultant pieces being handled with the in-pond manipulator. It is accepted that a length of some 200 mm adjacent to the cut will be compacted but this will not affect the later operations.

With the bottom stop adjusted the 900 mm lengths are handled vertically into the machine until the tube rests on the stop. Operation of the hydraulic cylinder will result in a 200 mm compacted length falling from the cutting blade area, on to the inclined tray and then into the collector box fitted to the tray. This procedure to continue until the box is sufficiently loaded with compacted samples, when the crane hook can be used to remove and park the filled box elsewhere in the pond. A replacement box can be fitted and the operations continued.

MAINTENANCE

The only regular maintenance envisaged for the machine is the inspection and, if necessary, the changing of such items as the hydraulic seals, sliding bearings and the shear blades. Obviously, the life of these items will depend on usage and the period between changes will have to be determined by operational experience.

The equipment most frequently changed, however, will be the shear blades, although the development program indicated that something in excess of

200 cuts should be achieved before their efficiency is impaired. It was for that reason that the shear blade assemblies were made removable.

CONCLUSIONS

The main conclusion reached at the termination of this design/feasibility study was that a machine with a capability to produce 1000 kN of force, would produce a compacted product measuring 69 mm x 17 mm x 200 mm, with the machine suspended in water from a pond wall, with all operations able to be achieved remotely.

Although this shear / compactor machine was designed to handle tubes of 48 mm diameter, the stroke of the main cylinder and the distance between the tie bars is such that larger diameter tubes could be treated, if required. To undertake the larger tubes, however, it would have to be accepted that the wall thickness would need to be less. For example, it has been calculated that a tube of 190 mm diameter could be treated in the machine, providing the wall thickness did not exceed 1.8 mm. Different shear blades would be required for such an operation.

Other candidate materials which could be considered for treatment would be fuel assembly structures and tie bars, etc.

The machine was designed primarily for underwater operation, but as it could be demounted from it's support structure, it could be transferred to a 'hot cell' if necessary. It could, in that case, still employ the demineralized water hydraulics but would need a bench stand to position the machine at a height suitable for viewing through a cell window.

All of the very detailed development work, design and calculations produced for this exercise provided the authors with the confidence that the equipment should be built to become an essential tool for any cooling pond decommissioning project. Being a light structure throughout, the results of decommissioning this tool at the end of it's life would not add significantly to the overall cost of any dismantling program.

ACKNOWLEDGMENT

The authors would like to express their gratitude to Gradel S.A. of Luxembourg and Golder Associates (UK) Ltd, for their support and assistance in the preparation of this paper.

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RAPID DETAILED CHARACTERIZATION OF CONCRETE SHIELDING BLOCKS UTILIZING INTERNAL NATURAL RADIONUCLIDES FOR CALIBRATION

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ABSTRACT

This paper presents a method for the rapid characterization of gamma-ray-emitting radioisotopes in large samples of earth-like materials: concrete shielding blocks in this case. Active regions are identified with a sensitive radiation-survey instrument and then examined in detail with a high-efficiency lead-shielded Ge spectrometer. Naturally-occurring gamma-ray emissions from the decays of uranium, thorium, and potassium are used to calibrate the spectrometer. A simple relationship exists between the observed counting rate in a characteristic gamma ray and the activity in the block. This method, taking only tens of minutes per sample at the

nano-Curie/gram sensitivity level, replaces much of the expensive coring and laboratory analysis methods needed otherwise.

INTRODUCTION

Following many years of productive work, the SuperHILAC and Bevalac accelerators at Lawrence Berkeley National Laboratory were closed, leaving thousands of concrete shielding blocks available for reuse or disposal. The process history of these blocks as shielding precludes free release pending radiological characterization.

For the first half of its nearly 40 year lifespan, the Bevalac accelerator ran beams of protons at energies as high at 6.2 GeV. (Subsequent heavy-ion beams contributed small amounts of additional activation.) Interactions of these primary-beam protons on targets and machine components generated secondary particles, predominantly neutrons with long mean-free-paths, that activated parts of the shielding. As detailed in Refs. 1 and 2, the dominant process by which accelerator shielding blocks are activated involves capture by thermal neutrons. The isotopes 46Sc, 59Fe, 60Co, 65Zn, 134Cs, 152Eu and 154Eu are produced this way. To a lesser extent, fast neutron activation accounts for the production of 22Na and 54Mn. As discussed in References 1 and 2, thermal-neutron activation may increase slightly with depth, reaching a maximum several cm into the concrete. Fast neutron-activation, on the other hand, is at a maximum at the surface. Activities fall off approximately exponentially after the maximum with a half-thickness on the order of 10 cm. The very few blocks at 0 to GeV beams could have very different distributions. Since most of the high-intensity light-ion runs at the Bevalac occurred prior to 1971, only the longest lived products remain, namely 60Co and 152Eu.

A large number of Bevalac shielding blocks have been examined with a 2.54 cm diameter x 2.54 cm thick NaI survey instrument and found to exhibit surface activities over a range from several mR/hr above background to over 1000 mR/hr above background. Subsequent to this survey, several examples of blocks were examined with a high-efficiency lead-shielded Ge spectrometer to identify and quantify the activity. This spectrometer can easily distinguish natural activity (U, Th, and 40K) from added activity (mainly 22Na, 60Co, and 152Eu). The natural activity may be used to calibrate the detector for the quantification of added activities. Following calibration, this technique makes most coring and laboratory analysis unnecessary.

THEORY

As shown in Fig. 1, the detector views a volume of the sample defined by the solid angle of the lead collimator and a depth determined by the gamma-ray attenuation in the sample. Each unit volume contributes gamma rays that may be detected in a lead-shielded detector located next to the slab. For an activity of Av Disintegrations/minute/unit volume, and a fraction (BR) of disintegrations that produce a particular characteristic gamma ray, Av x BR gamma rays at the energy E of the characteristic gamma ray are emitted per unit volume. The observed count rate in the detector in counts/minute (CPM) is:

Fig. 1

Eq. 1

where m(Z,E,r,ri) is the gamma-ray absorption in a thickness ri, (Z is the effective atomic number, E is the gamma-ray energy, and r is the density.) e(E,W) is the detector efficiency. (Dependent on gamma-ray energy E and detector solid angle W, and the sum extends over all

elements DVi. Note that e(E,W) depends on W, and not on the distance between the sample and the detector face.
Converting to disintegrations/gram: A r = Av
Eq. 2
where e is, essentially, the absorption- and efficiency-weighted mass of the sample examined and is determined by comparing the CPM observed with the activity determined by measuring the activity of a core sample in the laboratory.
The activity A is determined by:
Eq. 3
PROCEDURES
Part I: Determination of Intrinsic Activity by Laboratory Analysis
The intrinsic activity of the bulk concrete is determined by gamma-ray analysis of kg-sized samples obtained by drilling 2.5 cm diameter holes into the concrete. Typically 6 holes 15 cm deep were used to obtain enough material. The hammer-drill used pulverized the material into a fine powder which was collected in a plastic bag. The contents were transferred to a Marinelli beaker and counted for 1 day on a calibrated 30% p-type germanium spectrometer at the LBNL Low Background Facility. Concentrations of uranium, thorium, and potassium were determined for various samples of normal- and high-density concrete.
Part II: Detector Calibration
Prior to drilling, a lead-shielded 80% Ge spectrometer was used to measure activity in-situ at the same sites as the core samples were taken. (This procedure, shown in Fig. 2, will be described more fully in the next section.) The detector response e is determined by comparing the observed count-rate (CPM) from characteristic gamma-rays from U, Th, and K observed in-situ with the U, Th, and K activities (A) determined via laboratory analysis as described above.
Fig. 2
Eq. 4
Bevalac shielding blocks are made of two different kinds of concrete having different physical and radiological properties. Normal-density concrete, specific gravity 2.4, and high-density concrete, typically containing iron-ore aggregates of specific gravity 3.5 were used in different parts of the shielding. Normal-density concrete contains the natural decay products from U, Th, and 40K at levels 10 times higher than high-density concrete.
Part III: In-situ Measurements
The spectrometer used consisted of an ORTEC 80% p-type germanium detector mounted in a low-background cryostat and powered by an ORTEC Nomad system. The 8192 channel analog-to-digital converter and internal memory accumulated and stored data for read-out via an attached lap-top PC. Resolution was 1.9 keV @ 1333 keV. The spectrometer shielded with 5-10 cm of lead (5 cm in the early measurements and 10 cm in later measurements) and mounted on a cart as shown in Fig. 2. The cart is positioned next to the block for measurements. Note that the distance between the spectrometer and the block is not critical as long as the field of view from the detector sees only the sample.
Characteristic gamma rays from naturally-occurring radionuclides over the energy interval 238-2614 keV were used to generate detector response curves as shown in Fig. 3. The response to characteristic gamma-rays from added activities were determined by interpolation. The curve from Fig. 3 was used to generate a table relating the observed CPM in a

characteristic line to activity. This table was used to determine added activities.
Fig. 3
Most of our observations involved counting times on the order of 1 day, with the exception of background counts (with an additional 5-10 cm lead shielding over the front face of the detector) which lasted several days. With these counting times, we were able to measure activities <0.01 pCi/g for 60Co, <0.1 pCi/g 152Eu, and <0.2 pCi/g 22Na. For activities in the nCi/g range, counting times of only a few minutes are required. Figure 4 shows a spectrum from a sample containing 10 pCi/g of 60Co, adjusted to represent a 5 minute counting time.
Fig. 4
Part IV: Comparison with a Survey Meter
The naturally-occurring radioactivity in concrete can easily provide surface dose rates of several micro-R/hr above background, rates at which a survey meter with a 2.54 cm diameter x 2.54 cm thick NaI crystal can easily measure. Such a meter can be used to survey large areas in short times. For 60Co in high-density concrete, an empirical relationship between dose measured with the survey meter and activity was determined:
Dose rate (mR/hr) = 1.7 * Activity (pCi/g)
For example, 10 pCi/g of 60Co provides approximately 17 mR/hr reading on the survey instrument, an amount easily detectable above the several mR/hr background in the region where we did our measurements. Thus, release criteria of 10 pCi/g for 60Co, for example, can be determined reasonably well with a survey meter. Readings <10 mR/hr, uniform over the surfaces, are probably from naturally-occurring radionuclides alone. Readings above 20 mR/hr, particularly if they are not uniform over the surfaces, almost certainly indicate manmade isotopes. Criteria such as these, combined with judicious application of the detailed characterization described in Part III, can provide inexpensive and sufficient characterization of large samples, such as concrete blocks, for disposition.
CONCLUSIONS
The combination of survey-meter readings and the detailed characterization techniques described above, provide inexpensive and thorough characterization of concrete blocks for reuse or disposal. Following detector calibration, The characterization technique is simple enough that a technician can be trained to perform in-situ measurements in tens of minutes per sample, including analysis. On-the-spot decisions regarding the suitability of a sample for transportation or disposal can then be made. Spectra are retained as permanent records of the radioisotopes in the block.
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VALUATION OF SOLVENT EXTRACTION FOR REMOVAL OF RADIONUCLIDES FROM SOIL AND VEGETATION

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ABSTRACT

A solvent extraction bench-scale treatability study was performed for the U.S. Department of Energy (DOE) Rocky Flats Environmental Technology Site (RFETS) in Golden, Colorado by Resources Conservation Company (IONICS RCC) and Harding Lawson Associates (HLA) using IONIC RCC's patented Basic Extraction Solvent Technology (B.E.S.T.) and additional process technologies for which patents are pending. The bench-scale treatability study was performed in two phases. Each phase was designed to address feasibility study data needs regarding the effectiveness of solvent extraction to remove radionuclides from RFETS soil and vegetation.

Two surficial soil samples and one vegetation sample were collected from three separate areas downgradient of the 903 pad in Operable Unit Number 2 (OU 2). Before treatment, soil sample #1, soil sample #2, and the vegetation sample contained the following plutonium concentrations: 740 picocuries per gram (pCi/g), 1200 pCi/g, and 640 pCi/g, respectively. During the time this treatability study was performed, cleanup levels were not yet established. Therefore, the effectiveness of this technology was primarily evaluated on the basis of the percent of total plutonium-239 and plutonium-240 removed from each soil and vegetation sample following treatment.

The most favorable results of the bench-scale study showed plutonium concentrations in soil sample#1, soil sample #2, and the vegetation sample were reduced to 83 pCi/g (89 percent removed), 102 pCi/g (92 percent removed), and 23 pCi/g (97 percent removed), respectively. The plutonium was concentrated in a residual solvent sidestream that was estimated to be a small fraction of the original contaminated soil and vegetation sample weight. The success of the bench-scale treatability study indicates that plutonium can be removed from RFETS soil and vegetation effectively. After cleanup goals for the radionuclides have been established, a pilot-scale study should be performed to refine the process parameters and costs for full-scale application.

INTRODUCTION

Facilities at Rocky Flats Environmental Technology Site (RFETS) had two main historical missions during the period of operations from 1952 until 1990: producing triggers for nuclear weapons and processing retired weapons for plutonium recovery. The past manufacturing activities have resulted in the contamination of soil and vegetation with radionuclides (i.e., plutonium and uranium), organic compounds, and metals. The samples collected for this specific treatability study were contaminated primarily with uranium and plutonium, and included two soil samples and one vegetation sample from three separate locations downgradient of the 903 pad area.

The 903 pad is located on the south eastern side. This pad was established in 1958 to store plutonium-contaminated oil and uranium-contaminated oil drums outside. The pad covers an area of 113 meters by 120 meters. Deterioration of drums and contamination of soil in the 903

pad area was reported in 1964. From 1967 to 1968 the drums stored at the 903 pad were transported to a building (Building 774) onsite for processing. However, during the years of 1967 through 1968, high winds and heavy rainfall spread contamination to surrounding areas including a ditch near the 903 pad. From 1968 through 1969 installation of a concrete pad to cover the contaminated soil on the 903 pad was started and completed.

Under the Federal Facilities Agreement (1991) for the Rocky Flats Plant DOE has agreed to investigate and remediate contaminated soil at RFETS. Since 1991, an ongoing remedial investigation/feasibility study (RI/FS) has been conducted for the RFETS in an attempt to assess the nature and extent of contamination of all media. As part of the ongoing feasibility studies (FS) for remediation of soil and vegetation at the RFETS, solvent extraction was selected as a potential technology for further evaluation in the FS process. Harding Lawson Associates (HLA) and Resources Conservation Company (IONICS RCC) assisted the Department of Energy (DOE) in evaluating the feasibility of solvent extraction to remove radionuclides from RFETS soil and vegetation samples by performing a bench-scale treatability study. To fulfill the purpose of this treatability study, the following overall program objectives were established: 1) evaluate the effectiveness of solvent extraction on the basis of the percent of total plutonium 239 and plutonium 240 removed from each sample; 2) assess whether the technology could reduce uranium and plutonium concentrations to levels within the treatability study benchmarks (TSBs) range; and 3) identify the near optimum operating parameters including the number of extraction stages required to meet the TSB range. The target radionuclides identified for solvent extraction treatment for this study are presented in Table I with their associated TSBs based on residential hypothetical cancer risk.

Table I

OVERVIEW OF THE SOLVENT EXTRACTION PROCESSES

The specific solvent extraction processes selected for the RFETS bench-scale treatability study were the B.E.S.T. process developed and patented by IONICS RCC and a enhanced solvent extraction process for which patents are pending. The B.E.S.T. process can be used to extract organic contaminants from mixed wastes. The radionuclides can then be extracted using the enhanced solvent extraction process. Both of these solvent extraction processes are described separately in the following paragraphs.

The B.E.S.T. process exploits the unique solvent properties of triethylamine to remove water from soil or sludges and contaminants from both soil and water. In the B.E.S.T. process, triethylamine works to remediate soil or sludges in two basic steps. First, the triethylamine, through its inverse miscibility property (discussed below), effectively removes water from the soil or sludge. Secondly, after soil and sludge moisture is removed, the triethylamine can directly extract organic contaminants from soil particles.

Triethylamine exhibits an inverse miscibility property by being completely miscible with water at or below 60 degrees Fahrenheit (F); however, when heated above 60F, triethylamine and water are only partially miscible. The inverse miscibility property is used to remove water bound to contaminated soil particles by initially extracting contaminated soil with chilled (below 60F) triethylamine to form a homogeneous, single mixture of triethylamine/water/contaminant (primarily

organics). After soil moisture is removed, the efficiency of subsequent extractions, performed at warmer temperatures, is enhanced because the triethylamine is able to achieve complete contact with each component of the contaminated soil.

Before the extraction process is begun, feed material is screened to remove oversize material (i.e.,greater than 1/4-inch diameter at bench scale; greater than 2-inch diameter at full scale) and the pH adjusted to an alkaline condition (pH > 10.5). The feed is then mixed with triethylamine in an extraction vessel until equilibrium is reached, and the solids are removed from the solution phase by settling or, if necessary, by centrifugation. Multiple extraction stages may be required to achieve contaminant removal target levels.

The liquid fraction, a single-phase triethylamine/water/contaminant mixture, is heated to a temperature greater than 60F and two distinct phases from a heavier aqueous phase, and a lighter organic phase made up primarily of organic material and triethylamine. The phases are separated by decantation. The organic contaminant mixture/triethylamine phase contains contaminants initially present in the soil. The triethylamine is recovered from the organic contaminant mixture/triethylamine phase by evaporation, leaving a concentrated organic contaminant fraction. Recovered triethylamine is chilled and recycled for use in subsequent extractions.

The B.E.S.T. process produces water, a concentrated organic contaminant fraction, and treated solids. Ideally, the product water can be conveyed to a water treatment facility where it may require minimal treatment before discharge to the environment. The concentrated organic contaminant fraction, which contains contaminants originally in the influent soil, amounts to a very small fraction (i.e.,0.1 percent) of the influent contaminated soil weight. This fraction may require further treatment (i.e., incineration) if the contaminant levels in the feed soil are high. The treated solids fraction may be returned to the site and possibly revegetated or sent to a disposal facility.The enhanced solvent extraction process combines the pretreatment of contaminated soils or sludges with oxidizing and complexing agents,to remove contaminants (i.e.,metals and radionuclides) from the solid waste materials into the water phase. The contaminants can then be removed from the water phase by using the B.E.S.T. solvent extraction process to preferentially dehydrate the weak aqueous solution to concentrate and capture the contaminants by leaving the contaminants in a dense brine phase.

This paper presents the bench-scale solvent extraction program test design and the results of the bench-scale solvent extraction program performed for RFETS. The bench-scale solvent extraction program included three components: sample preparation, Phase I bench-scale tests, and Phase II bench-scale tests. Each of these components is discussed below.

BENCH-SCALE PROGRAM DESIGN

Sample Preparation

The two soil samples were prepared by screening, blending, and dividing each sample into separate test and analytical samples. The screening process involved using a standard Tyler sieve to remove material greater than 1/4 inch in diameter. The blending process involved splitting the screened sample into two portions, recombining the split sample, and mixing the recombined soil thoroughly. The blending process was performed a minimum of eight times. Following the blending, each separate soil sample was split into analytical and bench-scale test samples. Chemical

analyses of the analytical samples provided baseline chemical characterization data to evaluate the effectiveness of the blending process and plutonium removal efficiencies during bench-scale testing.
The "as received" vegetation sample consisted of two root balls and their accompanying stems and leaves. The stems and leaves were clipped with scissors and set aside. The root balls were rinsed with water to remove most of the soil adhering to the surface of the vegetation. Any floating material was skimmed off the water surface and set aside.
The remaining root balls, stems, leaves, and skimmed material were combined and then ground in a meat grinder. The resulting ground vegetation mixture was blended by hand using the split and recombine technique described above for the two soil samples. Following the blending step, analytical and bench-scale test samples were split. Chemical analysis of these vegetation analytical samples provided baseline chemical characterization data to evaluate the effectiveness of the blending process and to evaluate plutonium-239,240 removal efficiencies during bench-scale testing.

BENCH-SCALE TEST DESIGN

Bench-scale testing was performed in two phases. Each phase was designed to address feasibility study data needs regarding the effectiveness of solvent extraction to remove radionuclides from RFETS soil and vegetation. The bench-scale tests for Phase I consisted of screening tests and solvent extraction tests. The results of the screening tests were used to direct the approach in the Phase I solvent extraction tests. Phase II consisted of solvent extraction tests using the most effective combination of operating parameters identified during the Phase I solvent extraction testing.

Phase I Screening Test Design

Screening tests were performed to evaluate plutonium removal efficiency using several combinations of oxidizing, reducing, and complexing reagents as well as other potential solubilizing agents. Each screening test consisted of adding one or more reagents and conducting one extraction stage. Generally, an extraction stage consisted of adding reagent, mixing the sample with the reagent solution, separating liquids from solids (centrifugation), and recycling solids to the extraction vessel. Gross alpha screening was performed on the extract solution after each reagent addition to evaluate plutonium removal for each reagent or combination of reagents tested. Screening tests showing greater than approximately 20 percent plutonium removal were tested further using a maximum of six extraction stages. These subsequent extraction stages are referred to as solvent extraction tests. The specific technical approach used during the screening tests for soil sample#1 and soil sample#2, and the vegetation sample are described in the following paragraphs.
Fifteen screening tests were performed on soil sample#1 and eleven screening tests were performed on soil sample#2 during Phase I testing. A summary of the test parameters used in the soil sample#1 and soil sample#2 screening tests is provided in Table II. Generally, 100 grams of soil were used for each test. The reagents added to the soil for each test are presented in Table II. The addition of reagent resulted in a liquid-to-solid ratio ranging from 1:1 to 100:1 by weight. The liquid and soil mixture was then agitated for 30 to 60 minutes at temperatures ranging from 34 degrees Fahrenheit (F) to 190F. A small aliquot of extract solution was removed after each reagent addition for gross alpha screening. Screening tests showing greater than 20 percent gross alpha

removal were subject to additional extraction stages, up to a maximum of six extraction stages, with modifications to the screening test parameters. However, with the exception of soil sample#1, Test 1 was a control test and used five extraction stages.

Eight screening tests were performed on the vegetation sample during Phase I testing. A summary of the test parameters used in the screening tests is provided in Table II. Generally, the test parameters were similar to those described for soil samples #1 and 2. For example, 50 grams of vegetation were used per test, reagents listed in Table II were added to the vegetation resulting in a liquid-to-solid ratio of 8:1, the liquid and vegetation mixture was agitated for 30 to 90 minutes at temperatures ranging from 34F to 190F, and a small aliquot of extract solution was removed after each reagent addition for gross alpha screening. Screening tests showing greater than 20 percent gross alpha removal were subject to additional extraction stages with modifications to the screening test parameters.

Table II

Phase I Solvent Extraction Tests

Solvent extraction tests were performed following each soil and vegetation screening test showing greater than 20 percent gross alpha removal. Each solvent extraction test consisted of subsequent extraction stages following the screening test (the screening test being the first extraction stage of the solvent extraction test). Generally, each extraction stage consisted of adding reagent, mixing the sample with the reagent solution, separating liquids from solids (centrifugation), and recycling solids to the extraction vessel. After completion of all extraction stages, triethylamine was added to the separated reagent solution, contaminants were concentrated, and water and triethylamine were recycled. (Triethylamine was used to remove the water from the contaminant solution, allowing the water to be recycled without evaporation or other separation techniques.) A block diagram of the bench-scale test process is presented in Fig. 1.

Fig. 1

Four screening tests for soil sample#1 (Tests 1, 4, 10, and 15) and soil sample#2 (Tests A, B, C, D) were subject to subsequent extraction stages, with modifications to the screening test parameters, as shown in Table III. These subsequent extraction stages are referred to as solvent extraction tests. A step-by-step description of soil sample#1, Test 15 is given below to further clarify the extraction sequence used during Phase I sample testing.

Table III

A 100-gram portion of soil sample #1 was placed in a 1-liter extraction vessel. A solution of hydrogen peroxide, used as an oxidizing agent, was added to the extraction vessel to achieve a liquid-to-solids ratio of 8 to 1 by weight. The solution was agitated for 60 minutes at 150F. After stopping the agitation, the settling characteristics of solids were observed and it was concluded that centrifugation would be required. The solids were then separated from the extract by centrifugation. The extract solution was sampled and analyzed using a gross alpha screening technique to obtain an estimate of the extraction efficiency of the peroxide solution.

Citric acid, used as a complexing agent, was then added to the extraction vessel. The mixture was again agitated for 60 minutes at 160F. The solids were separated from the extract by centrifugation. The extract solution,

referred to as interstage extract solution, was analyzed using a gross alpha screening technique to obtain an estimate of the extraction efficiency of the peroxide/citric acid solution. This extract solution, free of suspended solids, was sampled and later analyzed for isotopic plutonium. The extracted solids, referred to as interstage solids, were sampled and later analyzed for isotopic plutonium.

The above extraction procedure was repeated three more times for Test 15, starting with the addition of hydrogen peroxide to the solids and liquid remaining in the extraction vessel, for a total of four extraction stages. The final treated solids and final extract solution were later analyzed for total uranium and isotopic plutonium. After the final extraction stage, a composite extract solution was formed by combining the extract solution from each extraction stage. Triethylamine was then added to the composite extract solution to concentrate the contaminants to a minimal volume. (Addition of triethylamine forms a two-phase system; a light phase containing triethylamine and water, and a heavy phase containing the contaminants and a small amount of water.) The heavy phase was then analyzed for isotopic plutonium.

Four solvent extraction tests (Screening Tests V-1, V-2, V-3, and V-7) were performed with modifications to the screening test parameters, as shown in Table III. The step-by-step procedure used for the vegetation sample was similar to that discussed for soil sample #1.

Phase II Solvent Extraction Tests

Phase II solvent extraction tests were performed using the most effective combination of oxidizing/reducing agents, complexing agents, triethylamine, extraction time, and extraction temperature identified during Phase I solvent extraction testing. The process operating parameters used in Phase II testing are presented in Table IV for soil sample #1, soil sample #2, and for the vegetation sample. The sample size and number of extraction stages were increased during Phase II testing as compared to Phase I testing (sample sizes were doubled during Phase II testing and 12 extraction stages were conducted during Phase II instead of the 3 or 4 stages used during Phase I testing). Generally, each extraction stage consisted of adding a reagent, mixing the sample with the reagent solution, separating liquids from solids (centrifugation), and recycling solids to the extraction vessel. After completion of all the extraction stages, triethylamine was added to the separated reagent solution, contaminants were concentrated, and water and triethylamine were recovered. (Triethylamine was used to remove the water from the contaminant solution, allowing the water to be recycled without evaporation or other separation techniques.) A block diagram of the solvent extraction bench-scale test process is presented in Fig. 1. Solvent extraction testing procedures used during Phase II for soil sample #1, soil sample #2, and the vegetation sample are described in further detail in the following paragraphs.

Table IV

Two solvent extraction tests were conducted during Phase II testing on soil sample #1 and soil sample#2. The test parameters used for each test are presented Table IV. A step-by-step description of one of the two tests for soil sample#1 is given below to further clarify the extraction sequence used during Phase II sample testing.

A 200-gram portion of soil sample #1 was placed in a 1-liter extraction vessel. A solution of hydrogen peroxide, used as an oxidizing agent, was added to the extraction vessel to achieve a liquid-to-solids ratio of 8

to 1 by weight. The solution was agitated and heated to 190F. Citric acid, used as a complexing agent, was then added to the extraction vessel. The mixture was agitated for 60 minutes at 190F. After stopping the agitation, solids settling characteristics were observed and it was concluded that centrifugation would be required. The solids were then separated from the extract solution by centrifugation. This extract solution, free of suspended solids, was analyzed for isotopic plutonium and total uranium.

The above extraction procedure was repeated 11 more times, starting with the addition of hydrogen peroxide to the solids and liquid remaining in the extraction vessel, for a total of 12 extraction stages. The final treated solids were analyzed for total uranium and isotopic plutonium. After the final extraction stage, two composite extract solutions were formed by combining the extract solutions from extraction stages 1 through 6 and the extract solutions from extraction stages 7 through 12. An aliquot of each of these two composite samples was analyzed for isotopic plutonium and total uranium. These two composite extracts were then combined to form a single extract solution composite. Triethylamine was then added to the extract solution composite to concentrate the contaminants to a minimal volume. (Addition of triethylamine forms a light phase containing triethylamine and water, and a heavy phase containing the contaminants and a small amount of water.) The heavy phase, produced by adding triethylamine to the composite extract solution, was then analyzed for isotopic plutonium and total uranium. The water recovered from the extract solution was sampled and later analyzed for isotopic plutonium and total uranium.

One solvent extraction test was performed on the vegetation sample using the test parameters shown in Table IV. The step-by-step procedure used for the vegetation sample was similar to that for soil sample #1.

RESULTS AND DISCUSSION OF THE BENCH-SCALE SOLVENT EXTRACTION PROGRAM

The Phase I and Phase II bench-scale tests produced both process and analytical results. Process test results for each soil and vegetation sample included approximate values for extraction temperature, extraction time, solids settling and centrifugation characteristics, oxidation/reduction agent addition, complexing agent addition, feed to reagent (i.e., oxidation/reduction and complexing agent) ratios, and solvent to reagent ratios. Phase I screening test showing greater than 20 percent plutonium removed were tested further using a maximum of six extraction stages and some modification to process parameters. The additional testing performed on these screening test samples were called the Phase I solvent extraction tests. These analytical and process parameter results from the Phase I bench-scale solvent extraction testing were then used to identify the process parameters to be used during Phase II testing. In addition, the analytical results provided data for mass balance calculations. Results from the three components of the bench-scale testing (sample preparation, screening tests, and solvent extraction tests) are summarized in the following paragraphs for soil sample#1, soil sample#2, and the vegetation sample.

Sample Preparation Results

Following the sample screening, blending, and splitting, the chemical characterization analytical samples were submitted to the laboratory for chemical analysis to provide data to evaluate the effectiveness of the blending process. The results of the plutonium characterization analyses for each sample are presented in Table V, which presents the analytical

results for both the Phase I and Phase II feed samples. The variations in the vegetation and soil feed concentrations may be attributed to analytical variance and the inherent heterogeneity of the sample matrix. A statistical evaluation of the chemical characterization analytical results was performed and the results are presented in Table V.

Table V

Phase I Solvent Extraction Tests

The process data collected during Phase I bench-scale testing of soil sample #1 are summarized as follows:

The extraction temperatures varied from 34F to 190F.

The extraction times varied from 30 minutes to approximately 14 hours.

Solids settling times of up to 30 minutes were tested and centrifugation was required.

The oxidation/reduction and complexing agents tested are presented in Table VI.

The ratio of feed to reagent (i.e., oxidation/reduction and complexing agents), expressed on a weight-to-weight basis, varied from 1:1 to 1:100.

The ratio of solvent to reagent, expressed on a weight-to-weight basis, varied from pure solvent to a ratio 19:1.

Table VI

Table VII

The results of plutonium 239,240 and total uranium analyses conducted on soil sample#1 and soil sample#2 feed, interstage (i.e.,first extraction, second extraction, etc.), and final treated solids from Phase I solvent extraction tests are presented in Table II. The analytical results show that in Tests 10 and 15, plutonium-239,240 was reduced from a mean feed concentration of 740 picocuries per gram (pCi/g) to 86 pCi/g and 95 pCi/g, respectively, in the final treated solids. The analytical results for soil sample #2 indicate that in Tests A and C, plutonium-239,240 was reduced from a mean feed concentration of 1200 pCi/g to 170 pCi/g and 180 pCi/g, respectively, in the final treated solids.

The process data collected during Phase I bench-scale testing of the RFETS vegetation sample are summarized as follows:

The extraction temperatures varied from 34F to 190F.

The extraction times varied from 30 to 90 minutes.

Solids settling times of up to 30 minutes were tested and centrifugation was required.

The oxidation/reduction and complexing agents tested are presented in Table VI.

The ratio of feed to reagent (i.e., oxidation/reduction and complexing agents), expressed on a weight-to-weight basis, was 1:8.

The ratio of solvent to reagent, expressed on a weight-to-weight basis, was 5:1.

The results of plutonium-239,240 and total uranium analyses conducted on feed, interstage (i.e.,first extraction, second extraction, etc.), and final treated solids from Phase I solvent extraction vegetation testing are presented in Table II. The analytical results show that in Test V-2, plutonium-239,240 was reduced from a mean feed concentration of 640 pCi/g to 87 pCi/g in the final treated solids. Because the vegetation samples could not be ground to a small uniform particle size, the observed variability in the interstage solid results may reflect heterogeneities between the small sample aliquots used.

SUMMARY OF RESULTS AND CONCLUSIONS

Results of the solvent extraction treatability study using the enhanced solvent extraction process indicate that significant removal of plutonium-239,240 from RFETS soil and vegetation samples was achieved. The most favorable Phase II test results indicate plutonium concentrations in soil sample#1, soil sample#2, and the vegetation sample were reduced to 83 pCi/g (89 percent removed), 102 pCi/g (92 percent removed), and 23 pCi/g (97 percent removed), respectively. Additionally, Phase II test results indicate plutonium was concentrated in a residual solvent sidestream that was estimated to be a small fraction of the original contaminated soil and vegetation sample weight. After cleanup goals for the radionuclides have been established, a pilot-scale study is recommended to refine the process parameters and costs for full-scale application.

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GLOSSARY OF TABLE TERMS

<	Less than
C12H27O4P	Tributyl phosphate
C6H8O7	Citric acid
H2O2	Hydrogen peroxide
HNO3	Nitric acid
min	Minute
N	No
Na2CO3	Sodium carbonate
Na2S2O4	Sodium dithionite
Na3C6H5O7	Sodium citrate
pCi/g	Picocuries per gramtemp Temperature
TSBs	Treatability study benchmarks
Y	Yes
F	Degrees Fahrenheit
g/g	Micrograms per gram

23-28

EVALUATING A REAL-TIME BERYLLIUM MONITOR AND THE IMPLICATIONS FOR ITS USE

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ABSTRACT

This study shows that a laser-induced breakdown spectroscopy (LIBS) monitor is effective at identifying peak times of worker exposure to beryllium and enables the industrial hygienist to better evaluate peak exposure levels of airborne beryllium.

Los Alamos National Laboratory (LANL, Laboratory), a US Department of Energy laboratory managed under contract to the University of California, has developed an instantaneous read-out instrument, which is based on laser-induced breakdown spectroscopy (LIBS) and indicates the airborne concentration of beryllium. The principal investigators (PIs, investigators) tested the instrument during operations at LANL's Beryllium Operation Facility. To evaluate the real-time instrument, we compared and contrasted data from current industrial hygiene monitoring methods to the instantaneous data from LIBS.

The investigators took short duration side-by-side personal samples by attaching the LIBS monitor onto one lapel of a worker's personal protective clothing and a standard personal industrial hygiene sampling train onto the other. Each sample collected by the LIBS monitor covered a 30-second interval. The computer then summed all intervals and also stored the data electronically for further study. The sum of all the 30-second intervals indicates the steps of the beryllium operation that are associated with peak personal exposures, providing an "exposure profile." Using the relative peaks and lows of real-time data, the investigators correlated work practices and control measures. In addition, we used the LIBS device to identify "hot spots" or potential areas of increased exposure. Our assessment of procedures at the Beryllium Operation Facility verified that instantaneous read-out equipment facilitates exposure assessments and control. n sulfide, carbon monoxide, cyanide, mercury vapor, and a few other substances. Their use is generally limited to intermittent evaluations to determine if respiratory protection should be used and when to initiate personal monitoring. Colorimetric tubes are also used for screening, but not generally used to determine an exposure profile, because it is difficult to take sufficient samples and the time lag in between samples ranges from one minute to 20 minutes.

This study shows that a laser-induced breakdown spectroscopy (LIBS) monitor is effective at identifying peak times of worker exposure to beryllium and enables the industrial hygienist to better evaluate peak exposure levels of airborne beryllium.

Beryllium is a highly toxic chemical with an occupational exposure limit (OEL) of 2g/m3 during an eight-hour workday (1,2). OELs also exist for the 30-minute ceiling of 5g/m3 and the instantaneous peak limit of 25g/m3. The proposed pathogenesis of chronic beryllium disease (CBD) is that certain individuals can develop a cell-mediated immune response to beryllium that has been associated with the development of a pulmonary granulomatous reaction. One CBD case revealed that a facility relied unduly on averaged beryllium concentrations and disregarded the significance of peak exposures that were hidden in the mean exposure figures. Excessive exposures that last hours, days, or weeks may cause CBD; peak exposures must be taken into account (3).

Established industrial hygiene sampling technology cannot measure instantaneous peak or ceiling OELS for many air contaminants, such as metals. If real-time instrumentation is unavailable, costly generation of a large number of samples is necessary to determine peak exposure times. Beryllium sample analyses range from $25 to $100 per sample. Large numbers of samples limit an in-house laboratory's availability to provide

analytical support for industrial hygiene evaluation of other processes. Even if a large number of standard industrial hygiene samples were taken, it remains difficult to determine what part of a facility's operation is creating the different exposure levels. The standard sampling method only provides average concentrations over the sampling period; therefore, one can make only an estimate of the actual time of highest exposure.

BACKGROUND

LANL scientists developed the LIBs monitor (see Fig. 1) in the early 80s (4). LIBS can rapidly detect airborne or surface particles. In the LIBS method, powerful laser pulses are focused to generate microplasma. Material in the plasma is vaporized and reduced to its elemental constituents, which are electronically excited. The LIBS monitor can identify the unique spectral signatures of atoms by spectrally resolving and recording the energy radiated as the excited atoms drop to lower or ground-state energy levels.

Fig. 1

METHOD

Investigators used a portable LIBS monitor equipped with a focused high-powered Q-switched neodymium:yttrium-aluminum garnet (Nd:YAG) laser. The laser beam induces a dielectric breakdown of air in the sample stream and forms a plasma spark. Beryllium particles in the region of the spark are vaporized and the beryllium is excited. A fiber-optic cable focused on the sample port collects the light emitted with excitation decay. The light is directed to the entrance slit of a small spectrograph tuned to the most intense beryllium (II) line of 313.1nm. In this study, a photomultiplier tube detected the beryllium emission and an analog-to-digital processor located inside a computer integrated the signal and digitized the resulting voltage. The resulting signals were stored in the computer's memory. The laser has a repetition rate of 10 hertz with a sample interval of 30 seconds.5 The laser spark volume was 0.03 cm3 and the output energy was 100mJ/pulse. The computer controlled all instrument operations.

A plastic shipping container completely contained the instrument to reduce potential contamination. Figure 2 shows a photograph of the LIBS monitor and the spray chamber. The computer was kept in an area of low contamination and attached to the monitor with a 30-foot cable. A 0.95-cm-diameter hose approximately 10-feet long, which had a flow rate of 10.7 Lpm, was attached to the LIBS monitor. The computer displayed two real-time histograms: a "shot" histogram for the 30-second intervals and an "average" histogram of the accumulated sums of the 30-second interval intensities. All histograms are stored electronically for future study.

Fig. 2

PIs studied the beryllium powder-spray operation by conducting seven sampling events. We collected side-by-side samples using both the LIBS monitor and standard industrial hygiene technique. For most of the sampling events, the LIBS monitor was inside the beryllium-limited access area, with the computer located outside in the air-lock room. Work inside this area requires full personal protective equipment that includes respiratory protection.

We conducted separate sampling events that took place during various operations: cleaning the spray chamber, spraying parts, and conducting equipment repair and troubleshooting. In Fig. 2, one can see the spray chamber (behind the LIBS monitor), a vessel approximately four-feet in depth by three-feet in diameter. The chamber is closed during spray

operations and filled with an inert gas. The chamber must be opened for workers to retrieve the sprayed part, clean the inside, or repair the equipment.

The investigators collected industrial hygiene samples using pre- and post-calibrated pumps operating at approximately 3.5 Lpm. We analyzed samples using inductively coupled plasma according to the National Institute for Occupational Safety and Health 7300 method (6). From the seven sampling events, the PIs could use eight air sample results to correlate with the LIBS monitor's response.

RESULTS

Of the seven sampling events, only four generated standard industrial hygiene samples. During two runs, the LIBS instrument had unreliable responses that were possibly due to either overheating from the laser enclosure or radio frequency interference from the plasma spray operation. Results from another run, submitted to an analytical laboratory accredited by the American Industrial Hygiene Association, did not provide valid results as evidence of a blind quality control sample.

For the first sampling event, the principal investigators (PIs, investigators) attempted to correlate area industrial hygiene samples with the LIBS monitor response. The sample location we selected held the highest potential for airborne beryllium. However, the LIBS monitor was not responding at the location of the area sampled; therefore, we conducted personal samples.

We obtained exposure profiles for sample operations using the "average" histogram and field notes. Two types of data are available for an exposure profile. The instrument logs the sum of intensities and the sum of the number of "shots" with response-above-background for the 30-second intervals. The latter is used for most figures in this paper to display the exposure profile data. Figure 3 shows a sampling event conducted when a worker was cleaning the inside of the spray chamber. The investigators did not collect all of the personal exposure because the LIBS monitor was sampling areas for the initial part of the sampling event. From this data, we found clear evidence that the spraying of vacuumed parts with a cleaning agent and wiping them creates airborne beryllium. We concluded this cleaning method should be avoided and replaced by either spraying a towel with the cleaning agent or using a wet sponge.

Fig. 3

Figure 4 shows a sampling event that occurred when workers were troubleshooting and repairing equipment. From the exposure profile, one may easily learn what parts of the operation create the highest airborne concentrations: airborne beryllium levels were relatively high for opening the chamber, working deep inside the chamber, removing the plasma spray torch, and wiping the surface of parts. Airborne beryllium levels were relatively low for repairing the equipment, vacuuming inside the chamber, and installing parts inside the chamber. The data show clearly that engineering controls do not capture all particles when workers initially open the chamber, parts must be vacuumed prior to their manipulation, and a vacuum attachment should be used to reduce the amount of time or eliminate working inside the chamber.

Fig. 4

Another sampling event showed that the workers themselves were generating airborne beryllium when they moved about in beryllium-contaminated clothing. Consequently, this study resulted in a changed work practice; workers now change their visibly contaminated clothing.

Airborne beryllium concentrations and sample times for the industrial hygiene samples range from 5.7 g/m3 to 105 g/m3 and 11 minutes to 62 minutes respectively. Figure 5 shows one sampling run in which standard industrial hygiene sample results were imposed on exposure profiles from the LIBS monitor. From this study, we observe correlation of LIBS response and standard industrial hygiene sampling does not result in a straight-line. Many factors could have contributed to the differences: concentration variability from one lapel to another; overheating of equipment; particle size effects; and the 2g upper particle size limit for total vaporization of the sample (7).

Fig. 5

CONCLUSION

The value of this study is that it proves the LIBS instrument detects airborne beryllium in the workplace. The instrument identifies operations of higher exposure potential associated with higher concentration levels. Use of the LIBS monitor demonstrates that some established, accepted work practices produce unexpected exposure situations. Conventional industrial hygiene sampling methods would not have shown the level of detail necessary to determine when exposures occur. Because of the data collected by this real-time instrument, the investigators identified work practices producing emissions which were than easily modified, therefore reducing potential worker exposure to a highly toxic chemical.

Regardless of the contaminant, most exposures in a non-manufacturing production line setting occur during short intervals. Personal experience shows that while screening tests may show higher airborne concentrations, a full-shift sample generally does not approach the OEL. Therefore, one could deduce that most of the exposure comes from higher short-term, intermittent levels that are not adequately evaluated by an eight-hour sample.

In general, there has been limited use of real-time monitors to create exposure profiles of work operations. Limited use could be due to the lack of available contaminant-specific equipment, difficulty in storing or logging results, time commitments, or undue reliance on averaged standard industrial hygiene sampling methods. However, this study makes it clear that even though an instrument may not be fully quantitative, real-time monitors can still be a very valuable tool. The industrial hygiene community should increase its use of real-time monitors in conjunction with personal sampling to obtain better evaluations of peak levels.

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23-30

THE CHEMICAL EXPOSURE ASSESSMENT PROGRAM AT LOS ALAMOS NATIONAL LABORATORY: A RISK BASED APPROACH

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ABSTRACT

The University of California Contract and DOE Order 5480.10 require that Los Alamos National Laboratory (LANL) perform health hazard assessments/inventories of all employee workplaces. In response, LANL has developed the Chemical Exposure Assessment Program. This program provides a systematic risk-based approach to anticipation, recognition, evaluation and control of chemical workplace exposures. Program implementation focuses resources on employee exposures with the highest risks for causing adverse health effects. Implementation guidance includes procedures for basic characterization, qualitative risk assessment, quantitative validation, and recommendations and reevaluations. Each component of the program is described. It is shown how a systematic method of assessment improves documentation, retrieval, and use of generated exposure information.

INTRODUCTION

The Los Alamos National Laboratory (LANL) Chemical Exposure Assessment (CEA) Program provides a systematic, risk-based approach to the anticipation, recognition, evaluation, and control of chemical workplace exposures. The program's purpose is to ensure that employees are not adversely affected from exposure to chemical stressors in the workplace. Primary functions of the program are the qualitative assignment of risk and the quantitative validation of potential chemical workplace exposures. Because workplace exposures cannot be totally eliminated, this program strives to control hazards to an acceptable level and be an effective primary prevention tool against occupational injuries and illnesses.

DEFINITIONS

Exposure Group (EG) A group consisting of an employee(s), job assignment(s)/task(s), and chemical stressor(s), such that exposure monitoring of one individual within the group is representative for all individuals within the same group.

Exposure Rating (ER) A numerical value between 0 and 4 that represents the qualitative employee exposure risk to a chemical stressor. The exposure rating is based on the level of hazard control, the frequency and duration chemical use, and the ability of a chemical to become airborne.

Health Effects Rating(HR) A numerical value between 0 and 4 that represents the severity of a chemical's health effect and/or its toxicity.

TWA Time-Weighted Average: The time-weighted average exposure concentration for a normal 8-hour workday and a 40-hour workweek, to which nearly all workers may be repeatedly exposed, day after day without adverse health effects.

STEL Short-Term Exposure Limit: A 15-minute TWA exposure which should not be exceeded at any time during a workday.

Ceiling Value An exposure concentration which should not be exceeded during any part of the working exposure.

PROGRAM IMPLEMENTATION

The components of the LANL CEA Program (described in the ensuing subsections) follow the AIHA strategy for occupational exposure assessment (1). Figure I graphically illustrates the relationship of each component.

Fig. 1

Basic Characterization

The first component in the LANL CEA process is basic characterization of the workplace, work force, and occupational chemical stressors. Workplace characterization highlights operations, activities, and areas with potential chemical exposure pathways. It provides information on process flow and process chemistry. Information is gathered on routine operating conditions (e.g., temperature, pressure, concentration), types of process equipment, types of process controls, and potential exposure considerations. Work force characterization involves gathering employee demographic information for an understanding of how employees interact with operations, processes, or tasks. Chemical stressors are characterized so that the industrial hygienist has sufficient information on frequency, duration, personal protective equipment, and controls to make informed qualitative decisions on exposure risk.

The outcome of Basic Characterization is a complete demographic inventory of employees, their job assignments/tasks, the chemical stressors that they are exposed to, and a description of the processes that they perform. Exposure Groups (EGs) are formed using the information gathered during basic characterization.

Qualitative Risk Assessment (QRA)

QRA is performed on each chemical stressor. The purpose of QRA is to identify the degree of exposure risk posed by each chemical within an EG. The element of the QRA process are the Health Effects Rating, Frequency of Use Rating, Level of Control Rating, Dispersion Rating, and Exposure Rating (2). It is important to note that decisions regarding the degree of exposure risk and its application to the CEA Program are often subjective and professional judgement by competent industrial hygienists is mandatory.

Health Effects Rating

All chemicals are given a Health Effects Rating (HR). The HR is a numerical value ranging from zero (low) to four (very high). The HR is used to define the toxicity or potency of target organ response to

hazardous material exposure, using criteria developed at LANL (3). The HR provides a first cut evaluation of the degree of risk for an adverse health effect upon exposure to a chemical stressor. The objective is that if a chemical has a low risk of conveying an adverse health effect then less scrutiny is placed on it. Thus, chemicals carrying an HR of 0 (zero) or 1 are documented and automatically assigned an Exposure Rating of 0 (zero). An exception to this occurs when professional judgement dictates that factors not adequately reflected in the HR (e.g. dose, oxygen deficiency, flammability, reproductive or mutagenic properties) produce an adverse exposure scenario. In this instance, chemical stressors remain in the QRA process for further exposure risk evaluation. Chemical stressors carrying an HR of 2, 3, or 4 automatically continue in the QRA process.

Frequency of Use Rating

The Frequency of Use Rating reflects the degree of workplace chemical use. Duration and other chemical use factors are taken into account and used with professional judgement to assign the appropriate Frequency of Use Rating. Table I is used to assign Frequency of Use Ratings. Chemical stressors assigned a Frequency of Use Rating of 2 or less are assumed to carry a reduced risk for an adverse exposure. This qualitative assignment is documented and the chemical stressor is given an Exposure Rating of zero. Again, professional judgement must be used to determine whether an adverse exposure could occur, even at low Frequency of Use Ratings. If so, the chemical remains in the QRA process for additional exposure evaluation. All chemical stressors carrying a Frequency of Use Rating greater than 2 continue in the QRA process.

Table I

Level of Control Rating

The extent an exposure is prevented or reduced through the use of engineering controls, work practices, or personal protective equipment is evaluated with the Level of Control Rating. As shown in Table II administrative controls and personal protective equipment are not assumed to be adequate substitutes for engineering controls. Even so, professional judgement is used to adjust the Level of Control Rating when administrative and/or personal protective equipment is effectively used in association with engineering controls. All chemicals carrying a Level of Control Rating equal to 1 are documented and are given an Exposure Rating of zero. Those chemicals having a Level of Control Rating greater than 1 continue in the QRA process.

Table II

Dispersion Rating

The Dispersion Rating reflects the ability of a chemical to become airborne and available to the inhalation pathway in the work environment. Table III defines the criteria used to assign the Dispersion Rating for each chemical. All chemical stressors making it to this step in the QRA process are assigned a Dispersion Rating and are applied to the ER Matrix discussed below.

Table III

Exposure Rating (ER)

An ER is a numerical representation of the degree of exposure risk to a chemical stressor. An ER of 1 through 4 is assigned to those chemical stressors that have been applied to each step in the QRA process. The ER is determined by use of Eq. (1) and its application with the ER matrix shown in Table IV.

Eq. 1
Where:
F = Frequency of Use Rating
L = Level of Control Rating
D = Dispersion Rating
Table IV
VALIDATION
A required follow up to QRA is quantitative validation of assigned ERs. Elements of the validation process are described below.
Sampling
A sampling strategy is developed based on a chemical stressor's ER. Table V shows how the ER dictates the number of annual samples required for quantitative validation. Exposure scenarios in LANL's research and development environment do not always allow for classical sampling strategies. Thus, professional judgement must be used to determine a practical approach.
Table V
Monitoring
All monitoring is conducted in accordance with LANL policy and with NIOSH or OSHA sampling and analytical methods. When a NIOSH or OSHA sampling and analytical method is not available for a chemical stressor, a chemist in the analytical laboratory is consulted for an alternative collection strategy.
Interpretation and Decision Making (2)
This element of the validation process provides statistical insight into the significance of exposure measurements collected. For the purposes of this paper all exposure distributions are assumed to be lognormal.
Centering Value
A Centering value is used to measure the center of an exposure distribution. For lognormal exposure distributions the centering value used is the geometric mean.
Tolerance Level Value
The Tolerance Level Value is calculated to measure the variability in the exposure distribution. This statistical tool reflects the percent of the expected exposure values that are below a set level. For example, the 90% Tolerance Level Value is the exposure value at which 90% of the exposure opportunities are below. Thus, the calculated geometric 90% Tolerance Level Value is that exposure level at which 90% (i.e., 900 out of 1,000) exposure values are likely to be at or below.
Eq. 2
Where:
Tol90 = 90% Tolerance Level
GM = Geometric mean
GSD = Geometric standard deviation
1.28 = Number of standard deviation units corresponding with the 90th percentile of the distribution
Confidence Level
The Confidence Level is a measure of the distribution of exposure values around the Centering Value. A 95% two-tailed Confidence Level is determined using the calculated standard errors of the exposure distribution.
Eq. 3
Where:
GSD = Geometric standard deviation of the distribution of exposure

values;
n = Number of samples in the data set.
A 95% two-tailed Confidence Level is now calculated:
Eq. 4
Eq. 5
Using the statistical tools above, this discussion now answers the following question:
Q: What is the exposure level of an employee in an exposure group?
A: The Centering value provides an estimate of the most likely exposure level. The Tolerance Level Value and Confidence Level provide an estimate of how extreme the exposures can be.
Validation of ERs
Qualitative ERs can now be validated by application of the 90% tolerance limits to the appropriate exposure rating matrix shown in Tables VII & VIII. If an exposure rating is not validated then errors in the QRA process are investigated.
Table VI
Table VII
RECOMMENDATIONS AND REEVALUATION
Recommendations
LANL CEA Program recommendations are based on validation of a chemical's ER. An exception exists when an imminent hazard is identified during basic characterization. In this case, a recommendation for increased control measures is made immediately, prior to assignment and validation of an ER. In general, typical program recommendations identify the need to alter existing control methods and the level of effort required for chemical sampling and monitoring. In a broader usage of LANL CEA Program information, recommendations are made to aid in the performance of reproductive health hazard assessments, carcinogen use hazard assessments, personal protective equipment hazard assessments, and determinations for the need of employee medical surveillance.
Reevaluation
It is the intent of the LANL CEA Program to reevaluate Exposure Groups (EG) annually. This periodicity may be insufficient for some EGs and too frequent for others. Thus, there are three guidelines which are followed to determine the necessity of a reevaluation:

Awareness by a field industrial hygienist of a new EG or change in status or scope of an existing one.

The presence of a highly dynamic EG. The more dynamic the EG the greater the periodicity of an exposure assessment.

Three years have past without the performance of an exposure assessment.

LIMITATIONS
The LANL CEA Program is limited to evaluation of normal operating conditions. Off normal occurrences are more complex and beyond the scope of this paper. This program utilizes a single stressor model for exposure assessment. This means that an overall ER does is not calculated for simultaneous exposure to multiple chemicals. In the case of exposure to a mixture of chemicals having additive (similar toxicological effects) or independent effects the exposure is documented and a professional industrial hygienist determines an appropriate characterization strategy. Also, ERs applied to sensitized individuals may not adequately reflect their risk for an adverse response to a chemical exposure. During Basic Characterization, if a sensitizer is detected then the information is

documented and a professional industrial hygienist determines the appropriate action to take to minimize the exposure to an acceptable level.

CONCLUSION

The LANL CEA program shows that chemical exposure assessment can be performed in a systematic fashion. Using a risk based approach aids in prioritizing time and resources to areas where they are needed most. The validation component of the LANL CEA Program gives the industrial hygienist a template for determining the type and degree of exposure sampling required for a given EG. Through the use of computer automation CEA information can be rapidly shared among the many program stakeholders. Occupational medicine can use employee exposure information as a tool for primary prevention of injury and illnesses. Exposure Ratings can be used to justify the need for increased control measures to operational and line management. Regulators can be shown the risks of chemical exposure, where they are occurring, and what employees are receiving them.

It is this author's opinion that exposure assessment is not a new concept, but simply a systematic way to apply the fundamental principles of industrial hygiene. Too often, industrial hygiene programs are forced to operate in a reactive mode to satisfy the multitudinous needs of their customers. This hinders program consistency and makes retrieval and historical use of exposure assessment information elusive and difficult to decipher. A standardized approach, like the LANL CEA Program, addresses these problems and helps an industrial hygiene program proactively manage exposure assessment information.

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UNDERGROUND RADIOACTIVE WASTE TANK REMOTE INSPECTION AND SAMPLING*

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ABSTRACT

Characterization is a critical step in the remediation of contaminated materials and facilities. Severe physical- and radiological-access restrictions made the task of characterizing the World War II-era underground radioactive storage tanks at the Oak Ridge National Laboratory (ORNL) particularly challenging. The innovative and inexpensive tank characterization system (TCS) developed to meet this

challenge at ORNL is worthy of consideration for use in similar remediation projects.

The TCS is a floating system that uses the existing water in the tank as a platform that supports instruments and samplers mounted on a floating boom. TCS operators feed the unit into an existing port of the tank to be characterized. Once inserted, the system's position is controlled by rotation and by insertion and withdrawal of the boom. The major components of the TCS system include the following:

- boom support system that consists of a boom support structure and a floating boom,
- video camera and lights,
- sludge grab sampler,
- wall chip sampler, and
- sonar depth finder.

This simple design allows access to all parts of a tank. Moreover, the use of off-the-shelf components keeps the system inexpensive and minimizes maintenance costs.

The TCS proved invaluable in negotiating the hazards of ORNL's Gunite and Associated Tanks, which typically contain a layer of radioactive sludge, have only one to three access ports that are usually only 12- or 24-in. in diameter, and range from 12 to 50 ft in diameter. This paper reviews both the successes and the difficulties encountered in using the TCS for treatability studies at ORNL and discusses the prospects for its wider application in remediation activities.

INTRODUCTION

As the contractor for the Oak Ridge National Laboratory's (ORNL's) Remedial Investigation/Feasibility Study (RI/FS), Bechtel is responsible for developing plans and procedures, conducting field investigations, and reporting characterization results. The 8,800 acres of the ORNL reservation have been divided into 20 Waste Area Groupings (WAGs) with approximately 250 known Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA)-contaminated sites and RCRA solid waste management units. In one of these WAGs, Bechtel developed and deployed a new robotic system to characterize underground radioactive waste storage tanks. These underground storage tanks were built in the 1940s of Gunite, a sprayed-on concrete frequently used for in-ground swimming pools. Detailed characterization of these tanks was required to provide information to an ongoing Treatability Study.

The tanks contain sludges and liquids. They have been used to collect, neutralize, store, and transfer radioactive and/or hazardous chemical wastes since the beginning of ORNL operations in 1943. A significant portion of the sludge in this tank farm was recovered during a sludge mobilization campaign in 1983. That sludge removal project was stopped before the tanks were completely cleaned; the present work is to characterize the remaining contents. These tanks are either 25 or 50 feet in diameter and have dome-shaped roofs up to 18 feet high. They are buried under approximately 6 feet of soil. There are between one and three available entry risers for each tank, with internal diameters of either 12 or 24 inches.

The needs of the ongoing Treatability Study program were as follows:

1. selectively retrieve samples of sludge and objects (from areas throughout the tanks) to determine waste properties for remediation system designs;

2. measure the depth of the sludge to refine volume estimates for waste management planning; and
3. videotape the inside of the tanks for waste management and remediation engineering planning.

To perform these studies, we designed and fabricated a number of unique tools and a deployment system to pass through 24-inch, and in some cases 12-inch, manholes and enter the tank. Seven Gunite tanks were characterized using the newly developed ORNL Tank Characterization System (TCS). The ORNL TCS uses simple principles and off-the-shelf technology to solve difficult problems. It includes four major subsystems: the floating boom and support system, the video camera and lights, samplers including a grab sampler and a concrete chip sampler, and a sonar depth finder. These systems aided in determining the locations, quantities, and compositions of materials in the tanks as the basis for remediation planning. With the TCS, we were able to inspect the tank above and below the water, map the sediments in the bottom of the tanks, take samples of the sediments or debris, and take small samples of the Gunite walls of the tanks. The TCS tools reach up to 50 feet horizontally away from the manhole.

FLOATING BOOM AND SUPPORT SYSTEM

This is the structural portion of the TCS and consists of a flexible floating boom and boom support (Fig. 1). It is used to transport tools and characterization instruments to any lateral location in the tank. The boom support is fabricated from aluminum and consists of a base, a "lazy Susan," and support channels that extend down into the tank through the riser. The boom support can be installed by two men in less than 30 minutes at the manhole. In our application, the base was simply clamped to the riser flange at the tank manway. The support channels can be lowered at least 25 feet into the manhole, and the "lazy Susan" can be rotated 360 degrees.

The floating boom is constructed from commercially available plastic chain. The chain is rigid laterally and is built to roll in only one direction. Foam is added for flotation, and the boom is deployed by sliding the chain down the support
channels to the water surface. By rotating and inserting or withdrawing the plastic boom, any part of the tank can be reached. By measuring the length of boom inserted and the angle of deployment, the operator can pinpoint the location of the tool at the end of the floating boom.

Fig. 1

VIDEO CAMERA SYSTEM

The TCS includes a custom-mounted video camera that can inspect either above or below the waterline. All camera system functions are controlled from a control box outside of the immediate work area. The camera system consists of the following sub-systems: camera with built-in pan-and-tilt, waterproof housing, and lights (Fig. 2).

The video camera is a standard resolution color camera with 8 times zoom, manual focus, 1 lux light rating, and auto iris. The pan-and-tilt unit is an integral part of the camera with 180 degrees pan and 180 degrees tilt envelope. This is an inexpensive off-the shelf unit that was modified for remote operation and placed in a specially fabricated waterproof housing. The waterproof housing was fabricated from PVC. It is designed to be rugged, lightweight, and able to float without the use of any additional flotation material. It houses the camera, pan and tilt unit, and an electrical connector for the camera control cable. The face of the

housing is a clear plastic dome that allows the camera to operate through its full pan-and-tilt range. The camera system can be removed for maintenance or adjustments by sliding the top cover out of the housing, The housing system also includes a pneumatic actuator that tilts the whole housing up to 90 degrees so that the camera's 180 degree tilt can cover floor-to-ceiling or front-to-back underwater.

The basic lighting system consists of a 50 watt quartz halogen light lamp. The bulb was placed inside the metal housing, which serves as a heat dispenser and allows it to operate both in and out of water. The unit is then placed in a waterproof PVC housing. The light is also designed to be self-floating. An electronic transformer in the control box is used to supply 12 volt AC to a dimmer that controls the brightness. Additional lights can be mounted on the floating boom or camera housing, and drop lights suspended from the boom to just a few inches above the sediments were also successfully tested.

Fig. 2

REMOTELY ACTIVATED SAMPLERS

The ORNL TCS includes two remotely activated samplers, a clamshell grappler for taking samples of bottom sediments and debris, and a concrete wall chip sampler. The clamshell grappler system is designed to retrieve sludge and debris samples and small objects from tanks with limited access. The grappler system consists of two sub-systems: grappler and winch.

The grappler is a double-jaw sampler operated pneumatically from a remote location (control box). It can be open or closed as often as required without the need for manual reset. Force at the jaws can be adjusted anywhere from 0-60 pounds on the current design, and more with minor modifications (Fig. 3). All parts are waterproof and fully submergible. The body is fabricated from PVC, which was selected for its light weight, low cost, and ease of cutting and machining.

The winch system is an electric driven winch floating on the water directly above the grappler. It will lower and raise the grappler to and from the bottom of the tank remotely. Deployment depth is determined by monitoring lowering and raising times, by an electronic pulse read at the control box, and/or by direct observation with a video camera. A worm drive gear reducer can stop and hold the grappler at any depth and at any distance from the winch system to facilitate the entry and exit from the tank manhole.

Fig. 3

The concrete wall chip sampler consists of a small pneumatic die grinder that is mounted on a float at the end of the floating boom. The grinder is maneuvered to the desired sample location and the bit is placed in contact with the wall. A small venturi is used to create suction and collect the grindings in a sample bottle. By grinding the wall, small samples of concrete are collected for radiological analysis. These data will be used to model dose rates and residual contamination levels.

SONAR SLUDGE MAPPER

An off-the-shelf sonar depth finder was mounted on the TCS floating boom to map the depth of sludge in the tanks. By varying the sensitivity of the sonar, the operator can distinguish the sludge surface (high sensitivity) and the concrete floor of the tank (low sensitivity). This technique allowed us to refine sludge volume estimates in any tank with at least 2 feet of water. Figure 4 shows examples of the sludge maps generated from TCS data.

Fig. 4

CONCLUSION

The ORNL TCS demonstrates that innovative use of off-the-shelf technology, combined with remote operations know-how, can solve difficult problems at low cost. This system was used to characterize seven highly radioactive underground storage tanks at ORNL. Data collected included sonar depth information, sediment samples, debris samples, concrete samples, and video observations. The TCS concept can be easily modified to accept a variety of samplers and sensors. For example, addition of radiation detectors would be a simple matter as would collection of liquid samples, if these data were required by a project. Data from the characterization campaign in now being used in planning and executing the treatability study that will lead to final remediation of these tanks. The floating boom concept is ideal for characterization of large underground tanks, aboveground tanks, waste basins, and ponds.

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MEETING THE REGULATORY CHALLENGE WITH COST-EFFICIENT TRAINING

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ABSTRACT

Navigating the regulatory maze to identify and interpret mandatory training requirements is often a perplexing, frustrating experience. By necessity, training regulations indicate only general categories of information that should be covered in a training program, not how the information relates to a particular site or how it should be presented to employees. Training regulations are, by definition, performance-oriented because they are designed as guidelines for meeting specific safety, health, and environmental management needs. The regulations leave it to employers to figure out how to implement a program that will ensure that workers can perform their jobs in a manner that meets the statutory requirements.

Cost-efficient training is not simply training that is cheap to design and deliver. Cheap training that is ineffective will be costly over the long term. Cost-efficient training is a function of effective, valid training outcomes. Because the training regulations are performance-oriented, the training program must be performance-based to ensure that critical job-related knowledge and skills are acquired by the trainee. The key to the challenge of creating cost-efficient training that meets regulatory requirements is to wisely invest time at the front end of the process, i.e., thoroughly analyze job functions and assess the tasks that are performed. When trainers clearly understand the individual components of a job, they can more effectively design training to address critical job-related knowledge and skills and integrate meaningful information that meets regulatory mandates. Cost-efficient training accurately targets the employee population requiring each specific training course thereby optimizing training resources and minimizing employee down time. This paper focuses on the job analysis process that must take place before designing a training program. Job analysis is a crucial, and often underutilized, step in a systematic approach to training. Three methods for acquiring job analysis data are presented, and the advantages and disadvantages of each method are discussed. This paper also describes the

process for assessing the analytical data and emphasizes the importance of using automated data processing tools to optimize the analysis process by increasing data management efficiency.

PROBLEM

A number of unsatisfactory outcomes, ranging from mildly irritating to potentially disastrous, may result from training programs that are designed without the benefit of a thorough job analysis. Some of the most common outcomes resulting from inadequate or absent job analysis information include the following:

Training content does not reflect relevant job-related knowledge and skills; therefore, it is of limited effectiveness.

Trainee time is wasted or is minimally productive.

Training is not useful because it deals with knowledge or skills that the trainee does not need or has already acquired.

Training does not ensure competent job performance.

Inadequate training programs compromise worker health and safety or may result in adverse environmental impacts. Training staff and managers may be held liable for these consequences.

A training program that is designed without the benefit of a thorough job analysis is like a sailboat without a rudder. The rudderless boat may look majestic and proud when its sails are unfurled, but it will have no directional control. It may by sheer chance reach its intended destination, but the route is likely to be costly in terms of time and energy. More likely, it will simply drift and flounder at the whim of the winds and currents, never quite reaching its goal. Data from job analyses give a training program directional control and clarify the destination. After that, it is up to the training design and implementation to provide the most direct route to the destination, but when the way is in sight, the probability of success greatly increases.

SOLUTION

Using a systematic approach to training, managers can create a sound, defensible training program by documenting the links between job functions, training requirements, and available training. The process will also identify gaps in the training program and serve as a navigational aid to future program development.

Job analysis forms the foundation on which the remainder of the training program is built. Assessing the job analysis data establishes the boundaries for the design and development steps that follow, ensuring that training resources are targeted appropriately. Job analysis data helps managers prioritize their training decisions based on sound rationale and a hierarchy of needs, especially important when training resources are limited.

Fortunately, job analysts have a choice of strategies available to derive the necessary data. The method that is ultimately chosen is based on several considerations as described in the examples that follow.

Method 1Observation/Interview

This job analysis strategy, which is the most comprehensive of the three methods described in this paper, involves observations of workers performing assigned job duties and in-depth interviews with key personnel. This method is applicable when work processes, procedures, and workers' job functions are not adequately documented and validated.

Under ideal circumstances, the job analyst observes workers perform the full suite of functions within a given duty area. A duty area is a set of similar job functions. For example, a worker who has a duty area

assignment of "waste handling" may perform several related waste-handling functions, such as waste acceptance, transport, storage, treatment, and disposal. A set of discrete "tasks" describes subactivities within a given function. For example, the waste storage function may include tasks involving overpacking, bar coding, palletizing, and stacking drums.
Typically, a given duty area comprises job functions that are performed at variable intervals, i.e., some are performed frequently (on a daily basis), while others may be performed only as a result of an unusual and infrequent occurrence (e.g., a spill or other emergency). For this reason, it is not always feasible or practical for a job analyst to be able to observe the performance of every job function. In such instances, interviews with one or more persons knowledgeable about the task is necessary.
Whether the information is derived from personal observation or interview, the job analyst must be an astute observer, a probing questioner, attentive to detail, and a meticulous note taker. Video cameras and audio tape recorders are useful tools for documenting observations and interviews. Videotape of the function can be used later when developing training or writing procedures. Video can be especially useful in this era of computer-based, multimedia training delivery.
The job analyst uses the observation and interview material to draft detailed job function descriptions. The draft descriptions are reviewed for accuracy and completeness by the workers who were observed and other subject matter experts (SMEs). Final descriptions provide the basis for identifying task-specific individual qualification requirements and training plans.
Method 2Table Top Job Analysis
An innovative method called Table Top Job Analysis (TTJA) optimizes the use of SMEs to develop comprehensive task lists through a consensus decision process. In contrast to the method described above, which can take weeks to complete, a TTJA can elicit detailed job information in a matter of two to three days, even for complex jobs.
TTJA can be used in place of the observation/interview method. It is especially useful when a new or significantly modified operation is being initiated and when work processes are not adequately described in written procedures. The TTJA process pools the collective knowledge of several qualified SMEs. (Three to five SMEs is ideal; less than three reduces the synergy effect and more than five becomes unwieldy. An odd number is best to avoid tie votes.) A skilled facilitator is an essential ingredient in the success of the TTJA method.
In the TTJA method, a facilitator first guides the SMEs through a brainstorming session that results in identifying major functions that comprise a given duty area. The facilitator must be skilled at keeping the discussions focused at the appropriate level during this stage and at establishing positive group dynamics for the difficult decisions to come later. When the group agrees that all major duty areas and functions have been identified, the discussions shift to identifying and sequencing all tasks associated with performing each function. Depending on the complexity of the job, this can be an interesting process, as the SMEs must reach consensus on each item.
An assistant helps the facilitator document the decisions as they are made by entering the functions and task statements into a database. The lists can become the basis for writing procedures and developing training for each job function.

Method 3Documentation Review

This final job analysis method is the simplest and fastest of the three methods; however, it is only applicable when job descriptions, procedures, and work processes are adequately documented. Taking advantage of a variety of written documents, such as standard operating procedures, detailed operating procedures, special work permits, management plans, emergency plans, health and safety plans, and job descriptions, the job analyst examines the documentation, identifies the major job functions, and extracts task statements that represent each function. Data gathering can be enhanced through personal interviews and/or the use of questionnaires. The task statements are validated by job incumbents, supervisors, and other SMEs and become the basis for training decisions.

PROS AND CONS

There is no one perfect job analysis method. Given careful consideration of each situation, the job analyst can optimize the advantages by using the most appropriate method. Disadvantages can be minimized by acknowledging the limitations imposed by the method. It is important to use a sound, systematic approach, and each of the methods described yields information that meets this requirement. A table comparing the advantages and disadvantages of each method is displayed in Table I.

Table I

TASK RATING CRITERIA

As previously mentioned, effective job analysis uses a systematic approach. As we have seen, the first step in the process is to identify duty areas, functions, and related tasks for each function using one or more of the methods described above. The next step involves assessing each task against task criteria to answer the following questions:

- How difficult is the task?
- How important is the task?
- How frequently is the task performed?

Applying an objective approach to answer these questions results in train, no train, or overtrain decisions.

To optimize the rating process, each task must be described in the form of a simple task statement, e.g.,inspect waste drums, use forklift to move drums to staging area, overpack damaged drums, contain spills, record data on inspection checklist, etc. SMEs rate each task against specified task rating criteria. Figure 1 provides an example of typical rating criteria. It is necessary to ensure that each rater assesses each task against clearly identified criteria. When the rating process is completed, the analyst compiles the numerical averages of the responses and uses a decision matrix to make initial determinations of tasks selected for training. Figure 2 is an example of a decision matrix that is useful to aid in the rating process. Decision matrices are available in job analysis textbooks or other publications, or they can be customized to fit the specific facility and operations environment. The matrix depicted in Fig. 2 was adapted from the DOE Training Accreditation Program Performance-Based Training Manual (1). The task rating process results in recommendations to train, not to train, or overtrain as follows:

- TRAINProvide initial training (either classroom, self-paced, on-the-job, drills, simulations, job performance aids, or a combination).
- NO TRAINNo formal training is necessary; the task can be learned on the job or through required reading.

OVERTRAINProvide a combination of formal training plus periodic retraining at specified intervals.The initial training decisions are not unyielding. They are intended to be guides to making reasonable training decisions using an objective approach. Training decisions can, and should, be modified using the analyst's knowledge of the task or by applying regulatory requirements as intended and neededselectively, as appropriate for the facility and job functionusing a graded approach. Modifications to initial training decisions should be based on sound, logical rationale and the reasons should be documented and consistently applied to other similar situations.

Fig. 1

Fig. 2

DATA MANAGEMENT

Because the job analysis process typically generates a significant volume of information, it is advantageous to use automated data processing software to manage the data. Benchmark developed a customized relational database for this purpose. The Total Training Tool (T3) can be used to record duty area and task statements. T3 simplifies the task rating process by automating the calculations and the decision process. Training specialists identify mandated training requirements, assess existing training, and create the linkage to functions and tasks. Using T3, individual training plans and other customized reports can be quickly generated for facility managers, line supervisors, auditors, and inspectors. T3 is useful for generating reports sorted in a variety of ways, e.g., by worker, regulatory requirement, job function, task, or training course.

CONCLUSION

Training should not be conducted simply as a knee-jerk reaction of blind adherence to a regulatory requirement. Training requirements do not exist solely for the purpose of being complied with. Compliance should serve a purpose, i.e., it should address behaviors and attitudes and provide workers with the necessary skills and knowledge to perform their jobs safely, effectively, and in a manner complying with applicable regulations. Training should support this purpose by being appropriate for the workers, applicable to the operations, and instrumental in fostering improvements in the workplace.

Facilities that use the "blanket" approach to training may have a compliant program, i.e., everyone is covered, thereby ensuring compliance, but the program is unlikely to be cost-efficient. Cost-efficient training goes beyond compliance by integrating mandatory training requirements with performance-based job-related knowledge and skills and selective application of training to the appropriate audience. The key to successful, cost-efficient training is understanding the finer points of the work processes and worker behaviors. Gaining this understanding requires putting in the effort at the beginning to save time and energy along the way. Job analysis information provides the navigational map and clarifies the goal. Using an appropriate job analysis method, assessing the job analysis data, and following through with an approach that reduces the subjectivity in the training decision-making process will result in a defensible training program that will withstand the regulatory challenge and meet the needs of workers.

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UNIVERSITY FACULTY AS A RESOURCE FOR STATE LEGISLATURES CONSIDERING NUCLEAR WASTE ISSUES

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ABSTRACT

In 1995 the Ohio General Assembly considered legislation that would enable Ohio to begin work on the siting, licensing, construction, and operation of a low-level radioactive waste disposal facility for the Midwest Compact. The legislation was signed into law on June 9. Prior to their vote on the enabling legislation, members of the General Assembly and their aides gathered information related to low-level waste. They frequently turned to a team at The Ohio State University that had produced educational materials designed to provide the citizens of Ohio and their elected officials with accurate, research-based, unbiased information on radiation and low-level waste. This paper outlines the types of educational materials developed and ways in which a State University was able to serve as a resource to the State Legislature as it prepared to deal with a technical, and somewhat unfamiliar, subject.

BACKGROUND

When Michigan's membership in the Midwest Interstate Low-Level Radioactive Waste Compact was revoked in the summer of 1991, and Ohio became the Compact's first host state, a team of faculty members at The Ohio State University (OSU) recognized that Ohio's citizens and their elected officials would eventually be discussing low-level radioactive waste. The team from the OSU Nuclear Engineering Program and OSU Extensiona anticipated the need for accurate, research-based, easy-to-understand materials on low-level waste. A proposal to prepare such materials was written, and funding was sought. In the spring of 1992, the Midwest Compact Commission agreed to fund the educational program.

MATERIALS DEVELOPMENT

A great deal of information on low-level radioactive waste has been prepared. Some of it has been written by groups with a specific point of view to express. Some is in technical sources such as textbooks, formal reports, and research journals. Materials written in an effort to persuade the reader to support a particular point of view can be inaccurate or incomplete. On the other hand, many of the technical books and journals, while accurate, are written for a small, technically-trained audience and are not widely distributed. These materials often fail to provide the information people want.

When citizens and their elected officials are discussing and making decisions on technical issues, they would like to have accurate, unbiased information that is easy to understand. A university is a logical source of such information. Faculty have access to the latest texts and research journals and the expertise to interpret them. In addition, faculty working in technical fields can team with professors having expertise in development of educational materials for the general public to produce easy-to-understand documents on highly technical topics.

Educational materials developed for the public on a technical topic should be made available in several formats. Each individual has a preferred method for receiving information. For example, some like to watch a video, while others prefer to listen to a presentation and ask questions, and still others want something they can read and study at their leisure. If the technical information is to be truly accessible to the non-technical audience, it must be presented in multiple formats that meet the audience's needs.

Under the Statewide Low-Level Radioactive Waste Education Program at Ohio State, materials development began with the writing of 27 fact sheets. Each fact sheet was only one or two pages long and addressed a single issue. The fact sheets fell into four general categories: 1) radiation science, 2) characteristics of low-level waste, 3) technology related to low-level waste, and 4) historical and legal topics. Information in these fact sheets was to become the basis for the development of educational materials in several other formats.

First, however, to ensure that each fact sheet was accurate, unbiased, as complete as possible, and easy to understand, an extensive review was conducted. A fact sheet written by a team of nuclear engineers and faculty with expertise in presenting technical information to the public was first reviewed by other people with technical expertise on the topic being addressed. After revisions were made based on comments by the technical experts, the fact sheet was sent to a 5-person review panel. This volunteer panel consisted of a physician working in nuclear medicine, a nuclear engineer with expertise in nuclear safety, an expert in science education, a representative of a national environmental group who specialized in nuclear energy, and a retired judge. The panel reviewed all fact sheets in an effort to ensure that they were technically accurate, easy to understand, and unbiased. Finally each fact sheet was sent to all "interested parties" for their comments. The "interested parties" included environmental groups in Ohio, generators of low-level waste, some legislators, journalists, educators, and anyone else who indicated that he or she would like to review the fact sheets.

After the fact sheets were completed, information in them was used to prepare several other types of educational materials. A set of four table-top exhibits was built for use in places such as mall shows, county fairs, and meetings of civic groups. Overhead transparencies were produced to be used at live presentations by team members, and a slide-tape program and a video were made using some of the same visual images. In addition, a large exhibit was built for use at the Ohio State Fair.

The educational materials were distributed both by the project team in Columbus and through the OSU Extension network which has an office in each county. Sets of fact sheets were delivered to County Commissioners, local health departments, hospitals, newspapers, libraries, high school teachers, college faculty members, civic organizations, and interested citizens. Approximately 13,500 sets of fact sheets were distributed. Exhibits were displayed at many meetings and community events. Dozens of presentations were made to local officials and interested groups. In addition, time was spent serving as a resource for the State Legislature. The remainder of this paper focuses on those activities.

RESOURCES FOR THE STATE LEGISLATURE

When the fact sheets were completed, about a year and a half before enabling legislation related to low-level waste was considered in the Ohio General Assembly, the Low-Level Radioactive Waste Education Project

Leader briefed members of the Ohio House and Senate and their aides on the fact sheets. A set of fact sheets was delivered to each legislator's office. They were intended not only for use by the legislators and their aides but also as reference material that legislators might copy and send to their constituents who were interested in the topic. Following elections, sets of fact sheets were sent to new legislators. Project Team members answered questions from legislators or their aides related to the fact sheets and provided additional sets to constituents when asked.

Approximately four months before the enabling legislation was introduced, the Project Leader presented a half-day seminar on fundamental concepts related to low-level waste to state agency personnel. A group of state agencies, including the Environmental Protection Agency, Department of Health, and Department of Natural Resources, asked that the seminar be held for people in their agencies who might be asked to field questions on low-level waste from state legislators. The seminar included information on radiation science, characteristics of low-level waste, technologies related to low-level waste (such as treatment, minimization, transportation, and disposal), and the history of low-level waste management in the United States. This seminar was designed to give agency personnel background information that could help them provide accurate and timely information to legislators.

Shortly after enabling legislation was introduced into the Ohio Senate, the Education Project Leader was asked to give a seminar for aides to the State Senators. The goal of the seminar was to provide the aides with basic information on radiation and low-level waste. The presentation focused on fundamental scientific concepts, sources, volumes, and characteristics of low-level waste, alternative technologies available for managing low-level waste, and additional sources of information that the aides might consult. The four table-top exhibits were set up at the seminar.

The Project Leader was also asked to testify at hearings on the low-level radioactive waste legislation held by the Senate Energy, Natural Resources and Environment Committee. Testimony at these hearings focused on the goals and organization of the Education Project. It seemed necessary to establish that a major effort had been made to have the educational materials reviewed by people with a wide range of views on low-level waste and to produce accurate, unbiased documents.

When the enabling legislation moved to the Ohio House, the Chairman of the House Committee on Energy and the Environment invited the Project Leader to be the first witness at the hearings on the bill. The Chairman specifically asked that the Project Leader not bring and read the usual written testimony but rather that she bring overhead transparencies and present a seminar. The goal of the presentation was to provide Committee members with an understanding of some of the basic concepts in radiation science, the nature and sources of low-level waste, and the technologies used to treat and dispose of that waste. Committee members were encouraged to ask questions, and several did. Copies of the overhead transparencies were provided to the Committee.

Two months later, the House Committee Chairman asked the Project Leader to return for another "seminar". Committee members had heard a great deal of testimony, some of which had prompted more questions about radiation science. In addition, Committee members had questions about how radioactive material moves through the environment and what factors should be considered when determining how hazardous a particular

radioactive isotope might be. The first part of testimony in the "seminar" format was a review of the material presented earlier. The second half focused on how radioactive material might escape from a disposal facility and how factors such as particle size, chemical form, solubility in water, and affinity for soil might affect how the material, once released, moves through the environment. Some Committee members also wanted clarification of the definitions of low-level and high-level radioactive waste.

As a vote on the enabling legislation neared, one Ohio Representative asked the Education Project Leader to present some fundamental concepts related to low-level waste at a town meeting in his district. Both proponents and opponents of the legislation spoke at the town meeting. The Education Project Leader's role was to speak first, providing background information and defining some of the terms that would be used in the ensuing discussion. She did not take part in the discussion other than to clarify definitions or to provide technical descriptions.

DISCUSSION AND CONCLUSIONS

From its inception, Ohio State University's Low-Level Waste Education Project was designed to provide accurate, research-based, unbiased information. It established that position with a project team made up of faculty with diverse backgrounds and by having all of its materials reviewed by dozens of people with widely varying views on low-level waste. The Project took no position on the low-level waste legislation. The Project Leader was, therefore, able to serve as a credible witness, one to whom legislators could turn for sound, fundamental technical information they required to make a decision.

States throughout the nation are struggling with the problems related to storage and disposal of radioactive waste, decontamination and decommissioning of nuclear facilities, and cleanup of contaminated sites. Legislators will have to make decisions on some of these matters, but nuclear science is not a topic familiar to most legislators. By anticipating the information needs of the legislators, preparing accurate, unbiased, easy-to-understand educational materials on those topics, and making themselves available to talk with legislators, their staffs, and their constituents, university faculty members with nuclear expertise can provide a valuable service to their states.

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USING THE WORLD WIDE WEB TO IMPROVE K-12 SCIENCE TEACHING

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ABSTRACT

The use of the World Wide Web (WWW) to improve science teaching and to provide more equitable access to science resources, including those that apply to nuclear energy and radioactive waste management, are addressed in this paper. For the past five years, the Yucca Mountain Office of the DOE has been supporting science education in Nevada through greater use of educational technology, most recently by supporting rural school connectivity to the WWW. This paper gives the details of how six rural Nevada schools are being connected to the Web. It further addresses how

they will take part in an international effort of the International Alliance in Education for Radioactive Waste Management to a) promote nuclear literacy and b) educate the public to be more receptive to the need for safe repositories to store radioactive nuclear waste.

INTRODUCTION

The improvement of science, mathematics and technology instruction in pre university level education continues to be an educational priority of government and business in the United States and other industrial countries in the world (1). In the belief that science and math education are tied to our economic well being and standard of living, the U. S. federal government has made it an educational priority and set goals for improving it. One educational goal of the Bush administration was to establish a coordinated government effort to improve science and mathematics education (2). The new national standards in mathematics and science and the proposed standards in technology are all aimed at improving education in what the federal government considers to be the essential areas needed to protect our technological advantage in the world economy (3).

Federal money continues to support two key areas in education. First, pre-service and in-service training for teachers of science and mathematics. Second, the development of better curriculum and instruction in science and mathematics. To support this second key area, the federal government has made networking and infrastructure development, particularly support of the Information Superhighway (the Internet) a priority. In 1991, the federal report America 2000, directed the secretary of education, in consultation with the president's science advisor and the Director of the National Science Foundation (NSF), to determine how electronic networks might provide American schools with ready access to the best information, research, instructional material and educational expertise (4).

By the spring of 1995 35% of American schools had access to the Internet but only three percent of instructional rooms (classrooms, labs, and media centers) were connected to the Internet (5). No data is currently available on what per cent of schools using the Internet have a graphical user interface (GUI) such as Mosaic or Netscape to give graphical retrieval ability from the World Wide Web (WWW). But, according to information compiled by mkgray@netgen.comnet.Genesis Corp. of Cambridge, Massachusetts, there are nearly 16,000 Web hosts in the world ranging from Armenia to Venezuela. [The World Wide Web is a special segment of the Internet that supports the transfer of documents containing text, graphics, sound and full motion video (6). The Web is further defined as a graphical representation of Internet resources based on the "hypertext" concept. In hypertext, the reader is not constrained to follow text in a linear fashion, but, can view randomly through resources on the WWW that include graphics, sound and video.]

This paper will address how the U. S. Department of Energy (DOE) is supporting the improvement of science instruction, through the use of educational technology. Specifically, it will discuss how WWW access is being developed in rural Nevada schools, and the potential that access to Web sites offers for improving science education in all schools.

RATIONALE

In line with two of the national goals of science educationa, the DOE, through the Yucca Mountain Project Office, already has a history of supporting the use of educational technology to improve science education

in Nevada, particularly in rural schools. Previous DOE help has included the donation of many personal computers, the purchase of modems, support for dedicated phone lines, and underwriting the training for teachers on how to use the Internet. One major justification of the DOE for this support has come from the need to try to equalize educational opportunities throughout the state. Research indicates that rural areas do not have the learning opportunities both in and outside the schools that are present in the large urban areas, e.g. Las Vegas and the smaller urban area of Reno (7). For many years, Canada and the U. S. have had telecommunications that carry educational programming to rural schools and other sites (8). The equipment required is generally quite expensive, especially when two way video is used at each site. The authors of this paper have suggested that the WWW might be a much less expensive way to provide instruction as well as make educational opportunities in science more equitable in rural schools.

From a science pedagogy point of view, further justification for the WWW is the support it offers to the Science Technology Society (STS) teaching method and the Constructivist Learning Model (CLM). The STS teaching method and the CLM are being promoted to improve science teaching in both Benchmarks of AAAS and the National Science Education Standards of the National Research Council. Currently the STS science teaching method and the CLM are thought to be the best ways of promoting science literacy in all students. Furthermore they can make science more relevant and interesting by helping students see the connections between science concepts and technology applications in society. In this paper, the STS teaching method is defined as the science teaching method and philosophy that believes science should be taught in the context of human experience and that science is a necessary part of the education of every citizen (9). From the perspective of the CLM, knowledge is a mental representation of the natural world and learning is a social process during which students use what is already known to make sense of new experiences (10).

Studies have shown that student performance in science and mathematics is enhanced by access to and experience with computers and various science equipment (11). The WWW offers the prospect for helping students experience a connection between their lives and the science that occurs outside the classroom. As science teachers try to make students see the connections of science to other disciplines, e. g., social studies, mathematics and foreign languages, the WWW can provide the information source. With the WWW, students have the possibility of interacting with a variety of resources inside and outside the U.S. that would not usually be available inside their classrooms. Moreover, the real life experience of collecting and sharing a variety of information on the WWW supports the STS teaching method and the CLM. Many science teachers and students already use networks such as Newton (where questions can be posed to working scientists in the DOE), NASA, the Weather Underground, Ask Eric (an education resource for teachers) and some of the environmental science networks, e. g. EcoNet, that address a number of common problems such as global warming and air pollution.

Unfortunately, only a small number of rural schools have the capability of accessing and downloading the color graphics found on the WWW. The possibilities for improving science instruction with the WWW are truly exciting. As a small example, think of a class studying and comparing various energy sources such as solar, nuclear, wind power and fossil

fuels. If that class had access to the WWW it could in fact access a computer in Sweden (http://www.ida.liu.se/~her/) and view a diagram of the Chernobyl, Soviet designed nuclear reactor. Think what an exciting learning experience this could be for a classroom in anisolated rural school and how much more relevant and interactive science would be to that class.

A MODEL FOR ESTABLISHING RURAL SCHOOL WWW ACCESS

Background

As mentioned earlier in the paper, the DOE has been active in helping Nevada schools gain more educational technology. DOE help for Nevada schools started four years ago with the donation of surplus computers and support for teacher training on how to use the Internet. The support for teacher training on the Internet has continued for the past four years. More recently funds were provided, in collaboration with the University of Nevada at Reno, for the purchase of a file server for the Nevada Schools Network (NSN), an Internet server which is now used by over 2,000 students and teachers throughout the state.

In the current project, the DOE Yucca Mountain Office has provided the money for some equipment, software and training to establish WWW capability in six rural Nevada schools. The number of sites is limited because of funds, hardware and connectivity pre requisites. Some requirements for rural school access to the Web include a computer with preferably at least eight megs of RAM, a Macintosh or a computer with Windows, a high speed modem and a local Internet access provider which in this case is the Great Basin Internet Service.

The establishment of WWW capability is part of a larger DOE grant that supports the International Alliance in Education in Radioactive Waste Management (IAERWM). The International Alliance has representatives from 12 of the OECD/NEA countries. All of these 12 OECD/NEAb countries have the need for an organization with links to public education and dedicated to: a) Finding ways of raising the level of nuclear literacy in the general public; and b) Educating the public to be more receptive to the need to site and build safe repositories to store radioactive waste. Acknowledgement that common educational strategies to address a) and b) above was first agreed upon in an educational workshop on radioactive waste management (RWM) held in Switzerland (12) and attended by educators from 12 of the OECD/NEA countries in 1991. The International Alliance of the 12 countries was formalized in the fall of 1991.

Since that time Alliance Conferences have been hosted by the UK, The Netherlands and Canada. Educational materials on RWM from participating countries were shared at each of the three conferences. One goal of the 1993 meeting in The Netherlands was to explore the use of the Internet for disseminating educational information on RWM. No Alliance meeting was held in 1994 but the 1995 meeting in Canada continued to explore the use of the Internet as a way of disseminating RWM information to schools. A four hour workshop dealt with the possibility of using the Internet and some of it's tools, e. g. gopher and the WWW, to establish nuclear education links between the Alliance countries. All participants of the conference agreed that having Web sites in their information agencies would be an ideal way to provide access to educational materials on nuclear energy and radioactive waste management. To carry this out it was suggested that each Alliance country; a) Establish a homepage and support access to it by students and teachers from a few schools; and b) List on

their homepage their educational resources on nuclear energy and RWM for teacher and student access and retrieval.

Nevada Web Sites

During the Fall of 1995 and continuing into the Spring of 1996 at least six sites with WWW graphical access will be established in rural Nevada with the monetary support of the DOE and technical support from the College of Education, University of Nevada in Reno.

The development of the school sites with web capability will be a cooperative effort which will involve each school's science teacher in an effort to establish specific education in science including nuclear energy and RWM. Additional curriculum materials will be available on the DOE Yucca Mountain Office web site (http://www.ymp.gov/). One author of this paper is responsible for part of the technical support, including teacher training and any help needed to establish a good platform for accessing the Web.

The Web development at the initial six rural school sites involved the consideration of several components; including specialized telecommunications lines, computer hardware, special 28.8 modems, and HTML editing software. The first phase of the development also involved discussion with the local telephone company for determining the best telecommunication line setup so that schools will not be hindered when graphically accessing the WWW. Initially the five of the six schools will have a regular dial-up remote connection at the 28.8 baud rate which is provided with a charge by Great Basin Internet Service. The goal is to move up to the faster speeds that are provided by ISDN lines. At this point only one of the six schools is scheduled for an ISDN line.

The second phase of the development is focusing on science education content and the process of developing the content into home pages. This phase will also include teacher training at each site so that each school can develop its unique educational interest in science education for the WWW. This second phase will also directly involve development of nuclear energy educational curricula by DOE at Yucca Mountain. At the end of the second phase, each school should have graphical access to the WWW with the knowledge of how to setup and manage a homepage. The six rural Nevada schools (Virginia City, Lovelock, Hawthorne, Battle Mountain, Silver Springs and McDermit) will also be helped to establish school partners in other OECD/NEA countries initially through AECL (http://www.aecl.ca) in Pinawa, Canada and OECD (http://www.nea.fr/) in Paris, France. Another site recently opened through British Nuclear Fuel in the UK and certainly other sites will continue to be developed.

As other nuclear information office homepages come on line with additional nuclear energy curriculum materials, Nevada students will make use of the information. At this point, the web server homepage (http:://nspweb.ed.unr.edu:443/) for the six schools is on line and located in the Department of Curriculum and Instruction of the University of Nevada, Reno. Many science links are already available on the homepage and more information is being added each week. Three rural schools are now using the WWW and the other three will soon be connected. Soon, training will be provided at each site to show the teachers how to use the Web for direct instruction and how to set up their own web page.

Evaluation

An attitude scale (see appendix) was designed to be taken by students prior to the establishment of WWW capability in their schools. So far, only three six in the three on-line schools have taken it. The other

three schools will take it as soon as they are connected. The attitude scale will not be repeated until after the Waste Management Symposium and that data will not be available until at least Spring after the students have had at least two months to use the Web. Control group classrooms, not on the Web, are also being used for comparison to classrooms with web capability.

The 15 item attitude scale is designed to measure the attitudes of the students toward science in general with a few specific questions on nuclear energy. It is broken into three, five question parts; feelings toward science, knowledge of science, and science behaviors. The independent variable is the new instructional strategy, use of the WWW. It is hypothesized that when students use the Web to access, retrieve and share information on science topics, e. g. nuclear energy, student attitudes toward the study of science in general and nuclear energy topics in particular will improve. The learning process provided by the Web as well as the additional science information available to students are both expected to be a positive influence on attitudes toward science learning, including nuclear energy. This information is important in Nevada, a state where there is a lot of negative feeling toward the possibility of Yucca Mountain becoming the U. S. nuclear waste repository.

CONCLUSION

Complete conclusions regarding the survey and any anecdotal information from teachers, students and principals can not be made until all six schools are on-line and have had a few months to use their WWW capability.

Some comments are offered concerning what changes in attitudes are expected as a result of the WWW use by students and teachers in the identified rural schools. It is generally expected that student attitudes toward science, including nuclear energy, will improve for several reasons; some of which are based on variables that cannot be realistically controlled in this kind of research. First, the teachers and schools were selected based on past interaction with them that indicated a greater probability for setting up successful WWW capability. The identified teachers already use computers and other contemporary technology in science instruction and they wanted to take part in a project that would give them WWW capability. They are teachers that have been identified as science teachers who are known to have characteristics that are attributed to exemplary science teachers (13). Such characteristics include making science interesting, using lots of materials and resources, showing a high interest in science, asking lots of questions, and letting students pursue ideas in which they have an interest. Furthermore, the principals of the schools generally have a record of supporting technology and helping their teachers gain access to better instructional tools.

It is hoped that the data from the six sites will help a) further the research on the use of educational technology to improve science teaching, b) improve student attitudes toward science and c) improve the knowledge base of students in nuclear energy and RWM. Unquestionably, the possibility for helping equalize education opportunities in rural schools by providing WWW capability is a worthwhile educational goal in its own right and should be pursued.

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APPENDIX

Survey of attitudes toward science

More and more information is in the daily news to explain the importance of studying science in school. It is said that students need a better understanding of science in modern society because science impacts on so many everyday life issues that good citizens in a democratic society are expected to have a understanding of even if they are not scientists or engineers. It is also said that students who have taken more science courses are easier to train for jobs that require some understanding in science. For whatever the reason, science is becoming a basic requirement in the education of American students as well as students in other industrialized societies such as Germany and Japan.

In the following 15 questions, please circle your response using the scale of 1-5. The key for the numbers is a follows:

1. Strongly Agree
2. Agree
3. No Opinion
4. Disagree
5. Strongly Disagree

1. I like science because it is more interesting than other subjects.

1 2 3 4 5

2. I like science because it is useful in my everyday life.
1 2 3 4 5

3. I like science because it affects me personally.
1 2 3 4 5

4. I like science because I know it is important for my future.
1 2 3 4 5

5. I like science because it makes me curious about the things around me.
1 2 3 4 5

6. I like science because it helps me analyze whether people know what they are talking about when they state an opinion.
1 2 3 4 5

7. I like science because it helps explain things in life that I have questions about, e. g., photosynthesis, spontaneous combustion and radioactivity.
1 2 3 4 5

8. I like science because it helps me solve problems outside of school, e. g. how to predict changes in the weather.
1 2 3 4 5

9. I like science because it helps me identify and try to solve problems that affect the community; e. g.; water resources, air pollution or hazardous waste disposal.
1 2 3 4 5

10. I like science because it gives me the knowledge to make decisions about important science based issues, e. g. the nuclear waste repository in Nevada.
1 2 3 4 5

11. I like science because it teaches me how to design an experiment to test a hypothesis, e. g. to test my ideas.
1 2 3 4 5

12. I like science because it teaches me how to use the right resource to find information when I need it.
1 2 3 4 5

13. I like science classes because I don't just study ideas but I get to do things with interesting materials and resources.
1 2 3 4 5

14. I like science because I get to discuss and compare my ideas with other students.
1 2 3 4 5

15. I like science because I get to identify problems and try to solve them.
1 2 3 4 5

23-37

TURNING INSPECTION REGULATIONS INTO TRAINING TOOLS

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ABSTRACT

In response to suggestions from internal and State of California auditors, the Hazardous Waste Management Division (HWM) at Lawrence Livermore National Laboratory prepared an Inspection Schedule and

Guidance Document that summarizes the Laboratory's inspection schedule and procedures for waste treatment, storage, and disposal facilities (TSDFs). Because it explains and comments in detail on the inspection schedule, forms, and procedures, this document is a centralized reference for HWM managers and personnel performing TSDF inspections at the Laboratory. It is also a training tool for experienced and new inspectors, standardizing the inspections of personnel with experience and explaining to novices what to look for and why. This poster presentation traces the team effort that created this document and provides specific examples of how the document was developed and how it is used.

THE REGULATORY FRAMEWORK

The environmental regulations governing the hazardous waste treatment, storage, and disposal facilities (TSDF) at Lawrence Livermore National Laboratory (LLNL) are basically the same as those governing any TSDF anywhere in the United States. Although our regulator and permitter is the California Department of Toxic Substances Control (DTSC), the regulations we must abide by are those of the U.S. Environmental Protection Agency (EPA) and the Resource Conservation and Recovery Act (RCRA). The EPA has given CDTSC authority over waste management operations at LLNL and throughout California because CDTSCs regulations are as stringent as or more stringent than EPA's and RCRA's.

So, what is true of waste management operations as a whole is therefore true of the inspections of the TSDFs at the Laboratory that keep them safe. At LLNL, waste management operations are the responsibility of the Hazardous Waste Management Division (HWM), a part of the Laboratory's Environmental Protection Department (EPD).

Both federal (40CFT264.15 - General Inspection Requirements) and State of California (22CCR666264.15 - General inspection Requirements) regulations require that an owner or operator of a hazardous waste management facility must inspect the facility for malfunctions and deterioration, operator errors, and discharges that may be causing or may lead to 1) release of hazardous waste constituents to the environment or 2) a threat to human health.

These regulations provide a general framework for inspections but rely on the owner or operator to provide the specifics in terms of what units are inspected and the format of that inspection. However, the regulator has approval and audit authority over all of the above, and ultimately, the inspection schedule and its particulars are subject to regulatory approval.

The owner or operator must develop and follow a written schedule for the inspection (i.e., an inspection log or checklist) of monitoring equipment, safety and emergency equipment, security devices, and operating and structural equipment (such as dikes and sump pumps) that are important to preventing, detecting, or responding to environmental or human health hazards. The inspection schedule must be kept at the facility and the schedule must identify the types of problems that are to be looked for during the inspection.

Specifically, the regulations require that aboveground portions of all tank systems be inspected once each operating day to detect corrosion or release of waste. Regulations covering tanks also include a schedule and procedure for inspecting overall controls. Data gathered from monitoring and leak detection equipment must be inspected to ensure that the tank system is being operated according to its design. The construction

materials and the area immediately surrounding the externally accessible portion of the tank system, including the secondary containment system, must be inspected daily to detect erosion or sign of releases of hazardous waste. The owner/operator must document all of these items on a daily basis.

Once each week (every 7 days), all container storage units must be inspected. During these inspections, the emphasis is on a wider range of issues (e.g., safety equipment, labeling, compatibility, etc.) as well as on leaks and spills.

Finally, other areas subject to spills but not always in continuous use must be inspected daily when-in-use. Examples are loading and unloading areas, container storage areas not in constant use, treatment and filtration units used intermittently, and staging areas used for temporary placement of waste during facility moves or prior to shipment.

IMPLEMENTATION OF THE REGULATIONS AT LLNL

The Hazardous Waste Management Division at LLNL has had an inspection program in compliance with federal and state regulations in place for many years. Over the years, the forms for the daily, weekly, and daily when-in-use inspections have evolved to cover the Laboratory's expanding and increasingly complex waste management and treatment needs, and the inspection process includes fairly complex corrective action and close-out procedures to make sure that health and safety issues revealed by the inspections are resolved.

Several years ago after an audit of the Laboratory's inspection schedule and process, State of California auditors requested that HWM prepare a guidance document that would summarize the Laboratory's TSDF inspection schedule and procedures and explain and comment on them as a centralized reference source for the HWM managers and personnel performing these inspections. In addition, our in-house quality assurance personnel came to a similar conclusion following an internal audit of the adequacy of HWM's inspections and the effectiveness of state/federal inspection program implementation. HWM therefore agreed to prepare and update annually an inspection schedule and guidance document. This document is not a procedure as defined at LLNL, but it has been formalized by the "controlled document" process and is readily available to all supervisors and inspection personnel. The inspection logs upon which this guidance document is based are included with the Laboratory's Part B Permit Application and its revisions and updates.

Document Development Team and Resources

The task of preparing and updating this Inspection Schedule and Guidance Document fell to HWM environmental scientists. The environmental scientist assigned the task formed a team that included an editor/writer and managers and supervisors of HWM storage yards and treatment facilities.

The environmental scientist and the editor/writer reviewed all of the existing inspection forms with the members of the team to determine the scope of each kind of inspectiondaily, weekly, and daily "when-in-use"and its general compliance with RCRA regulations. They also reviewed each individual inspection form with their contacts from HWM operations to learn the particulars of each feature of each storage area or treatment facility inspected. They accompanied inspectors on numerous daily and weekly inspections of all storage areas and treatment facilities to become knowledgeable first hand of what specifically was being inspected, how, and why. The goals of this long and sometimes arduous process were

first to learn the inspection forms and processes up close and then to reconcile, insofar as possible, differences in inspection forms and practices from one part of HWM's facilities to another as well as to standardize the forms accordingly based on the input and consent of those responsible for the inspections.

Then the environmental scientist and editor/writer set about making minor revisions in the forms and drafting the Inspection Schedule and Guidance Document. They chose an "Information Mapping"a format for the document because of its reader friendliness and organized the document around the basic types of inspectionsdaily, weekly, and daily "when-in-use," adding appendices for each type of form where samples of all the forms are collected as well as an appendix for the National Fire Protection Association (NFPA) 704 System signs and one explaining the various classes and types of fire extinguishers.

After an initial draft of the document was completed, it went through an extensive review process that included HWM management and supervisory personnel, a seasoned representative from the EPD's Operations and Regulatory Compliance Division (ORAD), and experienced HWM inspection personnel. The guidance provided by the document was revised repeatedly and extensively based on the input of these various reviewers before its ultimate review and approval by top HWM managers, publication, and distribution.

The Document Development Process

To provide a sense of how we developed the guidance provided by the Inspection Schedule and Guidance Document, we will focus on one part of one inspection form and the development of the guidance provided there. We have chosen the General Facility section of the Weekly Inspection Log for Area 514 Facility Storage Units. Figure 1 shows the entire Weekly Inspection Log for the Area 514 Facility Storage Units; Fig. 2 shows the General Facility Section of that log. We will concentrate on the part of the General Facility section dealing with emergency equipment, Question 10, and discuss how our research, team interaction, interviews, tours, and staff reviews led us to the guidance provided by the document.

Fig. 1

Fig. 2

Question 10 of the General Facility section of all weekly logs for all facility storage units is the same and asks for evaluation of the functionality and accessibility of fire extinguishers, showers, eyewashes, and telephones/paging system. According to our investigation of the inspection process, this part of the weekly inspection logs did not need to be revised, but it did need extensive clarification and explanation.

Fire Extinguishers. Through discussions with facility supervisors, HWM inspection personnel, and our ORAD contact, we discovered that all fire extinguishers in HWM's facilitiesand, indeed, at the entire Laboratoryare examined at least every 12 months by the Laboratory's Fire Department to ensure that they are charged and in proper, safe working order. This inspection is documented on an inspection tag attached to the extinguishers. Thus, one of the first jobs of an HWM facility inspector is to examine the inspection tag to see that the extinguisher has been inspected within the last 12 months. They then look at many of the same features of the extinguishers that are examined by Fire Department inspectors. For example, they must inspect the no-tamper seal attached to the pin that must be pulled to activate the extinguisher to see if the

seal is present and remains unbroken. They look at the pressure gauge to see that it is in the operating range. They also examine other salient features of the extinguisher, such as the hose and nozzle, for defects and wear. And they affirm that the extinguisher is unobstructed and visible, that the operating instructions on the nameplate are legible and facing outward, and that the unit is properly mounted off the ground or floor. Finally, the HWM inspector dates and initials the inspection tag. If, however, all is not well with a given extinguisher or extinguishers, the inspector notes this fact on the inspection form and follows the guidance in Section IV. Corrective Action and Close-Out to correct the problem(s).

As a supplement to the section on fire extinguishers, we included an appendix on the types of fire extinguishers available. We used the National Fire Code and a booklet entitled "Fire extinguishers: Your small fire NFPA Defense" to develop the appendix. The information is intended to serve as a quick reference for the technicians who attend specific training on this subject. We included the symbols for the classes of fires and types of extinguishers and other symbols that might be found on the extinguisher. These symbols include pictographs indicating what type of material the extinguisher is used for, e.g., trash, wood, liquids, grease, electrical equipment. Appendix E also explains the relative rating on Class A and Class B extinguishers. It also has an important reminder to the HWM inspector. In the event of a fire in the treatment and storage facilities, employees should call 911 immediately to report the fire. They should use a fire extinguisher to fight a fire ONLY if they are trained to do so and know, based on the training, that the extinguisher is appropriate to the kind of fire present.

Showers. All emergency showers and eye washes in HWM facilities are physically collocated, but because they have different purposes and function somewhat differently, they are covered separately in the guidance document so that the inspector will look at each piece of equipment in its own right. Both, however, must be accessible if they are to be used in an emergency. Therefore, one of the inspector's primary jobs is to make sure that the "Keep Clear" caution at the unit is being observed and that the unit is unobstructed and can be accessed quickly in an emergency.

From the experience of HWM personnel and that of our advisors in ORAD, we learned that other basic concerns with the showers are that the water 1) must be easily and quickly turned on, 2) must stay on by itself, and 3) should flow at a constant rate and sufficient pressure to do its cleansing work. We have provided guidance accordingly. In addition, we caution inspectors to capture the water generated by their tests in a large bucket stored near the shower so that the bermed area around the shower stays dry and the inspection tests do not, therefore, create a cleanup problem. Satisfied that the shower is working properly, the inspector initials and dates the tag attached to the shower. Again, if there are problems with a shower, the inspector notes the deficiencies on the inspection form and follows the guidance in Section IV of the guidance document to correct the problem.

Eye Wash(es). Basically, the inspection guidance concerns for the eye washes are the same as those for the showersflow initiation, rate, pressure. However, because the water from the eye washes is used to cleanse the eyes, it is important that the inspector also checks to see that it is free of rust by checking water clarity and color. Also, the

eyewashes have two nozzles, both of which should be operating at equal pressure and flow rate, and these nozzles are protected when the eye wash is not being used by caps. The water pressure should be sufficient to pop these caps off when flow is initiated; the caps need to be replaced after inspection and testing.

Telephone(s)/Paging System. Phones are found throughout our waste management facility. However, some of the phones are rarely used, so it is important to test them on a weekly basis to assure that they are working. The inspector tests the phone by calling the HWM operator. HWM's paging system is connected in the two largest of our facilities, the Facility 612 Area and the Facility 514 Area. To test them, the inspector calls the appropriate number listed in the guidance document. Again, the inspector reports his finding in the prescribed way on the inspection form and schedules remedial action if a deficiency is found as described in Section IV of the guidance document.

To illustrate the document development process and how it influenced the guidance provided, we have intentionally chosen a simple example with obvious health and safety implications. We should note, however, that the Hazardous Waste Management storage and treatment facilities inspected at Lawrence Livermore are numerous and include complex, sophisticated waste treatment equipment and storage areas containing many kinds of both hazardous and low-level radioactive waste. Yet the process used to develop guidance for this equipment and these storage areas was essentially the same as that used for the less complex emergency equipment discussed above. Through a variety of meansaccompanying inspectors on their rounds, discussions with facility personnel and managers, reviews by in-house experts, research of the regulationswe sought to get to the heart of the area or piece of equipment being inspected, to see what made for its safe operation within regulatory guidelines. Our goal was to provide the inspector with a clear sense of what to look for and why in order to assure a thorough and complete inspection. Such inspections are an important way of promoting the safe operation of equipment and storage of waste as well as the health and safety of those who use that equipment and maintain those storage facilities.

RESULTS

The Inspection Schedule and Guidance Document puts the inspection process in Lawrence Livermore's waste treatment and storage facilities into a comprehensive plan. Rather than a set of individual forms and a loose set of procedures for regulatory-driven inspections, we now have an organized approach to the complete inspection process that can serve as a refresher for experienced inspectors and a training tool for novices.

In one place, HWM inspectors and their managers can find:

The inspection logs.

A discussion of the rational behind and the regulatory basis for inspections.

A schedule of inspections.

A compendium of what facilities get inspected how frequently.

Procedures for documenting inspection findings, initiating repairs, and following up to ensure that a deficiency has been corrected.

And perhaps most important of all, what to look for in particular when doing an inspection of a specific area or piece of equipment.

The Information Mapping format of the document makes it easy to follow and use. And an annual update of the inspection Schedule and Guidance

Document ensures that the information about the inspection logs, the inspection process, and the equipment and facilities being inspected is kept current.

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U.S. DEPARTMENT OF ENERGY URANIUM MILL TAILINGS REMEDIAL ACTION PROJECT ENVIRONMENTAL SAFETY AND HEALTH AUDIT PROCESS

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ABSTRACT

The U.S. Department of Energy (DOE) Uranium Mill Tailings Remedial Action (UMTRA) Project has, over the last three years, developed an Environmental, Safety, and Health (ESH) audit process that is continuing to demonstrate positive results. The audit process is a cooperative effort between DOE and DOE contractors, the Remedial Action Contractor (RAC), and the Technical Assistance Contractor (TAC).

The UMTRA audit process stresses site compliance with numerous code enforcement authorities using best management practices to design site safety and health procedures and site operating procedures that can be applied to all sites.

The audit process demonstrates proactive ESH programs that are reducing incident rates on the DOE UMTRA Project. The UMTRA ESH Audit Process paper discusses the cooperative audit process.

UMTRA ESH AUDIT PROCESS

The U.S. Department of Energy's (DOE) Uranium Mill Tailings Remedial Action (UMTRA) Project is an environmental restoration and protection program. The UMTRA Project's congressional mandated mission is to clean up and control residual radioactive materials from designated DOE-owned inactive uranium processing sites. Also, the UMTRA Project is required to eliminate present and future environmental health hazards that may result from those sites.

The DOE goal is to achieve its mission in a manner that protects the public and the environment; ensure safety and health of workers; meets or exceeds requirements of applicable federal, state, tribal, and local laws, regulations, permits; and uses "best management" practices.

The DOE policy is to conduct operations in compliance with both the letter and the spirit of applicable environment, safety, and health

statutes, regulations, and standards. The DOE strives for continuous improvement and excellence rather than just meeting minimum compliance requirements. Thus, the UMTRA Project is firmly committed to implementing the most technologically advanced environmental protection practices, while following proactive safety and health management policies and goals.

The UMTRA Project is committed to sound environmental management and worker safety practices; minimizing risks to the environment, the public, and workers; while anticipating and addressing potential problems before they pose a threat to workers, the environment, or the public.

The UMTRA Project is committed to employing "best management" practices to minimize generation of contaminants, wastes, and other residual materials requiring disposal or release to the environment through source reduction, recycling, treatment, and other pollution-prevention methods.

Protection of the environment, workers, and the public is of paramount importance to the UMTRA Project. Project administration has always recognized this, and as a result, the Project experiences a level of public trust not evident in many environmental restoration endeavors.

DOE's policy is that all contractors must incorporate this commitment to worker safety and environmental protection in all of their activities. The DOE requires its contractors to conduct their operations in an environmentally sound and safe manner that reduces risks to the environment, workers, and the public to as low as reasonably achievable (ALARA). The UMTRA Project actively oversees its contractors' activities to ensure compliance with this policy.

The DOE developed the UMTRA Project using a Remedial Action Contractor (RAC) to perform field restorations at 24 former uranium mill sites located mainly in the western United States. To assist DOE UMTRA Project management in administering environmental, safety and health (ESH)policies on the RAC-controlled sites, a Technical Assistance Contractor (TAC) was retained. The TAC provides technical expertise to DOE UMTRA Project management.

The UMTRA Project, RAC, and TAC management have developed an audit process to ensure ESH procedures implemented at UMTRA sites are adequate for worker, public, and environmental protection and comply with all applicable regulations.

Audits are scheduled to correspond with site construction schedules to place UMTRA, TAC, and RAC auditors at UMTRA sites during times of critical operations. Auditors provide additional assistance and resources to site ESH personnel. Programmatic and records review audits are conducted throughout the operating seasons. Audits are conducted a minimum of three times per construction season at each UMTRA site. Audits are conducted more frequently at sites experiencing elevated ESH incidents. Unannounced audits are conducted throughout the construction season at all sites.

Audit reports are DOE documents, prepared for the UMTRA ESH manager by the TAC audit group. Reports focus on key ESH aspects of the fundamental implementation of an adequate ESH program at the UMTRA sites. ESH procedures, training, implementation, operations or practices that are less than minimum code requirements are reported as Findings. Comments that will improve ESH programs are expressed as Observations and Recommendations. Findings must be acted upon immediately by site personnel while Observations and Recommendations may or may not be acted

upon by RAC ESH personnel. UMTRA ESH management may require the site to respond to Observations and Recommendations with an action plan.
RAC ESH groups that have developed innovative ESH procedures or operating procedures, called Noteworthy Practices, are recognized by comments in the audit report. The comments describe the practice and are sent to all sites for possible application.
The audit process begins when the UMTRA ESH manager hosts a monthly scheduling meeting with UMTRA, RAC, and TAC ESH personnel. A tentative schedule is developed for each UMTRA site for the anticipated construction season. Audit schedules are flexible so audits can be conducted at the most opportune time to cover as many construction aspects as possible. Also, audit schedules must remain flexible so circumstances of personnel, equipment, weather, and site operations can be considered. Cooperation between all participants makes the audits a very valuable tool in providing effective health and safety programs on the UMTRA Project.
The UMTRA ESH manager is responsible to notify the RAC ESH manager in writing of the actual date the announced audit will be conducted. Notification is required ten (10) working days prior to the site audit starting date.
Upon arrival at the site by all personnel necessary, the DOE UMTRA Project ESH Quality Assurance Implementation Plan is followed. The opening meeting is held, all necessary site orientation and training of auditors are completed, and then the audit is conducted.
DOE UMTRA PROJECT ESH QUALITY ASSURANCE IMPLEMENTATION PLAN
INTRODUCTION
Provide guidance for use during DOE auditing activities at the UMTRA site.
POLICY
Statement requires DOE and contractors to take all reasonable precautions in the performance of UMTRA Project work to protect the health and safety of workers, the public, and the environment.
DEFINITIONS
Audit Opening Meeting
Daily Close-out Meeting
Site, Final Audit Close-out Meeting
Final Audit Close-out
Objective Evidence
Finding
Noteworthy Practice(s)
Observations
Recommendations
Stop Work Authority
SEQUENCE OF AUDIT
Audit Preparation
Site, Written Notification
Audit Opening Meeting
- Introductions
- Logistics
- Schedules
- Site-Specific Training

Audit Activities in the Field
- Contractor representative with auditors

Daily Close-out Meeting

Discussion of daily objective evidence
Site Close-out Meeting
Discussion of all objective evidence
Provide copy of all objective evidence
POST AUDIT ACTIVITIES
Initial Draft to DOE ESH manager 10 Days from site close-out meeting
Meet with DOE, RAC, and TAC to discuss Initial Draft Final Draft to DOE ESH manager 10 Days following draft discussion
RAC Corrective Actions Response to DOE ESH manager 10 days from receipt
DOE Review and Request Additional Responses or Close Audit
Post Final Audit Report at Contractor Site
AUDIT RESULTS REPORTING
Additional Site Conditions, Deficiencies, or Noteworthy Practices
AUDIT RECORDS
Complete Audit Files To Document Control
The UMTRA, RAC, and TAC auditors are escorted by the RAC site ESH management personnel and sub-tier contractor ESH representatives. All personnel travel together at the site(s) to observe the physical plant, operations, and personnel safety performance.
Whenever a question concerning health and safety of personnel, public, or property is raised, the site RAC ESH manager is made aware of the situation, and the information is recorded as "objective evidence." Objective evidence collected during the day is discussed with the UMTRA, RAC, RAC sub-tier contractors, and TAC representatives each day during the daily close-out meeting. Audit ESH checklists are used throughout the audit process.
Audit checklists have been developed from 29 CFR 1910 and 29 CFR 1926 for general health and safety codes, construction and demolition codes, and industrial hygiene issues. The use of checklists is for review by the auditors and are not specifically checked at each audited site.
Auditor on-site participation is completed upon conclusion of the final audit close-out meeting where all objective evidence is discussed with UMTRA, RAC, and TAC representatives. Objective evidence is compared to codes, regulations, and best management practices. Results obtained by reviewing the objective evidence is placed in the final audit report.
The UMTRA, RAC, and TAC auditors recognize that site work will be on-going during the audit, and attempts to minimize disruptions to site work are made. Site operational/ESH responsibilities come first and RAC and subcontractor site ESH personnel are to leave the auditors and manage their responsibilities to keep work activities safe.
The UMTRA, RAC, and TAC auditors follow the DOE UMTRA Project Audit Quality Assurance Implementation Plan and develop the site Health and Safety Audit Report.
HEALTH AND SAFETY AUDIT REPORT
INTRODUCTION
UMTRA site contractors, personnel, and personnel responsibilities.
BASIS FOR AUDIT
DOE, state, tribal, and local codes that influence operations at the site.
SITE OPERATIONS AND CONDITIONS
Current operations and conditions at the site.
DEFINITIONS
Objective Evidence
Findings

Observations
Recommendations
Noteworthy Practices
AUDIT SUMMARY
Audit team summary of audit, including specific objective evidence to support summary.
AUDIT RESULTS
Objective Evidence is discussed to demonstrate influence on operations and health and safety of workers, the public, and the environment. Findings are condition(s) as found, that violate specific codes. Observations and Recommendations are based upon auditor experience to improve operations. RAC response is not required unless UMTRA ESH manager requests.
LIST OF CONTRIBUTORS
All personnel conducting the audit, review, and technical investigations.
The ESH audit report format has been standardized for this project and is used for all sites.
The TAC auditors complete the final audit report and submit it to the UMTRA ESH manager for transmittal to the RAC. The TAC maintains records of all audits and subsequent correspondence for audit closure. This record provides valuable tracking of audit documents and is used to audit internal performance.
DOE UMTRA SITE AUDIT RECORD
  Site
  Date -- Doe Notification Letter To RAC
  Date -- Actual Audit Dates
  Date -- Draft Mailed to RAC for Review
  Date -- Draft Returned From RAC Review Date -- Final Draft Mailed to RAC
  Date -- RAC Response
  Date -- DOE Acceptance or Rejection of RAC Response
  Date -- DOE Audit Closure.
The audit process and documents currently developed for UMTRA audits have been in use for three audit seasons. The process is flexible in application but remains independent of biased influence and accurately reflects current ESH conditions at each site. The UMTRA, RAC, and TAC auditors must be flexible in approach to achieve the best possible ESH program for the UMTRA Project. Also, auditors must focus independently on the ESH issues and be determined to achieve quality ESH programs. Audits from each UMTRA site are shared with management at all other UMTRA sites. The audit process followed on UMTRA sites has produced positive results in program development, training, implementation, accountability, and has reduced the number of incidents and losses at UMTRA sites. RAC and TAC contractor records indicate incident rate at approximately 10-15% of the rate other contractors experience in similar industry categories.

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DEVELOPMENT AND IMPLEMENTATION OF WASTE MINIMIZATION TRAINING AND AWARENESS MARKETING PROGRAMS AT DEPARTMENT OF ENERGY FIELD OFFICES
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ABSTRACT
Waste Minimization and Pollution Prevention (WM/PP) programs are integral elements of waste management operations at Department of Energy (DOE)

field offices. One of the major challenges for these field offices is effective accomplishment of the activities required to develop, implement, and oversee training and awareness programs in support of a comprehensive WM/PP program.

WM/PP Training Programs involve the development of a Waste Minimization Program Training System and related courses for personnel involved in waste minimization programs, as well as for general site employees. The WM Program Training System will provide input for the updating of site-level general training courses to the latest WM/PP program requirements. The programs may consist of the following elements.

1) Development of a core Waste Minimization Training Program;
2) Updating the Waste Minimization Sections of site training courses such as Comprehensive Annual Training, General Employee Training, and Radiation Worker Training;
3) Evaluating and updating WM training modules for technical personnel such as design and operations engineers; and
4) Evaluating and updating training requirements and related training modules for Pollution Prevention Opportunity Assessments.

A properly implemented WM/PP Awareness and Marketing Program will support the development and implementation of efforts targeted to the DOE site office as well as for DOE prime contractors and subcontractors. This program may consist of the following elements:

1) Program planning -- This effort addresses development of a Pollution Prevention Awareness and Marketing Program Plan for implementation of a comprehensive pollution prevention education campaign.
2) Documentation of Accomplishments -- This involves documentation of WM-PP accomplishments and developing highlights in terms of reduced cost, site savings, and disposal avoidance.
3) Implementation of a "Green Building" Program -- This includes development and maintenance of a program to promote building-specific opportunities for WM/PP.

The proposed approach to WM/PP programs, development of modules for WM/PP training, and the targeting of these modules for specific audiences presents a specialized approach that may be of benefit to any personnel involved in WM/PP programs. An effective WM/PP training and awareness program may be implemented by targeted training and promotion of WM/PP concepts through an awareness campaign.

BACKGROUND AND OVERVIEW

WM/PP programs are becoming increasingly important elements of integrated environmental operations for industrial and federal facilities, and in particular at DOE field offices. At many sites, WM/PP operations have been instigated by organizations responsible for large-scale oversight programs such as waste management (WM), environmental restoration (ER), and decontamination and decommissioning (D&D). For these programs, WM/PP goals are accomplished by:

Administrative planning and controls to limit the generation of primary and secondary wastes, and to avoid creating uncontrolled releases;

Engineering considerations for systems, equipment, processes, and technologies to minimize the generation of wastes and to manage pollution and waste streams; and

Incorporation of operational concepts such as reuse, recycling, and free release to minimize the volumes of wastes and pollutants generated during site projects.

Due to the limited capacities of on-site waste treatment and pollution control facilities at many sites, the concepts of waste minimization and pollution prevention have become critical elements of continued operations and of cleanup efforts. The limited availability of adequate off-site waste treatment and disposal capacity and the protracted delays in permitting of planned disposal sites for wastes have contributed to the elevated emphasis on WM/PP programs.

Parsons Engineering Science (Parsons) has been active in providing support for the analysis of WM/PP opportunities and in the development and implementation of WM/PP programs. WM/PP programs have been designed to support the planned and future ER, WM, D&D efforts which are considered to be a major portion of the activities to be conducted at aging industrial and federal facilities. Since these programs and process have the potential to generate considerable amounts of radioactive, hazardous, and mixed waste products, the application of the concepts of WM/PP are paramount to effective and efficient environmental compliance and protection program management.

WM/PP PROGRAM OBJECTIVES

Most WM/PP programs have common bases, or objectives that necessitate the development and implementation of such programs. These bases must be integrated into the development of training and awareness programs. In most cases, WM/PP programs are designed to:

Foster a philosophy to conserve resources and create a minimum of waste and pollution in achieving environmental management objectives.

Promote the use of nonhazardous materials in ER, WM, D&D operations to the extent practicable in order to minimize the potential risks to human health and the environment;

Enhance employee awareness of pollution prevention goals, objectives, and methods;

Enhance communications of waste minimization objectives, goals, and ideas laterally and vertically among site/facility organizations;

Promote integration and coordination of waste generators and waste managers on waste minimization matters;

Create incentives for pollution prevention; and

Collect and exchange waste minimization information through technology transfer, outreach, and educational networks.

All of these elements must be included in comprehensive site-wide WM/PP programs, and emphasized as functional bases for training and awareness programs development.

Implementation of WM/PP policies and procedures is a function typically assigned to a specific site organization associated with waste management programs. This organization will oversee the publication of site-wide procedures that evolve from policy/program implementation. New procedures must be written, or existing procedures revised, to ensure that WM/PP goals and ideas are put into practice. Other means of communication such as presentations and articles in employee newsletters should be utilized to the extent practicable. Employees should be encouraged to attend seminars and to communicate with other sites on WM/PP practices.

REGULATORY BASIS FOR WM/PP PROGRAMS

Other than the advantages of voluntary establishment of WM/PP programs, there are regulatory 'drivers' and incentives that give emphasis to WM/PP program implementation. At the federal level, direction may be derived from specific sections of:

The Resource Conservation and Recovery Act (RCRA);

The Comprehensive Environmental Response. Compensation and Liability Act (CERLCA);

The Superfund Amendments and Reauthorization Act (SARA, of CERCLA), Title III, which includes the Emergency Planning and Community Right-To-Know Act, Sec. 313;

Executive Order 12856, Federal Compliance with Right-To-Know Laws and Pollution Prevention Requirements; and

The Pollution Prevention Act (PPA).

These laws are augmented by other EPA guidance that has been released in published format, such as "Waste Minimization Certification" (09/85), "Waste Minimization" (10/85), "Waste Minimization Requirements" (06/87), and "EPA Waste Minimization Program Activities." (06/87).

In typical applications, these rules and guidance are supplemented by program- or site-specific guidance. For example, the U.S. Department of Energy (DOE) derives guidance for WM/PP programs at DOE nuclear weapons production sites under several directives:

DOE Order 5400.1, General Environmental Protection Program

DOE Order 5400.3, Hazardous and Radioactive Mixed Waste Program

DOE Order 5820.2A, Radioactive Waste Management

DOE/S-0108, DOE Strategic Plan

In order to define, establish, and implement an effective WM/PP program, it is important to understand the directives under which this program must perform. When incorporated into the training and awareness program, these regulatory and statutory bases provide credence to WM/PP implementation.

TYPES OF WM/PP PROGRAMS

In most instances, an integrated program of WM/PP involves activities that take advantage of every opportunity for effectiveness. The types of WM/PP programs that may be developed, implemented, maintained and that must be addressed training include:

Source Reduction, which is the major priority in WM/PP programs; eliminates problems associated with waste handling; and includes concepts such as materials substitution, process optimization or modification, technology changes, and administrative changes (including inventory control and housekeeping practices such as waste segregation).

Recycle, Reuse, Reclamation, which is used for wastes that cannot be eliminated or minimized; allows potential waste materials or materials destined for disposal to be put to beneficial use; and includes resource recovery (use, reuse, reclamation).

Treatment of Wastes, which includes techniques to reduce the volume, toxicity, and/or mobility of waste streams and waste forms.

ELEMENTS OF WM/PP TRAINING

WM/PP training typically is performed by presentation of audio/visual material. The course presentation may be classroom, computer based, video, or self-study as deemed appropriate by the responsible organization. The course materials should be approved and entered into the site training database in order to ensure that the site program receives credit for accomplishments. Specific task elements required to establish a WM/PP training program may include:

Conduct of Job/Task Analysis;

Description of Course Terminal and Enabling Objectives;

Development of Lesson Plans;

Development of Study Guides;

Preparation of Training Support Material, including graphics, visual aids, computer-based training, case studies, and/or printed material for handouts, as appropriate;
Development of Test Plans;
Conduct of Examinations;
Issuance of Course Evaluation Forms; and
Documentation of Results.

Training activities for WM/PP are most effective when combined with other [existing] site-wide training programs. The goal is to make each employee and contractor aware of waste generation, its impact on the site and the environment, and means by which waste can be reduced and pollution prevented. The training program should include waste minimization concepts that can be incorporated into every job aspect, thereby ensuring that these concepts will be integrated into each phase of an operating procedure or design consideration.

PROGRAM PLANNING FOR WM/PP

WM/PP program planning involves the development of a Pollution Prevention Awareness and Marketing Program Plan for implementation of a comprehensive pollution prevention education campaign directed at both internal and external audiences. Key elements for an effective internal communications program will include developing information products that are customized for use within each DOE site employee communication framework, and providing liaison with the site organizations responsible for disseminating the information.

Key elements of the external communications program will include developing information products appropriate for external audiences, including other DOE sites, commercial industries and the general public, and coordinating with the site public affairs department in implementing offsite release of information.

Key elements of the external communications program will include developing information products appropriate for external audiences, including other DOE sites, commercial industries and the general public; providing liaison with offsite photographers/videographers or other documentary services during production of program elements; and coordinating with the appropriate DOE site public affairs department in implementing offsite release information.

DOCUMENTATION OF WM/PP ACCOMPLISHMENTS

In order to foster the culture of an effective, implemented WM/PP program, the awareness and training organization should document accomplishments and display this documentation in a manner that provides convenient employee access. The elements of this process may include: development and updating of a book (a living document with scheduled updates) to document WM/PP active programs and related accomplishments; gathering information on pollution prevention accomplishments by fiscal year; and developing text highlighting accomplishments in terms of reduced cost, site savings, and disposal avoidance.

IMPLEMENTATION OF A "GREEN BUILDING" PROGRAM

The "Green Building" concept has been popularized throughout the federal government, and has a high notoriety for involvement of personnel at the various levels of an organization. The idea is that a "Green Building" is a more environmentally friendly facility, providing some impetus for productive competition among corresponding units. This concept involves: initiating and promoting a "Green Building" waste minimization program that highlight building-specific opportunities for WM/PP; maintaining

open lines of communication with custodians of site buildings that have committed to the program of WM/PP; pursuing the "Green Building" program in non-participating buildings by developing education materials on recycling, eliminating waste streams, and the benefits of each; and encouraging site management support of the "Green Building" program by developing presentations with programmatic recommendations.

EMPLOYEE AWARENESS PROGRAMS

The WM/PP training program has a very important function: increasing the awareness of employees and contractors on topics of WM/PP. Awareness programs may include display of WM/PP progress and success stories by use of pictorial essays, graphs and charts that track specific elements, site bulletin boards, and communications utilizing site media (e.g., newsletters, electronic mail, routine staff meetings, etc.).

Slogan contests have been conducted routinely as awareness program activities. By involving the site personnel in these contests, the awareness of WM/PP issues is heightened.

Incentives such as awards, plaques, and other forms of recognition may be utilized to provide motivation, and to boost employee cooperation and participation. Meeting WM/PP goals provides a measure for evaluating the job performance of managers and employees through the performance management system and applicable performance objectives.

Suggestion boxes provide a vehicle to elicit participation from employees. Awards may be given to employees whose suggestions result in gains in WM/PP objectives such as volume or cost reduction.

PPOA TRAINING

A specialized and very important area of WM/PP training is Pollution Prevention Opportunity Assessment (PPOA) training. A performance-based WM/PP training course focusing on the PPOA program should be designed to examine methods for developing, conducting and using waste minimization audits to determine PPOA's. This type of course should highlight key elements and protocols for effective audits. The course is to be entered into the site training computer database. D&F will provide support in the form of instruction of the pilot presentation of the course to a selected audience in order to receive comments and incorporate changes prior to turnover.

BENEFITS OF WM/PP TRAINING AND AWARENESS PROGRAMS

A well designed and implemented WM/PP training and awareness program will yield the following benefits:

Reduction or elimination of liabilities associated with the generation of project wastes;

Compliance with applicable regulations;

Reduction of waste management and compliance costs;

Reduction of resource usage;

Reduction or elimination of reportable inventories and releases of hazardous chemicals; and

Improvement of public perceptions and concerns.

In order to achieve these benefits, the effective implementation of a WM/PP training and awareness program must include certain iterative and cyclic elements that ensure that appropriate measures are conducted.

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Session 24 -- INTERNATIONAL REGULATORY AND WASTE MANAGEMENT EXPERIENCE
Co-chairs: Jas S. Devgun, Consultant
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24-1

INDEPENDENT MONITORING OF SOLID LOW LEVEL RADIOACTIVE WASTE DISPOSALS IN THE UK

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ABSTRACT

Her Majesty's Inspectorate of Pollution (HMIP) has responsibility for authorizing and regulating releases of radioactive waste to the environment in England and Wales under the Radioactive Substances Act 1993. To ensure that solid low level radioactive waste (LLW) disposals are within authorized limits, waste consignments are seized before disposal and their contents assessed. This paper outlines the regulatory framework for control of LLW disposal in the UK, describes the methods used for independent assessment and indicates the findings obtained.

INTRODUCTION

UK Legislation

The disposal of radioactive waste to the environment is subject to the provisions of the Radioactive Substances Act 1993 (RSA'93) (1). Although a recent Act its purpose was to consolidate an earlier one, the Radioactive Substances Act 1960 (RSA'60) (2) with amendments introduced by subsequent legislation including Part V of the Environmental Protection Act 1990 (3). RSA'60 also replaced earlier legislation; the Radioactive Substances Act 1948. There is therefore a long-established legal framework in the United Kingdom for exercising regulatory control over radioactive wastes.

Authorization

Limits and conditions on the disposal of radioactive wastes are detailed in site specific Authorization Certificates. Over 1100 sites in England and Wales are authorized. The majority of these consist of hospitals, universities and industrial research or manufacturing centers. The more significant radioactive discharges however are from a relatively small number of sites licensed under the Nuclear Installations Act 1965 (4). These are generally referred to as "nuclear sites" and are also authorized under RSA'93 to discharge radioactive wastes. These nuclear sites include nuclear fuel fabrication and reprocessing plants, nuclear

power plants, atomic research establishments and isotope production centers.

Regulatory Authorities

Her Majesty's Inspectorate of Pollution (HMIP) is responsible for administration and enforcement of RSA'93 in England and Wales. In the case of nuclear sites in England, Authorizations for disposal of radioactive waste are issued jointly by HMIP and the Ministry of Agriculture, Fisheries and Food (MAFF). In Wales these functions are undertaken by HMIP and the Welsh Office with the support of MAFF. Separate but similar arrangements exist in Scotland and Northern Ireland where Her Majesty's Industrial Pollution Inspectorate and the Alkali and Radiochemical Inspectorate are the respective regulatory authorities.

Independent Monitoring

Operators are required to determine and record the radioactive content of waste disposals in accordance with conditions specified in authorizations.

In support of its regulatory function HMIP undertakes its own monitoring programmes to act as both a check on site operators' results and to provide independent data on the exposure of the public. These results are published annually (5). As part of these programmes HMIP carries out independent checks on solid LLW destined for land disposal. Consignments of waste are seized by HMIP Inspectors and sent to HMIP's Waste Quality Checking Laboratory (WQCL). The present paper focuses on this process and the subsequent checking procedure and describes how HMIP uses the results to obtain added assurance that disposals are in compliance with disposal Authorizations.

SOLID LLW ARISINGS AND DISPOSAL ROUTE

Low Level Radioactive Waste

In the UK, radioactive waste is classified under broad categories, according to its heat - generating and activity content. Low level waste (LLW) is regarded as waste containing radioactive materials other than those acceptable for disposal with ordinary refuse and with activity contents not exceeding 4 GBq/te of alpha emitting radionuclides or 12 Gbq/te of beta/gamma emitting radionuclides.

The largest volumes of solid LLW originate from the fuel cycle plants operated by British Nuclear Fuels plc (BNFL), the Magnox, AGR and PWR power stations operated by Nuclear Electric plc, and the research establishments of the UK Atomic Energy Authority and Ministry of Defence. More than 99% of the waste seized has originated from these nuclear sites and was destined for disposal at the landfill site in Drigg, Cumbria operated by BNFL for the disposal of solid LLW.

Drigg

The Drigg disposal site is located in West Cumbria about 6km south-east of BNFL's reprocessing facility at Sellafield. The site started operations in 1959. It receives waste mainly from Sellafield but also from other nuclear and non-nuclear establishments elsewhere in the UK. The site occupies about 120 hectares (300 acres) close to the Cumbrian coast.

Wastes were historically deposited by tumble tipping into trenches cut into clay to a depth of about 8 metres. This method of disposal ceased in 1994. Suitable wastes, mainly from Sellafield, are now compacted and placed in ISO-containers in a new facility at Sellafield. After transport to Drigg the wastes are fixed in a concrete grout prior to their orderly emplacement in a concrete lined vault.

The majority of the waste typically comprises discarded protective clothing (overalls, overshoes, gloves, paper hats etc.) and general trash from areas of low contamination. The waste is generally accumulated in 200 litre drums and the total activity of such a drum is typically 1 to 2 Mbq beta/gamma but can vary between 1 Kbq and 20 Mbq due to the inherent inhomogeneity of this type of waste.

At this stage the waste will be un-compacted and smaller items are likely to be in plastic bags within the drums. Waste generators are increasingly using compaction and there are also some sites which utilize super-compaction. It is HMIP policy to seize waste for independent monitoring before it has been super-compacted. The majority of consignments selected have been taken after they have been delivered to the Drigg site but as some waste may be compacted prior to delivery consignments are also seized from sites before despatch to Drigg.

WASTE QUALITY CHECKING LABORATORY

The independent monitoring or quality checking of solid low level radioactive waste is carried out at a laboratory established by HMIP for this purpose at the Winfrith Technology Centre in Dorset, England. The Waste Quality Checking Laboratory (WQCL) was first postulated in 1983 and a contract to build the laboratory was awarded to a consortium lead by Taylor Woodrow in 1985. During this first contract the design and construction of the facility were completed and the laboratory was equipped and staffed by scientists and technicians. The work took approximately three years to complete with the laboratory being commissioned in 1988.

From 1988 to 1991, the laboratory was contracted to perform research into the analytical methods required to identify and quantify the wide variety of radioisotopes which can potentially be found in LLW. During this period valuable foundation work on the quality system and quality checking techniques were completed.

The present contract to operate and staff the laboratory was awarded to Taywood Environmental Consultancy, one of the Taylor Woodrow group of companies in 1991. The main emphasis within this contract is the routine quality checking of solid low level radioactive waste consigned for disposal in England and Wales in support of HMIP's regulatory work.

Laboratory Description

The WQCL monitoring facility is located on the Winfrith nuclear licensed site operated by the UKAEA. This provides facilities for consignments of radioactive waste to be received at the laboratory and secondary waste arisings to be disposed of via site services. The laboratory also makes use of other site services such as health physics, dosimetry and site security. The facility is housed in part of a large building, once used for an experimental reactor, and consists of a suite of offices, laboratories, a workshop and waste receipt and storage areas.

Waste consignments can be transported to the laboratory in a variety of containers, these include full height and half height ISO freight containers, individual drums and loose or packaged waste in skips.

Following acceptance of the waste at Winfrith and receipt of the consignment at the WQCL, the transport container is moved to a suitable area for unloading. For most consignments this takes place at the WQCL waste receipt area, where the container undergoes a series of checks prior to opening and unloading. These are described in more detail under non-destructive testing. For drummed waste received in ISO freight containers the drums are unloaded and stored in this area. For loose or

packaged waste received in skips, a tented enclosure can be erected for repacking the waste into drums in preparation for non-destructive testing.

In addition to the waste receipt area the ground floor of the facility also houses the gamma spectrometry laboratories and a temporary X-radiography facility. The upper level of the facility comprises office accommodation and a suite of radiochemistry laboratories where the destructive testing and waste sampling is performed.

Quality Assurance

All the quality checking operations undertaken at the laboratory are carried out within a quality assurance system which was developed to ensure that all the work is performed to recognized and acceptable standards and that the results reported to HMIP are accurate and reliable. The quality system together with a number of key test methods were assessed by the National Measurement and Accreditation Service (NAMAS) in November 1993 and accreditation formally awarded to the laboratory in January 1994. Since then further test methods have been assessed and accredited by NAMAS as part of an ongoing programme to maintain and develop the laboratory's QA infrastructure. Formal accreditation provides assurance that the measurements made on the waste are accurate and traceable to national and international standards.

Non-destructive Testing

Upon receipt of a waste container at the laboratory, the consignment is given a unique identification number and each transport container is examined for evidence of damage or loss of integrity, any such findings are photographed and recorded. The labels attached to the transport container are photographed and the information on the labels also recorded. Seals placed on the container by HMIP are examined and photographed. The container is also checked for non fixed external contamination and radiation dose rate and finally the gross weight and external dimensions of the container and measured and recorded.

Following completion of the transport container checks, the waste consignment is opened and the contents unloaded. For drummed waste received in ISO containers the drums are unloaded directly into the waste receipt area of the laboratory and are logged into the QA system. Further checks are carried out on the waste drums at this stage, these include, radiation dose rate measurements, contamination checks, drum weight measurement and a note of the condition and integrity of the drum.

Since the non-destructive testing procedures carried out at the laboratory require the waste to be contained within 100 or 200 litre steel drums, any waste which is not received in this format such as loose waste in skips must first be repacked into drums. This requires direct handling of the waste by operators who must be suitably protected with the appropriate personal protective equipment. Repacking operations may also require a certain amount of size reduction in order to fit bulky items such as lengths of wood or pipe into the 200 litre drums.

Each drum received or packed at WQCL is given a unique identification number and a seal is placed on the lid to prevent tampering. The non-destructive testing is carried out on the whole of the waste consignment and involves two tests, these are X-radiography and Segmented Gamma Scanning (SGS).

Firstly each drum is X-radiographed to determine its contents. A total of nine X-radiographs are taken with three equally spaced around the circumference of the drum and at three equally spaced positions down the

height of the drum. Following processing, each X-radiograph is examined by trained staff to determine the contents of the drum. This is important for the identification of prohibited items as defined in Authorization Certificates and the BNFL's Conditions of Acceptance for wastes for disposal at Drigg. These include, free liquids, aerosol canisters, materials that are likely to cause fire or explosion hazards and large amounts of putrescible or rotable materials. The identity of any drum containing non-permitted items is noted for opening and more detailed examination and any non-permitted items found are removed from the waste. X-radiography is currently carried out for the laboratory under contract to a third party organization, however it is planned to construct a permanent X-radiography facility at the laboratory and to include this work in the WQCL repertoire of non-destructive testing.

The most important non-destructive technique used in waste quality checking is Segmented Gamma Scanning (SGS). Using this technique the gamma emitting radioisotopes within each drum can be identified and quantified. Each drum is placed in turn on a lifting turntable within the instrument which allows the radioactivity within defined segments of the drum to be determined. Up to 40 segments can be defined within a single drum. A schematic diagram of the SGS is shown in Fig. 1.

Fig. 1

The total radioactivity within the drum is then calculated by adding the results from each segment. A correction for the attenuating effect of the drums waste content is made by use of an external gamma emitting transmission source. The instrument is routinely calibrated and checked using reference radioactive sources traceable to national standards.

The wide range of waste material densities together with the large number of gamma emitting radioisotopes found in LLW can give rise to significant uncertainties in the radioactivities determined by the SGS. In an effort to reduce these uncertainties and achieve NAMAS accreditation for SGS measurements, an extensive research programme has been undertaken. Experimental measurements have been made using known reference sources with a wide range of gamma ray energies placed at different positions with a series of drums which have been filled with materials of differing densities. Analysis of the data collected from these experiments has enabled the laboratory to identify areas where improvements can be made. On completion of the non-destructive testing campaign the gamma emitting radioisotopes identified are listed and the total gamma emitting radioactivity for the waste consignment is calculated for comparison with the waste producer's declaration.

Destructive Testing

In order to determine the alpha and beta emitting radioisotopes within a consignment of LLW, destructive testing must be performed on a representative portion of it. In general approximately 5% of the packages or drums within a consignment are analyzed destructively for radiochemical content. The criteria used for the selection of drums for sampling is dependent on HMIP requirements and the nature of the waste being assessed. Examination of the X-radiographs for example, may reveal prohibited items such as aerosol canisters or free liquids which must be removed. The presence of dense objects seen on the X-radiographs may conceal sources of radioactivity which may not have been revealed by SGS monitoring. Drums may also be selected from examination of the gamma emitting radioisotope content as found by the SGS and by specific request from HMIP eg based on the origin of the waste within the producer's site.

Once a drum has been selected for destructive testing it is taken to the radiochemistry laboratory and attached to the waste receipt glove box. The lid of the drum is then removed from inside the glove box and the contents of the drum are examined. The waste receipt glove box is fitted with a fixed video camera and all drum opening operations are recorded on video tape. Any prohibited items found in the drum are photographed to provide evidence of the finding and segregated from the remainder of the waste which is then transferred to a second glove box. Here the waste is packaged if necessary and the contact radiation dose rate and weight of the package are measured and recorded. Representative sub-samples are then taken and transferred to a fumehood for radiochemical analysis.

Destructive testing begins with the preparation of an aqueous solution of the solid sample taken from the waste. This can be accomplished in a variety of ways depending on the type of waste material found. Methods such as acid dissolution, fusion, digestion or leaching, are commonly used, the principal objective being the extraction of all the radioactive species into solution. Once the primary solution has been prepared, aliquots are first taken for the determination of total alpha, total beta, total and individual gamma emitting radioisotopes.

The total alpha measurements are made by preparing a counting disc from the primary sample solution by evaporation onto a planchette. This is then analyzed in one of eight alpha spectrometer cells, counting times being calculated from count rate measurements. The results are used to identify the component alpha emitters and quantify the total alpha radioactivity of the sample.

Total beta determinations are made using Liquid Scintillation Analysis (LSA). An aliquot of the sample solution is added to a vial containing a scintillation medium and is then analyzed using a liquid scintillation counter. As for the alpha measurements the results are used to identify component beta emitters wherever possible and calculate the total beta radioactivity of the sample.

Component gamma emitting radioisotopes within the sample are determined by analyzing a 50ml aliquot of the primary solution in a fixed geometry on a gamma spectrometer. A separate low energy gamma spectrometer is used to determine low energy gamma emitting radioisotopes such as Fe-55, I-125 and I-129.

For all three of these techniques the chemical and radiochemical concentration of the solution must be controlled to optimize the counting characteristics and reduce interferences. All the analysis instruments used are regularly calibrated and checked using reference sources traceable to national and international standards. To ensure that the methods used and the results obtained from destructive testing are acceptable the laboratory participates in regular inter-laboratory comparison exercises organized by HMIP.

In addition to the total alpha, beta and gamma techniques described, the laboratory has a number of other specific radioisotopic methods which can be used for destructive testing.

The determination of specific radioisotopes by destructive testing first requires radiochemical separation from the other species found in the primary sample solution. The method adopted will depend on the chemistry of the element being isolated and may involve solvent extraction, distillation or ion exchange. The laboratory has analysis methods for most of the radioisotopes commonly found in low level radioactive waste.

REPORTING AND APPLICATION OF RESULTS

All the results produced by the laboratory from quality checking operations on waste consignments are reported to HMIP. Written reports are produced on the finding of the non-destructive and destructive testing campaigns, these are then forwarded to HMIP for review.
Some typical results obtained from the quality checking of a waste consignment are shown in Table I. The tables compare the results obtained by NDT and DT for 4 different drums taken from 3 separate waste streams within a single waste consignment. It can be seen that in general there is very good agreement between the two techniques.
Table I
From the regulatory point of view the results of the checking process may be considered as being in two distinct categories. Firstly, there are qualitative issues such as whether there was free liquid in the waste or whether the waste contained non-permitted items. Secondly, quantitative results can be compared with the activity as declared by the waste producer.
Results reports are always issued to the relevant HMIP site Inspector who makes the judgement as to what action, if any, should be taken against the waste producer. Qualitative issues are usually an indication that either the operator's procedures are deficient in some way, or that the procedures have not been complied with. These are concrete issues which the Inspector would take up formally with the operator and would ensure by subsequent site inspections that adequate corrective actions had been taken. Quantitative issues can be much more complex, particularly where the results are from destructive testing and analysis. Results from SGS analyses are not dependent on sampling as all drums in the consignment are analyzed whereas for chemical analyses a proportion (typically 5%) of the drums are selected and a further selection of material within a drum is sampled. Nevertheless the correlation of total activities calculated by the two methods is generally much better than might be expected. This increases confidence in the results. If the check analyses indicated that authorized or declared activities had been exceeded further samples or analyses would be carried out to confirm the results. In all cases the follow up action is taken by the Inspector for the appropriate site and in severe cases an operator would be liable to prosecution. Independent monitoring has so far given confidence that operators have taken a responsible and thorough approach to complying with disposal authorizations. Non-compliances found have been in the nature of qualitative breaches as described above and appropriate corrective actions have been undertaken by operators.
EUROPEAN NETWORK ACTIVITIES
Since October 1992 the laboratory has participated in the European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages. This network was formed to promote co-operation between laboratories within the European Union who are involved in quality checking activities. The countries represented in this network are: Belgium, Germany, France, Spain, Italy, The Netherlands, Austria and The United Kingdom. Each country has laboratory participants and regulatory observers represented on the Steering Committee and Executive board of the network and a number of working groups have been established to focus on important aspects of quality checking. There are currently 5 working groups whose remits are: gamma measurements, volatile releases from waste packages, quality assurance, neutron measurements and destructive testing.

The Steering Committee and Working Groups meet twice a year to discuss technical issues and matters of mutual interest. The network is currently involved in jointly submitting a number of research proposals to the European Commission as part of the fourth framework programme on nuclear fission safety. This research programme attracts joint funding from the European inter-laboratory comparison test for gamma measurements on 220 litre waste drums. This exercise will have obvious benefits to the quality assurance of SGS measurements performed at the WQCL.

CONCLUSIONS

Independent monitoring carried out by the Waste Quality Checking Laboratory is a key element of HMIP's regulatory work under the Radioactive Substances Act 1993. The quality and reliability of the laboratory's work is underpinned by formal accreditation of its test methods under the UK's National Measurement and Accreditation Service and its participation in the European Network of Testing Facilities for the Quality Checking of Radioactive Waste Packages.

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ROUTINE QUALITY CHECKING OF LOW LEVEL WASTE PACKAGES BY DESTRUCTIVE AND NONDESTRUCTIVE METHODS

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ABSTRACT

Nuclear waste packages have to comply with the specifications of interim storage facilities and have to meet the waste acceptance requirements for final repositories. In the FRG, the compliance with the acceptance criteria is preferably proved by process qualification of the waste treatment. However, the quality of waste packages originating from non-qualified processes has to be checked by control measurements.

On behalf of German nuclear regulatory authorities the Research Center Jlich (KFA) currently performs quality inspections on Low Level Waste (LLW) packages. Nondestructive assay methods as well as destructive analysis methods are used for quality checking. The routine inspection procedure consists of a fixed set of basic methods applied to all waste drums transported to KFA Jlich and a variable set of extended measurements applied to selected packages for which further information is desired.

Experience has been gained in the application of these methods over a wide range of waste types to be disposed of. The paper describes the current working program and gives a survey of the inspection procedures and quality checking methods applied routinely. Examples are discussed

where extended nondestructive testing methods (e.g. digital radiography and computerized tomography) are used to validate and improve the accuracy of activity determination.

INTRODUCTION

In the FRG all types of radioactive waste including low and intermediate level waste will be disposed of in deep geological underground repositories (salt or hard rock formations). Conditioned radioactive wastes have to meet the specifications and acceptance requirements defined by national regulatory and waste management authorities in the FRG. These requirements are specified depending on the type of waste, the waste form and the type of container (1). The fulfillment of the acceptance criteria for interim or final storage is preferably proved by the qualification of waste treatment and conditioning processes. In addition quality checking of waste packages is done on the basis of random sampling inspections (2).

The Research Centre KFA Jlich is engaged in R&D activities for the quality checking of nuclear waste packages since many years. Within national and European funded projects KFA has developed various nondestructive and partially destructive methods to characterize waste packages and to quantify the activity inventory. The nondestructive systems range from gamma scanning and neutron counting to transmission and emission computerized techniques (TCT, ECT, DR).

Many waste packages produced in the early 1980's and before originate from non-qualified waste treatment processes. As a consequence they do not generally meet the acceptance criteria without quality control actions. On behalf of German nuclear regulatory authorities KFA Jlich is currently responsible for the realization of quality control measurements on 1200 conditioned waste packages stored at the interim storage facility of Gorleben.

For waste characterization and quality checking nondestructive testing methods as well as destructive analysis methods are applied. The routine inspection consists of a fixed set of basic measurements applied to all waste drums and a variable set of extended measurements applied to selected packages when more information is desired. After the completion of the inspection procedure a report is produced for each individual waste package that forms the basis for the documentation of all necessary waste characteristics. The documentation of these drums is then checked by the German 'technical inspection agency' (Technischer berwachungsverein, TV) which operates on behalf of the radiation protection board BfS (Bundesamt fr Strahlenschutz). If the acceptance criteria are fulfilled the waste packages may be disposed of in the final repository Morsleben which will be operating until 30.06.2000. The disposal of the waste packages has to be accepted by the radiation protection board (BfS). For waste packages which do not fulfill the requirements specific waste treatment processes will be suggested (conditioning, drying etc.).

On the average the time spend to examine and handle one waste package amounts to half a day. This means that the current work program (1200 drums of which 400 have been examined up to now) will last roughly two and a half years. Therefore one major objective is to set up and optimize an inspection program for routine application while maintaining a high quality in performing the methods.

WASTE FORMS AND CHARACTERIZATION

Site specific acceptance criteria have been derived from radiation protection regulations and from the safety assessment of a repository. The waste acceptance requirements distinguish between several types of waste with respect to their chemical and physical composition and different types of containers. Concerning solid waste (waste category A1) to be disposed of in the German final repository of Morsleben requirements on the waste forms and activity limitations for certain radionuclides and nuclide groups must be fulfilled. Activity limitations have been derived for more than 150 nuclides depending on different considerations like normal operation, assessment of incidents, criticality and long-term radiological impacts. The inventory of at least sixty-four of those nuclides must be given in the documentation.

Acceptance Criteria for Waste Category A1 - Solid Waste

Solid waste to be disposed of in the repository of Morsleben has to fulfill the following basic requirements (1):

- solid waste form,
- no free liquids, not even in bottles,
- no internal canisters,
- no toxic materials,
- no chemical and biochemical reactivity (no corrosion, rotting or fermentation),
- no flammable substances,
- normal atmospheric pressure inside the drum (no gas production).

Activity Limitations

The waste category A1 is subdivided into five radiation protection groups (S1 to S5). Solid waste belonging to the lowest radiation protection group (S1) must fulfill the following limitations:

- dose rate near the surface D < 2 mSv/h
- total activity of a-emitters A < 0.4 GBq/m3
- total activity of b/g-emitters A < 4 GBq/m3
- surface contamination a < 0,5 Bq/cm2 , b/g < 5 Bq/cm2

For waste contained in standard 200 l drums some activity limits are listed in Table I for radioactive nuclides that can be detected by nondestructive assay. The second column gives typical detection limits for a real waste drum on the basis of nondestructive gamma scanning (measurement time: 40 minutes; weight of the matrix: 400 kg; 10 MBq of Co-60 and Cs-137 activity in the matrix).

Table I

The activity inventory is usually derived from quality checking of the waste treatment processes, random sampling inspections and calculations (enrichment of the nuclear fuel, burnup history, material activation, cooling time etc.). In fact, many relevant isotopes cannot be measured directly by nondestructive means because of their low concentration in the waste.

Therefore the concept of 'detection of key-nuclides' has been established for quality checking which is applied for instance within closed operational procedures at nuclear power plants. From the measurement of certain key-nuclides (mainly Co-60 and Cs-137) the activities of all other relevant isotopes are calculated on the basis of 'predefined' correlation factors. The correlation factors are derived from detailed radiochemical analysis of the respective waste streams. These correlation factors have to be approved by nuclear regulatory authorities.

Classification of the Types of Waste Encountered in the Program

Within the current quality checking program all waste packages result from nuclear power plants. Sometimes they may also contain cross-contaminants from fuel fabrication or research activities. Thus the activity is mainly due to fission and activation product activity. Nevertheless, it is necessary to assure the absence of a-emitting and fissile nuclides above certain limits. The type of waste examined so far can be subdivided in the following categories:

- unconditioned waste (scrap etc.) with low to medium density and low activity,
- cemented waste (density ca. 2g/ccm) with medium to high activity content,
- compacted waste - so called 'pellets' - (ashes, iron scrap etc.) with medium to high density (up to 3g/ccm) and medium to high activity,
- packages with internal shielding structures (weight more than 600 kg) and high activity content (up to more than 1 GBq),
- filter cartridges in special shielding containers,
- special items: packages containing free liquids, elevated pressure or containing radioactive gases (Rn-222 or tritium release from corrosion processes).

EQUIPMENT AND PROCEDURES

The routine inspection procedure consists of a fixed set of basic measurements applied to all waste items and a variable set of extended measurements applied to selected packages where more information is desired.

Basic Measurement Program

The basic measurement program is intended to characterize the waste according to the desired specification items (weight, dose rate, surface contamination, pressure, gas production rate, gas composition, integrity of the drum etc., integral gamma spectrum).

The basic measurement program involves the following methods and procedures:

- visual inspection of the surface of the drum,
- g-dose rate measurements in contact and in 1 m distance,
- a- and b/g-surface contamination,
- weighing the drum,
- measurement of the pressure inside the drum,
- gas sampling and gas analysis
- integral gamma scanning with a collimated HPGe-detector and evaluation of the total inventory of all detectable radionuclides,
- segmented gamma-scanning to analyze the spatial distribution of the prominent g-emitters,
- visual inspection of the drum filling by opening the drum.

The following gas components are analyzed by:

- standard organic and inorganic components by mass-spectrometry/gas-chromatography (MS/GC), thermal conductivity detection (TCD):
  H2, O2, N2, CH4, CO2, CO, FCKWs, other organics
- radioactive gas components by radioanalytical GC:
  H-3, C-14, Kr-85, Rn-222.

After the completion of the basic inspections and evaluation of the data the technical inspection agency decides whether further investigations have to be conducted. On the average about 90 percent of all drums can be released for final disposal without additional measurements.

Extended Measurement Program

The extended measurement program serves two purposes. One objective is to obtain more information on the internal structure of the drum (internal shielding, density or activity distribution). The other objective is to take samples for destructive analysis. Very often the first step helps to identify 'regions of interest' within the drum where representative samples can be taken.

The extended analysis procedure may involve the following methods:

- taking samples for radiochemical analysis,
- core drilling of cemented wastes,
- radiochemical analysis of samples by a- and g-spectrometry and by LSC techniques,
- transmission and emission computerized tomography (TCT, ECT),
- digital radiography (DR),
- passive neutron counting,
- correction techniques for internal shielding.

DETERMINATION OF THE ACTIVITY INVENTORY

Activity of the Key-Nuclides

Measurement of the activity of the key-nuclides in the drum is achieved by the well established method of gamma-scanning (4). A collimated HPGe-detector is moved along the drum axis, while the drum itself rotates on a turn table (Fig. 1). The standard measurement time is 40 minutes for a scan. The collimator has a diameter of 40 mm.

Fig. 1

The activity inventory is calculated for all detected nuclides from the evaluation of the integral gamma-spectrum according to the following relation between the measured net peak count rate Z and the activity concentration a = A/m (3) which is valid for homogeneous fillings of the drum:

Eq. 1

The above relation can be extended to situations where a homogeneous activity concentration is shielded by an additional inner shielding structure. For all undetected relevant isotopes the activity will be calculated in correlation to the measured key-nuclides (mainly Co-60 and Cs-137) using the specific isotope vector of the waste under consideration.

Activity of a- and b-Emitting Nuclides

If the inventory of a- and b-emitting nuclides has to be measured samples are taken for radiochemical analysis. This yields the isotopic composition of the waste in correlation to the prominent g-emitters. The a/b-inventory of the drum is then calculated by multiplying the activity of a key-nuclide with the relative abundance of each a/b-nuclide.

Problems Associated with Standard Procedures

The main difficulties associated with the standard methods of activity determination are well known. Uncertainties are mainly due to the following effects or difficulties:

- the activity of activation and fission isotopes generates a background in the gamma-
- spectrum so that the LLD for other nuclides is deteriorated,
- non uniform activity distributions and inhomogeneous matrix compositions may cause deviations between real and measured total activities,
- external shieldings must be identified to account for g-absorption,
- samples taken for destructive analysis may not be representative for the complete drum (more than one pellet per package).

To account for inhomogeneities of activity or matrix distribution more advanced nondestructive testing methods like computer tomography and digital radiography are used in addition to gamma-scanning. These techniques generate tomographic pictures of selected slices of the object or projected side views (6). Thus the opportunity is given to evaluate in detail the distribution of relevant gamma-emitting nuclides which yields more reliable values for the activity within a drum. The analysis is especially useful for the quality checking of radioactive waste packages resulting from non-qualified conditioning processes or with insufficient documentation. Figure 2 shows a simple sketch of the top view of the detection system to perform TCT/ECT- and DR-measurements (5).

Fig. 2

The tomographic scanning system developed with financial support of the European Union in the Institute for Safety Research and Reactor Technology (ISR) at KFA Juelich combines two detection systems. Three HPGe-detectors can be used for gamma scanning and spectroscopy simultaneously. For transmission measurements in computed tomography (TCT) or digital radiography (DR) four fast plastic scintillators are used. The transmission source is Co-60 with an activity of 7 Ci. The time for generation of a tomographic slice amounts to typically 15 minutes whereas radiographs are produced in 1-2 hours depending on the desired quality of the data. The system set-up and testing has been finished this summer.

SELECTED RESULTS AND DISCUSSION

In the final section of this paper some specific 'problems' encountered during the work program will be discussed. The intention is to introduce and discuss some of the extended nondestructive testing methods that have been developed to examine waste packages which cannot be investigated satisfactorily with the standard procedures.

Generation of Nuclide Specific Activity Distributions

The first example is a drum containing several waste 'pellets' (compacted canisters filled with waste). Some of these 'pellets' contain ashes produced by incineration and/or non-combustible waste coming mainly from nuclear power plants. In rare cases small quantities of a-contaminated waste can also be found originating from nuclear fuel fabrication.

The identification of a-contaminants like U-235 or Am-241 can be achieved by g-assay of the emission lines at 185.6 keV or 59.5 keV. These energies may be strongly affected by absorption in the waste matrix especially in the case of higher densities (> 2 g/ccm). The total a-activity inventory is calculated on the basis of destructive sampling and radiochemical analysis of the isotopic vector. The reliability of the determination of the isotopic vector is strongly affected by the representativeness of the samples taken out of the drum. In order to obtain 'representative samples' it is important to quickly identify which pellets contain a-contaminants and whether the activity is distributed homogeneously.

As an example Fig. 3 shows the two dimensional distribution of the count rate of the 185.6 keV peak of U-235 measured at the drum's surface. It is clearly visible that a-contaminants are located only in the upper pellet (the drum contained two of them). The angular profile at the circumference suggests that the material is not distributed homogeneously within the pellet.

Fig. 3

Due to the detailed analysis of the distribution of the a-contaminants samples could be taken of the hot spot regions of the upper pellet where

an activity concentration of approximately 20 Bq of U-235 per gram of waste was detected.

Density Correction by Transmission Mmeasurements

The second item is an example of a drum with a mass of more than 800 kg. This weight implies the existence of a heavy shielding structure. Another result of the basic inspection was that the Co-60 and Cs-137 were concentrated in the middle part of the drum (Fig. 4b). Thus it was necessary to perform shielding corrections for both key-nuclides. Therefore a digital radiograph of the drum and additional transmission and emission tomograms at 500 mm height above the bottom of the drum were produced. The results are shown in the Fig. 4 a-d).

Fig. 4 a-d

The radiograph clearly shows the existence of a massive cylindrical shielding structure (probably cast iron) with an inactive layer of cement at the bottom. From the visual inspection only a cement layer at the top can be identified.

The effective thickness of the shielding structure as well as the average density of the region filled with active material can be determined from the tomographic data. The correction factors for absorption are then calculated by computerized modeling of the shielding structure on the basis of the measured values. For the above mentioned drum the detected shielding amounts to an absorption factor of ca. 120 +/- 15 for of Co-60 and ca. 660 +/- 80 for Cs-137.

CONCLUSIONS

The effort to characterize waste originating from non-qualified waste treatment processes can be very time consuming and expensive if the waste producer has to proof the fulfillment of the requirements for interim storage or final disposal. Thus emphasis is given to the development of a comprehensive quality control system for all waste treatment and conditioning processes. In Germany there are roughly 60000 to 80000 waste packages with insufficient quality evidence concerning up to date requirements which have been produced before ca. 1980. Therefore, nuclear regulatory and management authorities have to define and initiate quality inspection programs before these waste packages can be disposed of in a final repository. Such programs should be routinely applicable with high throughput and high quality.

The work described in this paper has shown that quality control inspections for waste drums originating from non-qualified processes can be performed on a routine basis with sufficient quality. The application of state-of-the-art nondestructive assay techniques can help to reduce the time effort and costs spend for such inspections by decreasing the amount of destructive measurements. Another advantage is the validation and improvement of accuracy of the activity determination which is based on other techniques like gamma scanning. Furthermore such extended assay techniques can help to minimize the radiation exposure for the personal staff involved.

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THE RETRIEVAL OF INTERMEDIATE LEVEL MAGNOX WASTE AT SELLAFIELD, UNITED KINGDOM A REGULATORY PERSPECTIVE

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ABSTRACT
The safety of nuclear installations in the United Kingdom is regulated by Her Majesty's Nuclear Installations Inspectorate (NII), on behalf of the Health and Safety Executive (HSE), using a largely non-prescriptive licensing system. This paper summarizes the roles of these organizations and the way in which they exercise regulatory control. It also touches on the inspection and assessment which NII carries out to satisfy itself that a licensee's safety case is adequate. In particular it considers the application of regulatory control to projects associated with the retrieval of intermediate level Magnox waste at Sellafield, which is operated by British Nuclear Fuels plc.
Drawing on two examples of current projects, the paper analyses some safety aspects related to waste retrieval. Of central importance is the need to ensure that waste forms are chemically and physically stable, and so suitable for safe long term storage.
NII has gained valuable experience through its regulation of these projects. This paper sets out the key regulatory lessons learnt in the hope that this experience will benefit delegates to this conference.
Note: The views in this paper are those of the authors and do not necessarily represent those of Her Majesty's Nuclear Installations Inspectorate. Introduction
1. Since the 1950s, the reprocessing of irradiated Magnox (magnesium alloy clad) fuel at Sellafield, operated by British Nuclear Fuels plc (BNFL), has given rise to a considerable volume of intermediate level waste. The waste of special interest to this paper is the Magnox cladding which, when removed prior to reprocessing of the irradiated fuel rod, is known as swarf.
2. Until recently the stripped cladding was stored underwater in silos. One of the drawbacks of this is that, under certain conditions, Magnox

swarf corrodes under water giving rise to an essentially insoluble residue, referred to as sludge. Magnox sludge and swarf can be physically and chemically unstable, and have the potential for giving rise to a significant radiological release should an incident cause a breach of containment. Therefore, the form of the waste is a matter of regulatory concern.

3. This paper outlines the work carried out by Her Majesty's Nuclear Installations Inspectorate (NII), part of the Nuclear Safety Division of the Health and Safety Executive (HSE), to regulate projects associated with the post operational clean out and decommissioning of intermediate level Magnox waste storage facilities at Sellafield. In particular it refers to projects currently in progress, or planned, to retrieve this waste.

4. The first part of the paper describes the UK's regulatory framework for the control of safety on nuclear licensed sites; briefly discusses the HSE's publication 'Tolerability of Risk' and how the philosophy described therein is applied through the 'Safety Assessment Principles for Nuclear Plants'; and relates this to the specific regulatory assessment of licensee's safety cases insofar as they relate to radioactive waste. The paper then summarizes the UK policy on radioactive waste management and decommissioning, and goes on to describe the regulation of work carried out by BNFL to retrieve Magnox waste. From this, the paper discusses the safety significance of waste retrieval and draws out several significant regulatory lessons learnt, each of which has generic application.

5. BNFL's paper to this conference (1) deals with waste retrieval from the operator's view point and contains more details of plants and processes at Sellafield.

UK NUCLEAR REGULATORY SYSTEM

6. All civil nuclear installations in the UK are subject to the Health and Safety at Work etc. Act 1974 (2). Amongst the relevant statutory provisions of this Act are those parts of the Nuclear Installations Act 1965 (3) which refer to issues of safety. Under the Nuclear Installations Act, no site may be used for the purposes of installing or operating any prescribed nuclear installation unless a nuclear site license has been granted by the HSE and is, for the time being, in force. The HSE has delegated responsibility for administering this licensing function to NII.

The Licensing Regime

7. The licensing regime is established by NII through powers under the Nuclear Installations Act to attach conditions to the site license which are enforceable in a court of law. In addition there are the powers available under the Health and Safety at Work etc. Act, for example to issue enforcement notices. This regulatory regime has been successfully applied to a wide variety of nuclear installations within UK over many years and has been shown to provide a powerful yet flexible system of control capable of being matched to the degree of hazard involved. The licensing regime covers a nuclear installation through its full life cycle from design to decommissioning and takes into account the need to regulate and control the management of radioactive waste.

8. The site license is predominantly non-prescriptive and most conditions attached to it require the licensee, who has the sole responsibility for ensuring safety, to make and implement adequate arrangements to take

account of specified requirements. An example of this for Accumulation of Radioactive Waste (License Condition 32) is:

The licensee shall make and implement adequate arrangements for minimizing so far as is reasonably practicable the rate of production and total quantity of radioactive waste accumulated on the site at any time and for recording the waste so accumulated.

This Condition also gives the HSE the power to formally approve the arrangements, to specify any limitations to the arrangements or, if it wishes, to specify the place and manner of waste accumulation.

9. A similar Condition gives the HSE the powers to direct the licensee to dispose of the accumulated or stored waste in accordance with an authorization, issued by the appropriate Authorizing Department, under the relevant legislation. Other Conditions relating to the control of nuclear matter on the site include monitoring, record keeping, radiological protection, training, documentation, emergency arrangements, quality assurance and, in particular, the requirement for the licensee to produce a safety case to justify safety during all stages of the installation's life. Safety aspects of the movement of radioactive matter within, onto and from the site and the keeping of radioactive material on site are also regulated by NII under the provisions of the Ionizing Radiations Regulations 1985, made under the Health and Safety at Work etc. Act.

The Regulators

10. The aim of the NII is to secure the maintenance and improvement of standards of safety at civil nuclear installations and to protect workers and members of the public from ionizing radiation. To achieve this aim the NII is structured to enable priorities to be decided centrally; to respond rapidly to technical innovation and operational experience; to be consistent in the development and application of standards; and to apply the regulatory requirements consistently and coherently across the whole nuclear industry.

11. The NII is divided into inspection, assessment and policy branches. The inspection branches, as their name would suggest, are primarily involved in carrying out site inspection activities to confirm that licensees are complying with their legal obligations. The assessment branches provide specialist technical advice on the adequacy of the licensee's safety cases. The policy branches co-ordinate the setting of NII policy on a range of issues, making sure that it accords with that set by other government departments and agencies.

12. A licensee's safety case must include reference to radioactive waste management and accumulation facilities on the site. In addition the licensee has to obtain authorization for discharges from the site. These discharges are regulated by authorizations issued under the Radioactive Substances Act 1993 by the Authorizing Departments (Department of the Environment - principally HM Inspectorate of Pollution - the Welsh Office and the Ministry of Agriculture, Fisheries and Food or the Scottish Office, as appropriate). In order to ensure consistency within the UK national policy on radioactive waste, and to ensure a minimum of duplication of work between government bodies, a memorandum of understanding clarifies the responsibilities of the various government departments and agencies involved.

Tolerability of Risk

13. In response to a recommendation by Sir Frank Layfield in his 1986 report of the Sizewell Inquiry into the UK's first commercial pressurized

water reactor HSE produced 'Tolerability of Risk' which, after consultation and revision, was republished in 1992 (4). The document is a straightforward account of risk written for the general public. It discusses how people normally approach risk, shows how industrial risks (and nuclear risks in particular) are regulated, the nature of risk from radiation and how risks are calculated.

14. Three levels of risk are identified:

a. a risk which is so great or the outcome so unacceptable that it must be refused altogether - the 'unacceptable' or 'intolerable' risk. These cannot be justified except in extraordinary circumstances;

b. a risk which is or has been made so small that no further precaution is necessary - the 'broadly acceptable' region where no detailed working is needed to show that risks are as low as reasonably practicable;

c. risks that fall between these two states, that have been reduced to the lowest level reasonably practicable bearing in mind the benefits flowing from its acceptance and taking into account the cost of further risk reduction. The Health and Safety at Work etc. Act requires that such risks must be reduced so far as is reasonably practicable - the 'Tolerable' region.

The document goes on to quantify these regions which, in turn, form the basis for the quantitative guidance to NII assessors developed in NII's Safety Assessment Principles.

Safety Assessment Principles

15. The NII's Safety Assessment Principles (5) are used as guidance by NII assessors in their examination of licensees' safety cases for all licensed installations including nuclear chemical plant. The Principles are not 'standards' imposed on licensees but nonetheless they have been published so that anyone can be aware of the criteria against which safety cases are judged. They are intended primarily for use with new plant and major modifications, but are also used when reviewing older plant in comparison with modern standards.

16. The majority of the Principles reflect good engineering practice which can be regarded as the basis of safe design. However, they also contain overall risk targets (derived directly from Tolerability of Risk) which NII uses to assess whether the licensee's probabilistic analysis demonstrates that the risks from its plant are balanced and not dominated by a single failure or fault sequence. Tolerability for risks associated with normal operation and accident conditions are separately addressed, each having defined levels of tolerability denoted by a Basic Safety Limit, and a broadly acceptable level by a Basic Safety Objective.

17. In the UK's non-prescriptive regime the licensee is free to propose any means to achieve an appropriate level of safety but must demonstrate in a clear and unambiguous fashion that the proposals are adequate. This is achieved through the safety case which demonstrates, in an objective and traceable manner, the safety adequacy of the proposed operation from the design through commissioning and operation to the end of plant life and decommissioning. In particular it must link the underlying safety concepts, supporting research and development to the design criteria. From this the safety case identifies the operational limits and constraints for safe operation of the plant throughout its lifetime. It will be appreciated that not all this information is available from the very start and so the safety case is usually developed in a phased fashion.

18. The involvement of the regulator does not remove the licensees' responsibility for safety. NII's assessment of a safety case is carried out using its experience and expertise on a sampling basis to establish confidence in the arguments put forward and to determine whether the licensee has, as a minimum, met their own criteria. The assessment is carried out in a probing manner, using the Principles as the framework, to test the licensee's claims and assumptions - taking into account the safety significance of the chosen aspects. Dialogue between the assessors and the licensee is maintained until NII is satisfied with the safety arguments.

19. In pressing for the licensee to meet acceptable standards the test of 'reasonable practicability' is always an argument open to the licensee to limit their obligations. As a result much depends on the magnitude of the potential hazard. For example, the qualitative engineering Principles are more likely to be liberally interpreted where the potential hazard is low. The underlying rationale for the NII's assessor is to ensure that the licensee has avoided, or adequately contained, all identifiable potential hazards within the bounds of reasonable practicability.

UK POLICY ON RADIOACTIVE WASTE MANAGEMENT AND DECOMMISSIONING

20. The UK national policy on radioactive waste management was reviewed in 1994/95 and the conclusions of that review were set out in 'Review of Radioactive Waste Management Policy' (6). From this, NII has derived policy statements on radioactive waste management and decommissioning to ensure that UK policy is properly implemented, taking due notice of best international practice. In summary, the policy statements cover amongst other things the need for the licensee to:

- undertake strategic and site/plant specific planning of radioactive waste management and decommissioning;
- prepare programs and arrangements for decommissioning, and commence decommissioning at a time acceptable to NII;
- demonstrate a systematic and progressive reduction of hazards presented by the site or plant;
- take a balanced account of radiological risks to workers, the public and the environment;
- demonstrate that wastes are not unnecessarily created, and that generation and accumulation has been reduced as far as reasonably practicable;
- segregate and characterize radioactive wastes, where practical and cost effective;
- store wastes in a passive and retrievable form and, where appropriate facilities are available, dispose of the wastes as soon as reasonably practicable;
- have a valid safety case for waste storage facilities, including maintenance and surveillance programs and periodic safety reviews.

The two examples of waste retrieval projects which follow show how NII applies UK policy in practice.

RETRIEVAL OF INTERMEDIATE LEVEL MAGNOX WASTE

21. Historically, NII were concerned that certain facilities for the storage of potentially high hazard waste were being operated for longer than first envisaged. Over a period of time NII encouraged BNFL to develop a strategy for safe long term storage; and BNFL themselves realized the need to take action, for both safety and commercial reasons. An acceptable overall strategy is being developed, and considerable progress has been made. Central to the strategy is that the waste should

be retrieved, and then immobilized to render it safe for long term storage and disposal. The remainder of this paper concentrates on the first step of waste retrieval. To illustrate some of the generic nuclear safety issues and regulatory lessons learnt, two examples of current waste retrieval projects have been chosen:

Example 1 - Sludge Retrieval from Magnox Pond and Decanning Plant

22. The Magnox pond and decanning plant was commissioned in 1959/60 for the purpose of receiving and decanning Magnox fuel, and for exporting irradiated fuel rods for reprocessing on site. It operated until 1986, when its function was taken over by a new building. Spent Magnox fuel rods were placed in skips in a pond for a period of cooling, to allow shorter lived isotopes to decay to acceptable levels, and to provide a shielded environment.

23. BNFL opted for storage of Magnox clad fuel under water to minimize the likelihood of fire if Magnox and irradiated uranium were exposed to air. However, Magnox cladding corrodes in untreated water to form an essentially insoluble residue mostly made up of magnesium hydroxide, commonly referred to as 'sludge', and this corrosion reaction evolves hydrogen. Overtime the amount of sludge in the Magnox pond and decanning plant built up so that by 1986 the plant had accumulated around 1800 cubic meters of sludge. The levels of radiation in the building and of airborne activity started to rise, and the potential existed for personnel to exceed statutory dose limits unless special preventive measures were applied.

24. BNFL have since embarked on a series of plant improvements (particularly the addition of new ventilation systems) and post operational clean out projects, with the objective of removing all fuel, sludge and debris and so reduce local levels of radiation, to enable further decommissioning to proceed safely. Although this work is scheduled to continue well into the next century, conditions within the plant have already improved significantly.

25. NII has reviewed the overall program of post operational clean out activities and has decided to carry out a detailed examination of a sample of projects, of which the most significant is the retrieval of sludge from one of the five original wet bays. BNFL has designed the bay sludge retrieval facility to transfer this sludge to settling tanks as custom-built interim storage. Facilities are now being planned to enable treatment and immobilization of this sludge at a future date.

26. The facility has been developed from commercially available units, though with the addition of specialist control equipment. The real area of difficulty in this project lies in the requirement to install and commission the facility safely in an area where the levels of radiation and contamination are high and where access is restricted. This called for a thorough study of possible options and methods of work to ensure that dose uptake is as low as reasonably practicable. These studies required a knowledge of the working environment, and a precise breakdown of specific operations. The culmination of this work is a series of BNFL documents which lay down the steps to be taken and identifying those responsible for specific actions.

27. The facility was tested extensively, using a scale model and a full size off-site rig. Off-site commissioning enabled operating and maintenance personnel to gain a first hand experience of the facility in totally inactive conditions, whilst at the same time proving the equipment and demonstrating the correct operation of safety systems. The

use of off-site commissioning to eliminate dose uptake during inactive commissioning has been a prominent feature of the Sellafield waste retrieval projects, one which NII wholeheartedly supports.

28. Active commissioning of the facility is underway and quantities of sludge have been transferred to settling tanks. A review of early commissioning results was carried out to gain a better understanding of the physical and chemical processes involved during sludge retrieval, and so to make sure that the levels of conservatism were sufficiently high given the uncertainties inherent in work of this type. The review confirmed that commissioning could proceed on a careful, step by step, approach; and that at each step commissioning results need to be compared to expectations and any deviations properly justified.

Example 2 - Swarf Retrieval from Solid Active Waste Storage Facility

29. Swarf from the decanning process has been stored underwater in concrete compartments at the Solid Active Waste Storage Facility from 1964 until recently, when the Magnox Encapsulation Plant started receiving swarf directly from the fuel handling plant for encapsulation. The risk of a fire due to the generation of hydrogen as a result of swarf corrosion is a dominant feature of the safety case for the building. The corrosion reaction is exothermic, and so cooling and inerting capabilities have been fitted.

30. The other significant aspect is the potential for leakage of water from the compartments, and indeed a major leak occurred in 1976. NII carried out an investigation of this leak, and published their findings (7). The incident helped clarify the position with respect to interim storage: further storage capacity was limited, and the strategy of retrieval, immobilization and safe long term storage adopted.

31. It was agreed that BNFL would first retrieve the more recently deposited and essentially intact swarf, because it represents the greatest risk of hydrogen generation. No equipment existed at the time to fulfil this objective, and BNFL commissioned the swarf retrieval facility, a plant novel to the company. In essence, the design principles of this facility were that it should be able to work within the same physical envelope as those existing machines which service the compartments, be capable of retrieving swarf and keep it under water for export using a standard flask, and to effect a seal so that inerting of a compartment could be carried out in the event of detection of significant levels of hydrogen.

32. The basis of the facility is a simple petal grab which is lowered into a compartment, collects a quantity of swarf, and then deposits the swarf into a water-filled bin which is raised into a flask and transported for later encapsulation. It was designed in a modular form to accommodate existing crane capacity. Its substantial weight (around 170 tonnes) represents a significant additional load for the building to support and necessitated verification that the building could support the facility. The seismic capability of the building also had to be re-examined to make sure that the facility did not affect it significantly.

33. The facility was fabricated, and subjected to a thorough inactive commissioning program, before being brought on-site. This was done by BNFL to iron out any operational problems without attracting any dose detriment, and also for training operators and maintenance personnel. 34. BNFL has drawn up a strategy to empty the building, and waste retrieval has started. At present, recovery of swarf from one compartment is complete, and retrieval from another compartment is underway. In the near

future, we expect BNFL to present to NII its plans for post operational clean out of the remaining eighteen compartments in the building, which will be developed from experiences gained to date.

REGULATORY LESSONS LEARNED

35. Several issues have emerged which are common to other waste retrieval projects, and their identification will aid the assessment of future post operational clean out and decommissioning activities:

i) Waste Retrieval and Immobilization. The regulator's aim is to ensure that the licensee renders any potential hazardous waste safe at source. However, Magnox sludge and swarf has existed for some years mainly in a form which presents a potential threat due to its physical and chemical characteristics. Clearly the top priority for dealing with such waste is to retrieve and immobilize the waste as soon as reasonably practicable, and so render it suitable for safe long term storage or disposal. BNFL has recognized the importance of this work, and retrieval of the waste is well underway.

ii) Understanding the processes. The removal of sludge from the Magnox pond and decanning plant exemplified the need for operator and regulator to fully understand the processes involved in waste retrieval. The uncertainties inherent in work of this type call for high levels of conservatisms which can only be refined through experience gained from predictable and consistent commissioning results. The regulator should look for the licensee's commissioning plan to proceed on a careful, step by step, approach; and at each step commissioning results should be compared to expectations and any deviations properly justified.

iii) The Role of the Regulator. Once the NII had accepted BNFL's strategy for the treatment and safe long term storage of intermediate level waste, BNFL displayed firm commitment to hazard reduction by cleaning out the plants and encapsulating the waste on a reasonable timescale. It can take time for the regulator and licensee to reach agreement about the long-term strategy for such activities, but it is an essential first step in that it provides the necessary framework. Thereafter, provided the licensee maintains adequate progress and demonstrates adequate control of the safety of its work, regulation is relatively straightforward.

iv) Regulatory Standards. For the two waste retrieval projects described earlier, the buildings did not represent satisfactory long term storage arrangements, and there was a need for action to prevent further deterioration. In such situations, there may not be sufficient time available to allow the licensee to develop ideal improvements - it may be better to adopt a solution which is fit for purpose and is available quickly. Hence there is a need for a balanced approach to regulation, wherein the regulator should apply appropriate standards, although modern standards remain as a benchmark against which to base these regulatory judgements.

v) Risks associated with modifications. The regulator should ensure that the licensee has demonstrated that there is an overall net benefit to be gained by modifying plant to effect waste retrieval. But the regulator should also bear in mind that a short term increase in risk may be necessary to secure real long term improvements to safety.

vi) Safety during Installation and Commissioning. Regulation of waste retrieval projects has reinforced the need to consider the risks to the work force associated with performing a modification in a hazardous environment. The regulator needs to ensure that the operator has carried

out a systematic study of all potential hazards (nuclear and non-nuclear) and that they have in place effective managerial controls.

vii) Design Safety Case. The plant buildings described above were mostly constructed around 30 to 40 years ago. They were not designed to cope with the additional loads associated with waste retrieval, and in general they were not specifically designed to resist earthquakes. In such situations, the regulator should ensure that structural capability has been assessed retrospectively, and that the integrity of the structures is sufficient to sustain waste retrieval over a given period of time.

viii) Off-Site Commissioning. Waste retrieval equipment at Sellafield often has had to be installed in areas of complex layout and with high background radiation, making working conditions difficult. This has necessitated remote handling, extensive use of off-site facilities for testing and training, and special attention to the need to keep doses as low as reasonably practicable, which called for a thorough identification of safe working practices and effective administrative controls. Off-site commissioning has come to be regarded as an essential component of the safety case of a waste retrieval project. Therefore the regulator has to be involved in the off-site activities of the licensee.

ix) Novel Plant. The use of plant novel to the nuclear industry to effect waste retrieval is often unavoidable, in which case the licensee must pay more attention to design proving studies, model testing and full scale mock-ups. It also means that the regulator should pay special attention to the assessment of the adequacy of the plant and its ability to fulfil its function without undue risk.

CONCLUSIONS

36. This paper has outlined some of the ways in which the NII has used and continues to use its regulatory influence to improve the safety of projects associated with the retrieval of Magnox waste. Progress has been possible through the co-operation of BNFL, who have responded positively and responsibly in the course of dialogue with NII inspectors and assessors to the regulatory requirements of the NII.

37. The NII has gained valuable experience through its regulation of waste retrieval projects at Sellafield, experience which we hope will be of interest to delegates at this conference.

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ANNEX 1

UK Classification of Radioactive Wastes

The reader should note that Magnox sludge and swarf are classed as Intermediate Level Waste, in accordance with the classification system set out by the UK's Radioactive Waste Management Advisory Committee, Ref 8.
The categories adopted by the Committee are :
High-level or Heat Generating Wastes
Wastes in which the temperature may rise significantly as a result of their radioactivity, so that this factor has to be taken into account in designing storage or disposal facilities.
Intermediate-Level Wastes
Wastes with radioactivity exceeding the boundaries for low-level waste, but which do not require the generation of heat to be taken into account in the design of storage or disposal facilities.
Low-level Wastes
Wastes containing radioactive materials other than those acceptable for dustbin disposal, but not exceeding 4 GBq/te alpha or 12 GBq/te beta/gamma.
Very low-level Wastes
Wastes which can be safely disposed of with ordinary refuse (dust-bin disposal), each 0.1m3 of material containing less that 400 kBq of beta/gamma or single items less than 40 kBq of beta/gamma.

24-4
LOW-LEVEL TRANSURANIC WASTE ASSAY SYSTEM USING SEQUENTIAL PHOTON INTERROGATION AND ON-LINE NEUTRON COUNTING SIGNATURES

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ABSTRACT
A comprehensive program is currently in progress at several laboratories for the development of sensitive, practical, non destructive assay techniques for the quantification of low- level transuranics (TRU) in bulk solid wastes.
This paper describes the method being developed to assay high density TRU waste packages using photon interrogation. The system uses a pulsed electron beam from a linear accelerator (LINAC) to produce high-energy photon bursts from a metallic converter. The photons induce fissions in TRU waste package which is inside an original Neutron Separating and Counting Cavity (NS2C). When fission is induced in trace amounts of TRU contaminants in waste material, it provides "signatures" from fission products that can be used to assay the material before disposal. We give here the results from counting photofission-induced delayed neutrons from 239Pu, 235U and 238U in sample matrices. We counted delayed neutrons

emitted after each pulse of the LINAC by using the Sequential Photon Interrogation and Neutron Counting Signatures -SPHINCS- technique which had been developed in the present framework. The SPHINCS method enhances the available counts by a factor about 20 compared with the counting of delayed neutrons only after the irradiation period. Furthermore, the use of SPHINCS measurement technique coupled with NS2C facility improves the signal to noise ratio by a factor about 30. This decreases the detection limit.The electron linear accelerator operates at 15 MeV, 140 mA, and 2.5 ms wide pulse at a 50. and 6.25 Hz rate. The dynamics of photofission and delayed neutron production, NS2C advantages and performances, use of an electron linear accelerator as a particle source, experimental and electronics details, and a future experimental works are discussed.

INTRODUCTION

Radioactive waste management is an issue in every country with a civil and/or military nuclear program. Thus, assaying waste contaminated by alpha emitters with long half-lifea is vital for both safety and long-term waste management. The wide variety of materials and contaminants, the low concentrations and large volumes involved mean that such type of assay is a complicated matter. In France, actinide concentrations in surface-stored waste packages are limited to a maximum 100 nCi[a]/g . The average value on storage sites must be below 10 nCi[a]/g.

Over the last few years, considerable progress has been made in the field of assay techniques for low-level a contaminated waste. References (1) and (2) contain a wide survey of these developments.

This document presents an active detection method for radioactive waste embedded in high-density matrices, mainly concrete packages. The high density of the packages, as well as their high water content (up to 20%), means that only high-energy neutrons or gamma particles have an enough high range to activate the enclosed actinides. Our aims were to evaluate the feasibility of dosing transuranians by induced photofission, and to optimize an experimental system with a view to improving detection limits.

The experimental process described below (SPHINCS) coupled with the NS2C irradiating cavity is one of the first to use a LINAC assay method combined with sequential detection using delayed neutrons.

PRINCIPLE OF THE METHOD

The induced photofission assay method consists of irradiating radioactive matter with high-energy photon bursts in order to produce (,fission) reactions in the enclosed transuranics. Measurement is based on detecting prompt and delayed photofission neutrons to establish the quantity of "photofissile"b material present. Two measuring methods can be used:

i) irradiation at low levels of photon energy followed by counting the prompt photofission signal,

ii) high energy irradiation followed by counting of delayed neutrons.

The first method has the advantage of being statistically satisfactory. The number of prompts neutrons emitted in fission is approximately 100 times greater than the number of delayed neutrons emitted after fission. But the (,n) reactions on different materials other than TRUs produce signal interference that increases proportionally to the energy of the incident photons. Consequently, we have to use relatively weak levels of gamma energy. Since the photofission cross-section increases in proportion to the energy of the photons, this reduces the number of useful reactions. It should also be noted that the interrogation photons may "blind" the measurement system for a period of about one millisecond

(the "gamma flash") (3). This limits the counting of the prompt signal, which population decreases within a few hundred microseconds.
The chief drawback of the second method, i.e. counting the delayed signal, is the small proportion of delayed neutrons. This results in increasing measurement uncertainly. However, the method has the advantage of being unaffected by photoneutron interference and the "gamma flash". Thus using high energy photons increases the photofission reaction rate without affecting the background noise during counting.
Our investigations led us to opt for a pulsed irradiation and counting of the delayed signal between the LINAC pulses which is called Sequential Photon Interrogation and Neutron Counting Signatures ; the SPHINCS technique. This method enables us to increase sensitivity at least ten times.

PHOTOFISSION AND DELAYED NEUTRONS PRODUCTION

The production of delayed neutrons resulting from the decrease in radioactivity of fission fragments is directly linked to the photofission rate T. This is given by:
Eq. 1
The number of delayed neutrons emitted by group i during time tc, after a pulse is:
Eq. 2
Since the detection follows each pulse, the total number of delayed neutrons for all the groups (i=1,...6) emitted during the n counting periods will therefore be:
Eq. 3
If counting is only carried out after n pulses, the total number of delayed neutrons is:
Eq. 4
With:
Eq. 5
Counting the delayed neutrons signal between the LINAC pulses (SPHINCS) increases significantly the sensitivity.
As Eq. 1 shows, the photofission rate is in proportion to the integral on the energy of the product of the effective cross-section s[g,F](E) and the Bremsstrahlung spectrum S(E). This integral increases as a function of the electrons energy (see Fig. 1).
Fig. 1
The choice of this energy is limited by the thresholdc of the reaction
Reaction
equal to 15.9 MeV. It is also limited by the rate of photoneutrons that increases with the energy. These photoneutrons can interfere with the delayed signal when counting between pulses.

EXPERIMENTS

Photon Source

Our series of experiments was carried out using the linear accelerator (LINAC) of the DGA/ETCA/DPN at Arcueil. this accelerator has the following characteristics: pulse width 2.5 ms, repetition frequency ranging from 6.25 Hz to 400 Hz, electron energy ranging from 15 MeV to 30 MeV and a peak current intensity of 140 mA. The tungsten braking target is 3 mm thick. When the electron beam strikes the target, photons are emitted following the deceleration of the electrons in the nuclei coulomb field. This is the Bremsstrahlung radiation phenomena.

Experimental Setup

Figure 2 is a diagram of the experimental setup used. The distance between the target and the element to be irradiated is approximately 120 cm. Neutron counting is carried out using proportional counters with a useful length of 100. cm and a diameter of 2.55 cm. The counters are filled with at a pressure of 4 atmospheres. These counters are covered in cadmium-coated polyethylene and are placed approximately 40 cm away from the beam axis. Neutron detectors around the outside of the waste package are then used to detect neutrons from the photofissions. Fission neutrons passing through the cadmium barrier (the cadmium energy cut off is approximately 0.417 eV) are thermalized in the polyethylene before being detected by the 3He counters. We use the end point energy (electron energy) of 15 MeV. The duration of irradiation pulse is 2.5 ms.
Fig. 2
Electronics
Four 3He counters are arranged as in Fig. 2. The counters are linked by 40 m cable to a signal amplifier operating in current collection mode (drawer type TADS manufactured by the French Eurisys Mesures company). Pulses transmitted by this setup are used by a multiscale board operating in multisweep mode Counting synchronization versus time is performed by the start pulse of the LINAC.
Detection efficiency is measured for a single counter using an Am-Li source (without any matrix) with an energy spectrum close to that of the delayed neutrons and located approximately 40 cm from the detector unit.
Insert A
Due the large size of the accelerator hall, the photoneutrons produced instantaneously on the components of the embedding matrix or the braking target, persist as thermal neutrons long after the gamma flash. Their half-life is of the order of 4.80 ms. Hence the need for the cadmium shielding. Figure 3 shows a recording versus time of neutrons (g,n) made by a counter unit with and without cadmium shielding.
Fig. 3
NS2C IRRADIATION CAVITY
As shown in Fig. 3, the surrounding cadmium of the detection system absorbs the majority of the photoneutron background noise. However, the experimental results obtained with and without cadmium cover are not in agreement with our predictions. This could be explained either:
- by "leakage" of the cadmium shielding,
- or by a non-thermal neutronic component which still active during the counting time.

The first reason is, to our knowledge, not significant due to the high quality of the cadmium shielding manufacturing.
A contrary to, the second possibility appears more probable. This could be explained by the "huge" size of the experimental cell (13m x 13m x 3m) which leads to a greater thermalization time for he photoneutrons (about 20 ms). Consequently, during the useful counting time (12 ms after photon burst) a significant amount of background noise coming from this origin still exists.
In order to reduce the life time of such neutrons, we build a specific cell which surrounded the sample and it matrix (see Fig. 4). This new cell, called NS2C (Neutron Separating and Counting Cavity) allows us to cut both external photoneutrons (coming from the bremsstrahlung target and cell walls) and internal photoneutrons (coming from photon interactions on matrix and on NS2C walls). Moreover, the NS2C leads us to

detect, with low background noise, the delayed neutrons from photofission reactions (useful signal).
Experimental results using the NS2C are very satisfactory: The corresponding signal to noise ratio is about 200. compared to the one obtained without such new cell which is about 30. (see Table I)
Table I
Fig. 4
EXPERIMENTAL RESULTS
Samples of UPu (weight 3.83 g) , 238U (weight 100g, 194g and 292g) and 235U (weight 1.90 g) are used (see Table II). They are placed in blocks of polyethylene (diameter 10 cm and height 20 cm), glass (diameter 10 cm and height 9 cm) and concrete (diameter 56 cm and height 110 cm) with respective densities of 0.95, 1.70 and 2.35 g/cm3. The background radiation level is obtained by irradiating the blocks devoid of all TRU matter. The foreground is obtained by subtracting the background from the global signal ("photofissile" matter + package).
Table II
Figure 5 shows a global recording for 100 g 238U sample without matrix - inside the NS2C casemate- and the corresponding recording without the NS2C casemate.
Fig. 5
Neutron counts versus time for 100 g 238U inside the NS2C casemate and the corresponding recording without the NS2C casemate
DETECTION LIMITS
The samples shown above are contained in three different matrices (except for the 238U).
Measurements are taken using two counting units.
The number of delayed neutrons emitted by the 1980-pulse cycle for the UPu pellet and the 235U sample placed at the centre of the matrix are 2.36 106 and 7.23 105 neutrons respectively.
The detection limits are calculated taking the false alarm and non-detection risks to be equal to 5%. Background and useful signal count times are approximately 294 s. We shall consider the least favorable situation where all the fissile material is concentrated at the centre of the package.
Table III gives the detection limits in grams for an "irradiation-counting" time of 300 s and electron energy of 15 MeV. The irradiation occurs on the TRU sample which is inside the NS2C cavity.
Table III
Extrapolating for Entire Concrete Container
The diameter of the photon beam at 1 m from the Bremsstrahlung target is taken to be around 10 cm. The photon-irradiated section has a volume of 4.4 liters. The least favorable situation is considered where the fissile material is situated along the longitudinal axis of the container. Measurement is taken by axial scanning of the package. It consists of 10 reading each at a height of 10 cm during 100 s (the height of the package being around 1 m). Detection limits for each section are used to approximate the one of the whole package. The respective detection limits in counts CmD for the whole package CmD1 and for a section are related by:
Eq. 6
Where:
t = counting time for the whole package
t1 = counting time for a single section.

The mass detection limits are related by:
Eq. 7
Where:
Esp1 = Recording in counts per gram of TRU in the case of the section
Esp = Recording in counts per gram of TRU in the case of the package.
For our hypothesis this gives:
Eq. 8
We therefore obtain:
Eq. 9
For the hypothesis above (i.e. TRU matter along the container axis and signal and background measurement time of 300s) the detection limits in grams of actinides per gram of matrix for the UPu and the 235U samples are 8.94 10-7 g/g and 1.77 10-6 g/g. These values correspond to 3.24 10-6 Ci[a]/t and 4.10-2 Ci[a]/t for 235U and UPu respectively.

CONCLUSION AND FUTURE PHOTOFISSION EXPERIMENTS

Several important conclusions can be drawn from these studies. First, LINAC operation in the bremsstrahlung mode can produce sufficient photons in each burst to assay TRU with high sensitivity. Second, the use of pulsed irradiation and counting between pulses (SPHINCS technique) enhances the available counts by a non negligible factor. Furthermore, using the NS2C cavity improves the signal to noise ratio considerably. Following this way, the systems used here offer important advantages in waste management applications. First of all, they are compatible with passive counting systems. They can use the same or similar detection systems and electronic processing, since most passive systems assaying large containers will also utilize neutron detection. Secondly, they have energy variability and control of beam current. Both can be important in shorting assay time. Thirdly, they can assay large packages of waste barrels. or large crates. This is due to the high intensity available from the interrogating beam and to the potential for sweeping the beam across the package. Fourthly, the beam sweeping can be used to scan containers to locate contaminated portions of the volume that can sometimes be removed for recycling or to reduce the overall volume to below 10 nCi/g. Fifthly, the photon interrogation assay, will minimize the effects of the waste matrix on the assay.

In the future, we will concentrate on three areas of importance for application of the present technique to waste assay. First, we will determine the yields of photoneutrons for various matrices and the ratio of prompt to delayed events when TRU references are included. From these determinations, we can set the limits of practicability in counting prompt fission signatures. Second, we will build and evaluate a prototype assay system scaled up to large sample volumes and based on our experimental setup. With higher efficiency, better neutron gamma discrimination and a uniform mixture of matrix and TRU in a sample, we would expect to decrease the detection limit.

Finally, we will study the background problems that occur when radioactive isotopes that emit beta and gamma radiation are introduced into the sample matrices. For 3He detector systems, photon-induced pulses are lower in amplitude, but they can "pile up" to signal amplitudes above the counting threshold and result in false counts. Shielding materials and thickness ones and detector operating conditions are the important areas to explore.

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RADIOACTIVE WASTE MANAGEMENT IN SLOVENIA

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ABSTRACT
The problem of radioactive waste management is both scientifically and technically complex and also deeply emotional issue. In the last twenty years the first two aspects have been mostly resolved up to the point of safe implementation. In the Republic of Slovenia, certain fundamentalist approaches in politics and the use of radioactive waste problem as a political marketing tool, brought the final radioactive repository siting effort to a stop. Although small amounts of radioactive waste are produced in research institutes, hospitals and industry, major source of radioactive waste in Slovenia is the Nuclear Power Plant Krko. When Krko NPP was originally built, plans were made to construct a permanent radioactive waste disposal facility. This facility was supposed to be available to receive waste from the plant long before the on site storage facility was full. However, the permanent disposal facility is not yet available, and it became necessary to retain the wastes produced at the plant in the on-site storage facility for an extended period of time. Temporary radioactive storage capacity at the plant site has limited capacity and having no other options available NPP Krko is undertaking major efforts to reduce waste volume generated to allow normal operation. This article describes the Radioactive Waste Compaction Campaign performed from November, 1994 through November, 1995 at Krko NPP, to enhance the efficiency and safety of storage of radioactive waste. The campaign involved the retrieval, segmented gamma-spectrum measurement, dose rate measurement, compaction, re-packaging, and systematic storage of radioactive wastes which had been stored in the NPP radioactive waste storage building since plant commissioning.

INTRODUCTION

The Krko Nuclear Power Plant is a 2 loop Westinghouse-designed PWR nuclear electric generating station located by the Savariver, just outside the town of Krko in the Republic of Slovenia. Krko NPP, since startup in 1981, has supplied electric power to the republics of Slovenia and Croatia providing 15 to 20 % of total electricity generation. As a normal byproduct of the generation of electric power by a nuclear power plant, wastes from plant processes are generated. Some of these wastes contain radioactive materials resulting from the normal operations of a nuclear steam supply system, and must be stored in a safe and efficient manner in order to protect the public and plant staff from exposure to radioactive materials.

The Krko project was originally supposed to be the beginning of a very ambitious Yugoslav nuclear program. At that time, during the seventies, the radioactive waste management policy was also of general Yugoslav interest, concern and responsibility. When the construction of nuclear facilities was banned in 1987, the radioactive waste policy became a problem of republics of Slovenia and Croatia because these two republics owned the only nuclear power plant. According to the agreement between the governments of Croatia and Slovenia, a Project team, responsible for preliminary activities and preparation of all necessary documents and licences for the construction of final repository for low and intermediate activity waste, was established within the organization of NPP Krko. During six years of Project team existence a conceptual design for shallow ground and tunnel type repository was prepared, together with preliminary safety assessments for both types of repository - to name just a few among more than ninety documents produced. Preliminary screening was performed, based on available geological, seismic, hydrogeological and other relevant data defining suitable candidate macro locations for the final repository according to recommendations of the International Atomic Energy Agency combined with methods successfully implemented elsewhere. Preliminary results have indicated that suitable locations for final repository are available in both Republics.

In January 1991, the Slovenian Republic Administration for Nuclear Safety issued "Guidelines for the Low and Intermediate Level Radwaste Repository Site Selection in Slovenia". According to the Guidelines, the site selection procedure is executed in four steps taking into consideration forty three criteria. In February 1993 the second step results were reviewed and public announcement was made for thirty-six potential locations covering the total area of 896 hectares situated in eastern and north-eastern parts of Slovenia. It should be noted that the public reaction after the announcement was much milder as compared to the reaction after the presentation of the first step results in June 1990. Despite considerable effort the Project team was never publicly accepted. Unfortunately, radioactive waste management was understood in Slovenia as an exclusive problem of the nuclear power plant, instead of being discussed and resolved within the scope of waste management at the state level. For this reason in 1993 the Slovenian government founded the Radioactive Waste Management Agency, reporting directly to the government. As a result of the third step of the selection process, five most suitable locations were identified and presented to the public provoking strong disapproval within the local communities where the locations have been identified. At that point the siting process was stopped and it is not expected to resume any activities in the near future. When the level of ecological consciousness of Slovenian people

becomes high enough to understand the obligation to dispose of waste, including the radioactive waste, in a controlled manner, the process will continue with detailed investigations of candidate sites. The direct prerequisite for such a development is the clear standpoint of the Slovenian Government, Assembly and ecologists, indicating the firm intention to build the repository as a part of changing the attitude towards surroundings. Temporary radioactive storage capacity at the plant site has limited capacity and having no other options available, NPP Krko is undertaking major efforts to reduce waste to allow normal operation. Among many other activities, supercompaction of existing waste was one of viable options.

KRKO RADIOACTIVE WASTE SUPERCOMPACTION CAMPAIGN

Waste Types and Quantities

The types of wastes stored at NPP Krko include the concentrate from evaporation of plant process waste liquids (EB); used process filter cartridges (F); used protective clothing, gloves, and rags-compressible wastes (CW); depleted ion exchanger resins used in water purification systems (SR) and previously supercompacted waste (SC). Continuous operation since the commissioning of the plant had, by November 1995, resulted in the production of 9924 55-gallon and 617 85-gallon storage drums containing radioactive wastes.

Table I

Krko Radioactive Waste Storage Facility

The Krko Radioactive Waste Storage Facility is a seismically qualified 1470 m2 building located within the protected area of Krko NPP (Fig. 1). Construction is of steel-reinforced concrete, with 1 meter thick outer walls and 60 centimeter thick inner dividing walls. The roof slab is 1 meter thick except for the slabs above the entrance compartment, which are 60 centimeters thick. The facility is built on a reinforced concrete base pad, and is designed to withstand predicted potential earthquakes and extreme weather conditions. The waste storage areas interior to the building consist of six corridors 4.17 meters wide and in excess of 7 meters high. A second level of storage area is provided, supported on steel beams which are in turn supported on steel plates bolted to the dividing walls spaced at 0.5 meter intervals. The facility is continuously monitored for airborne radioactivity, and contamination surveys are regularly performed in accordance with the Krko site radiological control program. Prior to performance of the compaction campaign, the existing 55-gallon waste drums were stored, in two tiers 5 layers deep, horizontally on the concrete floor and on steel shelving. The steel shelving was supported by steel plates bolted to interior walls. Access for inspection and monitoring within the storage matrix was difficult.

Modifications to the Storage Facility

The storage facility was modified to accept the new TTC (overpack container) and an overhead crane added to transport them. The saddles used in the old storage configuration to support horizontally placed drums were removed since the new containers specially designed for the supercompaction campaign are stored in a vertical position. For the second level of waste container storage, the steel support structure required only minor modifications to accommodate the additional weight of the compacted wastes. Steel grating was installed on the steel support beams, and horizontal steel restraint structures were added in each compartment to provide additional stability for the new containers. Steel

ribs were added to the support beams to prevent sliding of the grating during any seismic event.

Fig.1

Extended Storage Requirements

Construction of a new radioactive waste storage facility, in addition to the existing one, was considered, but the licensing process for construction of a new building would take considerable time, and the new facility would thus not be available before the current facility was completely full. Thus, NEK decided to explore techniques for more efficiently utilizing the available space in the existing storage facility.

In addition, the drums in use were not suitable for long-term storage in a temporary facility, and were also not suitable for off-site transportation or for permanent disposal. Thus, it was decided to repackage the existing filled drums into superior, specially designed containers or "overpacks". These containers had to be suitable for extended temporary storage, transportation to the final waste repository, and to facilitate handling for permanent disposal.

By 1995, the Radioactive Waste contained in the storage building occupied approximately 94% of the available storage space. With the plant designed to operate until the year 2023, the remaining storage space would be inadequate. However, it was determined that by supercompacting and repackaging the 8600 drums containing solidified evaporator concentrated compressible waste, it would be possible to gain enough space for 3000 additional drums.

Waste Volume Reduction

In 1988 and 1989, a quantity of 1924 standard 55 gallon storage drums containing compressible wastes were supercompacted using a large mobile "supercompactor" owned by Westinghouse Electric Corporation.. The compressed drums, referred to as "pucks" were placed into 617 overpacks (85 gallon steel drums) and returned to storage in the storage building. Additionally, tests were performed in order to determine the efficiency of the supercompactor when used to compact drums containing evaporator concentrates solidified in vermiculite cement. Due to the porous nature of the vermiculite cement, and the high compaction power of the mobile supercompactor, the tests showed that these drums could be reduced in volume by up to 50%. Thus, it was decided by NEK management to use waste compaction technology to address the lack of storage space and to meet the requirements of volume reduction.

Tube Type Containers (TTCs)

Use of the 85-gallon overpacks, which were used for storage of the compacted waste drums resulting from the 1988-89 compaction tests, was determined to be impractical. These overpacks are not qualified to contain weight greater than 500 kg. The Tube Type Containers (TTCs) were thus specially designed for use in the Krko radioactive waste facility, and were produced on the basis of NEK specifications. The TTCs are qualified as IP-2 transportation package per IAEA Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Wastes" (IAEA, 1985, amended 1990). The TTCs are made of steel, 2700 mm high, with an internal diameter 640 mm, wall thickness of 2 mm, bottom and top closure cap thickness 2.5 mm, and total weight, including waste, of 2500 kg. The height of the TTCs was chosen to optimize their placement in the storage building, and to contain from 4 to 10 pucks depending on the type of waste and the volume reduction achieved or 3 standard 55 gallon waste

drums if repackaging is needed. Calculations addressing different types of overpacks showed that the use of these specially designed overpacks would dramatically reduce the volume of the stored wastes following the supercompaction campaign.

The TTCs are qualified for use for extended temporary storage, and for later transportation to a final disposal site. As part of the qualification procedure, they were subjected to a severe regime of stacking and drop tests. Each TTC is coated, interior and exterior, with qualified primer and paint. These coatings ensure that moisture from outside cannot enter the container, and that any trace corrosive material (boric acid) will not corrode and degrade the container walls. In addition, these coatings provide a high degree of fire resistance.

Before the start of the TTC filling operation, a measured quantity of desiccant material was added to the bottom of the TTC. The same desiccant material was also used to fill the voids during the filling process and to cap off the remaining empty space on the top of the TTC. This ensures that the pucks inside the TTC's are securely immobilized, while at the same time absorbing any residual moisture. After a TTC has been filled with pucks of compacted waste, the closure lid is welded to the body of the TTC. Welding of the closure lid of filled TTCs was performed on a specially designed base table, which turns on interior ball-bearings. The welder stands behind lead shielding to minimize the radiological dose received while welding the closure lid, and the TTC turns in place, allowing the welding to be performed in minimum time and with minimum manipulation by personnel.

SUPER COMPACTION CAMPAIGN EQUIPMENT

The Westinghouse/Scientific Ecology Group (SEG) Mobile Supercompactor used for the campaign is a 1000 ton hydraulically operated press contained in one standard 12 m truck trailer, along with its equipment: a hydraulic power unit, a waste drum loading system, an air filtration system, a liquid collection system, and a compacted drum unloading crane.

To facilitate handling of the overpacks used for the compaction campaign, a 4 Ton Electric Overhead Traveling Crane was provided. This crane conforms to FEM 1.001 (3rd Edition 1987) rules for hoisting appliances. Crane rails to support the equipment were installed permanently in the 6 storage cell ceilings. The crane itself is constructed and installed so that it can be removed from one storage cell, and then installed in another. This provides the capability to quickly recover any overpack or drum which may be damaged, and to facilitate repackaging and replacement within the new storage configuration.

In order to facilitate the loading of the overpacks and prevent dropping the overpack or contents, a loading platform was built for the campaign. The platform was designed to accommodate six (6) overpacks, thereby allowing the platform operator to select an overpack based on partial fill and compacted drum size. The platform was equipped with a hoist capable of picking up a compacted drum at the compactor exit conveyor and lift them to the top of the platform, there to be lowered into the selected TTC. Sufficient working space is available to permit the lowering of uncompacted drums into the overpacks should that be desirable.

A conveyor system specially designed for use with the platform was also installed. This included a loading conveyor where the drums to be compacted were deposited by the forklift, to be fed by gravity to the compactor loading arm. A special interlock built into the system allowed

only one drum at a time to roll into the loading arm, allowing the next drum to take its loading position once the loading arm was lowered. At the exit end of the compactor, a gravity type conveyor system was installed. This conveyor, completely enclosed to prevent the spread of airborne contamination, ran from the compactor unloading chute to just below the overpack loading platform where the compacted drums were picked up by the platform lifting hoist. A system of drip pans was also provided to prevent spills of liquids onto the compacting area floor when drums containing liquids were ejected from the compactor.

COMPACTION CAMPAIGN ACTIVITIES

In order to ensure proper identification of the contents of the existing drums prior to compaction, drums were inspected, weighed, and scanned using a segmented gamma spectrometer. The segmented gamma spectrometer was chosen for use in the supercompaction campaign in order to provide state-of-the-art measurement technology, capable of categorizing the wastes much more accurately than the standard external dose rate measurements employed at most commercial nuclear stations. The gamma spectrometer was used on all but a few drums during the campaign, and scanning of each drum took approximately fifteen (15) minutes. The remaining drums were above the weight limit for the scanning equipment, and so were measured using the standard practice of external dose measurement.

Using the segmented gamma spectrometer, the radiological (isotopic) content of each drum and dose rate measurements were taken, and the data from each drum recorded. These measurements provide a detailed record of the contents of each TTC, and the data necessary to calculate, in advance, the dose rate which will result when several pucks are placed into a TTC. Once the weight, activity and dose measurements had been determined and recorded, drums were sorted by these criteria for compaction. Drums identified as containing spent ion exchanger resins or filters were separated and not compacted. The scanning of all waste drums handled during the campaign was performed by NEK personnel in accordance with the NEK Radiological Protection Procedures.

The first radioactive waste storage drum was compacted, with representatives of the Slovenian Safety Authorities present, on November 3, 1994. The compaction operation was interrupted due to a hydraulic system failure in the supercompactor, which was returned to the Westinghouse European Service Center in Belgium for repairs. Repair work was completed on March 21, and the unit was immediately returned to Krko. Operations resumed on March 27, 1995. At the end of the campaign, a total of 8770 drums had been compacted. These include 7135 drums containing evaporator concentrates, and 1635 Compressible Wastes drums. In addition, 387 drums containing filters and spent ion exchanger resins were repackaged into TTC overpacks without being compacted, and 766 of these were relocated to shielded storage positions without being repackaged. A total of 1745 TTC's were used for the campaign. The final volume reduction achieved was 27% of the original volume. Refer to Fig. 2 for an illustration of volume reduction achieved for different waste types. Figure 3 illustrates the type and number of drums compacted, repackaged, and stored during the performance of the campaign.

Fig. 2

Fig. 3

Radiological Controls

Health Physics coverage was performed by the NEK Health Physics department. Monitoring, dose assessment, and radiological controls were performed in accordance with Krko radiological control procedures. An ALARA study was performed prior to the start of the campaign, which provided estimates of doses expected for each member of the campaign staff. Due to extra measures taken by Krko Health Physics and the campaign staff the actual total dose accumulated during the campaign (238.81 mManSv) was substantially lower than originally estimated dose (514.31 mManSv).

Each member of the campaign crew was provided with presentations and instruction prior to the start of the project. Each member's responsibilities and duties were explained, along with requirements of site procedures for radiological control and personnel monitoring. By systematically storing the drums within the storage matrix, dose rates resulting from handling were reduced, and the remaining resin/filter drums could be stored separately in cell D where they are shielded by filled TTC's. TTCs containing waste with higher dose rates were placed interior to the storage matrix, so that high activity TTCs are surrounded by the lower activity TTCs which provide additional shielding. This resulted in a considerably reduced dose rate at the boundaries of the storage matrix in keeping with ALARA principles. The dose rate in the access areas of the radioactive waste storage building varies from zero to 50 (Sv per hour). In the accessible areas in front of the storage cells, the highest dose rate in any area accessible by personnel is 150 (Sv per hour).

Records Keeping and Traceability

Project records were kept on printed paper and in a computerized data base. These records show the exact number of drums compacted, their type, weight, dose rate, final TTC location, and the TTC location in the storage building. Records were kept during the performance of the campaign by the QC/Record Keeper, who recorded each drum location in the TTCs, the drum and TTC numbers, and ensured documented TTC desiccant fill and lid welding verification. The location of each closed TTC within the storage building was recorded, as were dose rate measurements on contact, at 1 meter, and at 2 meters.

Gamma spectrometer data were used to calculate predicted dose rates from filled TTCs. When the fill was complete, predicted and actual dose rates were compared to ensure accuracy. Logs were kept of the activities on each shift.

CONCLUSION

The Radioactive Waste Compaction Campaign was performed from November, 1994 through November, 1995 at Krko NPP, to enhance the efficiency and safety of storage of radioactive waste. The campaign involved the retrieval, segmented gamma-spectrum measurement, dose rate measurement, compaction, re-packaging, and systematic storage of radioactive waste which had been stored in the NPP radioactive waste storage building since plant commissioning. The final volume reduction achieved was 30% of the original volume. Additional storage space was provided for at least five years of normal plant operation.

In addition to the reduction in volume, the compaction campaign brought additional benefits. By placing the supercompacted drums into new type container with thicker walls, superior stability is achieved, which diminishes the probability of handling accidents during storage, transportation and final disposal activities. During the supercompaction

campaign a more precise estimate of types and quantities of radionuclides contained in wastes was achieved using a direct segmented gamma-spectrum measurement, and by a combination of external dose and gamma-spectrum measurements.

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24-6

DEVELOPMENT AND IMPLEMENTATION OF A SITE RADIATION PROTECTION PROGRAM FOR A RADIOACTIVE WASTE VITRIFICATION AND RCRA CLEAN CLOSURE PROJECT AT THE SAVANNAH RIVER SITE

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ABSTRACT

A systematic implementation of a radiological protection program at the GTS Duratek (GTSD) Vendor Treatment Facility (VTF) at the Savannah River Site. The project is unique in that it incorporates a turnkey approach to operation and control of a waste treatment facility by a single subcontractor at a DOE site. It should be of specific interest to those DOE M&O and ERMC prime contractors who subcontract waste treatment, radiological control, and industrial safety technologies and services.

INTRODUCTION

The purpose of this project was to implement a radiological protection program at the GTS Duratek (GTSD) Vendor Treatment Facility (VTF) at the Savannah River Site (SRS). The VTF is a temporary installation in the M-Area of SRS consisting of buildings 341-1M and 341-8M and tanks 100-7, 100-8, and 100-10. The objective of the VTF is to convert approximately 660,000 gallons of mixed (radioactive/hazardous) waste in sludge form to a durable, stable glass wasteform. The scope of the VTF project also includes RCRA clean closure of the tanks following removal of the sludge. Facility-specific radiological controls for the VTF are necessary to minimize safety and health risks to occupational workers, as well as members of the general public. The Radiation Protection Program (RPP) establishes radiological requirements for all VTF operations and support activities.

DESCRIPTION OF THE M-AREA VENDOR TREATMENT FACILITY PROCESS

Mixed wastes (hazardous and radioactive) were originally stored in nine tanks of the Interim Treatment/Storage Facility and in the Mixed Waste Storage Shed by the Reactor Materials Department (M Area) of SRS. The tank waste has been transferred to tanks 100-7 and 100-8 for mixing prior to vitrification. The wastes are aqueous based sludges from a wastewater treatment facility supporting an electroplating operation containing nickel and uranium, which classify them as hazardous and radioactive respectively. The wastes partially occupy three 500,000 gallon tanks, six 35,000 gallon tanks, and one-hundred twenty-five 55 gallon drums. GTSD

has coupled a single stage treatment unit to these tanks and, with the addition of common glass forming chemicals, convert these wastes into a smaller volume of chemically durable glass, with a total waste volume reduction of 83.7%. At the conclusion of the vitrification process, the storage tanks will be decontaminated to RCRA clean closure standards. A conceptual process flow diagram of this operation is given in Fig. 1.
Fig. 1
Operation
The DuraMelter 5000 will be operational 7 d wk-1, 24 h d-1, 52 wk y-1 producing glass at a nominal rate of 5 tons d-1, with a maximum production rate of 7.5 tons d-1. The system will operate continuously until all of the waste has been processed. At this rate, it will take less than one year to vitrify all 660,000 gallons of waste.
Vitrification Melter
The waste feed slurry mixture is pumped to a single stage vitrification unit (DuraMelterTM 5000). Here, the slurry is deposited on the surface of a molten glass bath. Heat from the glass bath is transferred vertically through the accumulated feed pile and evaporates free water from the slurry, calcines inorganic salts to metal oxides, and fuses the resultant mixture into a uniform melt which mixes with the molten glass inventory. Any organic species in the feed are oxidized to carbon dioxide and water.
Vitrification Off-Gas Treatment
The melter is coupled to a multistage off-gas treatment system, which maintains the melter at a constant slightly negative pressure and treats emissions. The reactions that occur when the slurry enters the melter generate steam, a variety of gases (NOx, SOx, CO2, and trace halides), and particulate. The front end of the off-gas system is composed of a water spray quencher followed by two aqueous based packed bed towers connected in series. Following the packed bed scrubbers is a mist eliminator and a dry filtration process which assures the absence of particulate radionuclides (uranium) in the final process exhaust. These filtration units are baghouses and high efficiency particulate air (HEPA) filters. A sampling port is provided after the HEPA filters for air monitoring.
Vitrified Glass Waste Form
The waste glass produced is a borosilicate glass designed to meet TCLP release rate requirements while maintaining a high waste loading (large waste volume reduction). The glass will be transformed into flat, marble-shaped objects called "gems," and loaded into 71 gallon steel drums. The 71 gallon drum has a square cross section which allows for a higher drum packing density.
Radionuclide Inventory
The Curie balance for all systems is schematically shown in Fig. 1. The radionuclide emission from the off-gas system, post-HEPA, is calculated to be 1.67 x 10-9 Curies d-1.
RPP DEVELOPMENT
The contracting organization for the VTF, Westinghouse Savannah River Company (WSRC), recognized that the successful operation of the facility was dependent upon a dedicated workforce of not only operations personnel, but a health and safety staff, as well. The ability of the contractor to provide qualified radiological control personnel to the project at significantly less cost to WSRC was an added benefit of a contracted health and safety staff. The WSRC Site Technical Representative (STR) for the VTF communicated these concepts well

throughout the technical and contractual decision-making processes at WSRC, resulting in a dedicated VTF Radiological Control Organization.
The RPP was developed in accordance with the requirements of three primary source documents: WSRC Manual 5Q, Radiological Control; the DOE's Radiological Control Manual, DOE/EH-0256T; and WSRC-SCD-4, Operational Readiness Functional Area Requirements for Radiation Protection; as well as contractual commitments between GTSD and WSRC. In late 1995, the RPP was reviewed and revised to incorporate requirements of 10 CFR 835, Occupational Radiation Protection. Through the use of a requirement matrix, a Project Instruction was developed which defined the RPP. It was then necessary to develop specific Implementing Procedures to meet the requirements of the developed RPP. A total of forty-two (42) procedures were developed under the following groupings: Organization and Administration, Internal Audits and Investigations, Radiological Protection Procedures and Posting, External Radiation Control Program, External Radiation Dosimetry, Internal Radiation Exposure Control Program, Internal Radiation Dosimetry, Fixed and Portable Instrumentation, Air Monitoring, Radiation Monitoring/Contamination Control, ALARA Program, Records, and Training.
Development of the RPP Project Instruction and Implementing Procedures was initiated in January, 1995. After addressing minor comments from WSRC, the completed program met all requirements of the source documents and contract. The program was accepted in full by WSRC in early May, with the concurrence of the SRS DOE site representative and the GTSD Quality Assurance organization. Program development time, including the review process, was approximately three months. The review and revision process for 10 CFR 835 compliance was completed in approximately three weeks in late 1995.

RADIOLOGICAL CONTROL ORGANIZATION AND RESPONSIBILITIES

One of the challenges that GTSD faced in providing its own dedicated radiological control staff was to maintain the important delineation of the operations and health and safety organizations. A Radiological Control Organization (RCO), consisting of a Radiation Protection Manager, Radiation/Safety Supervisor, and five (5) Radiological Control Technicians was formed to establish and conduct radiological control operations at the VTF. No member of the RCO holds an operational position in the VTF organization, although RCO personnel are often called upon to "pitch in" on tasks such as decontamination, housekeeping, and general labor. Each member of the RCO is empowered to stop work or mitigate the effect of an activity if he/she suspects that the activity will result in the violation of VTF RPP implementing procedures or result in imminent danger or unacceptable risk. The relationship of the RCO to the overall VTF organization is shown in Fig. 2.
Fig. 2
The Radiation Protection Manager (RPM) is the corporate manager with overall responsibility for the successful development and implementation of the VTF RPP. In this capacity, the RPM provides program direction and assistance to the Radiation/Safety Supervisor (RSS). The VTF RPM is empowered with final authority for resolving radiological issues and concerns. For the purposes of the VTF project, he reports functionally to the VTF Program Director.
The Radiation/Safety Supervisor (RSS) is the individual primarily responsible for the daily management of the radiation protection, occupational health & safety, and industrial hygiene programs for the

VTF. In that capacity, the shift Radiological Control Technicians report to him and he in turn reports to the Director of Operations. The RSS receives technical direction and assistance in the implementation of the RPP from the RPM. The primary duties and responsibilities of the RSS are to:

Work closely with the Facility Operations Manager to ensure that radiological control support is available for VTF operations and maintenance, as appropriate.

Implement methods to ensure that personnel exposure, contamination of areas, equipment, and personnel, and solid radioactive waste volume are minimized.

Review radiological surveys, sample results, logs, deficiency reports, and incident reports as specified in VTF RPP implementing procedures.

Ensure that the appropriate radiological data are forwarded to the WSRC STR for assessment of RPP effectiveness.

Act in the capacity of VTF RPM, as specified in VTF RPP implementing procedures.

The Radiological Control Technician (RCT) is responsible for routine and non-routine monitoring of radiological conditions within the VTF, and for assisting and guiding VTF personnel in the radiological aspects of their jobs. VTF RCTs report directly to the RSS, and may receive direction in operational aspects of their jobs from the Shift Supervisor or the Facility Operations Manager. The primary duties and responsibilities of the RCT are to:

Perform routine and non-routine monitoring of radiological conditions within the VTF, including airborne radioactivity, contamination levels, and external radiation fields.

Operate and perform calibration checks on all radiological monitoring and detection equipment in use at the VTF.

Install, verify, and maintain radiological postings at the VTF.

Install, verify, and maintain radiological engineering controls at the VTF.

Monitor access and egress from radiological areas at the VTF.

Generate radiological documents (such as Radiation Work Permits and Radiological Incident Reports) and maintain radiological records.

Assist and guide VTF personnel in the radiological aspects of their jobs.

In addition to RCO personnel, all on-site VTF personnel are qualified as Radiological Workers. Each radiological worker must understand that proper radiological control is an integral part of their daily duties. Radiological workers are trained to recognize that their actions directly affect contamination control, personnel radiation exposure and the overall radiological environment associated with their work.

RPP IMPLEMENTATION

The RPP development and implementation schedule, relative to the operational schedule of the VTF, is shown in Fig. 3.

Fig. 3

The first step in implementing the RPP was the selection of the RSS and one RCT to set up radiological control facilities and equipment during the construction phase. Two additional RCTs were added at the beginning of the sludge transfer phase. A fourth RCT was added at the mid-point of sludge transfer, and the full complement of one RSS and five RCTs was present on site prior to startup. Prior to startup, the onsite RCO typically worked five eight-to-ten hour dayshifts per week. From startup

through operation, the RSS typically works straight dayshifts, not including off-shift tours and inspections, with one RCT assigned to each of five rotating operational shifts. For the tank closure phase, RCO staff will be reduced to the RSS and two RCTs. During the sludge transfer phase, and prior to assignment to rotating shifts, each of three RCTs was assigned to a dedicated duty area: sludge transfer operations, tank modifications, and count room. Defined work scope and RCT familiarity associated with each particular duty area maximized RCO efficiency in supporting VTF operations. To minimize front-end qualification time, RSS and RCT candidates were selected from applicants with current training qualifications, including DOE RCT Core Academics, SRS Site Radworker II, SRS Respiratory Protection qualification, and OSHA 40-hour HAZWOPER. The willingness and ability of the RSS to perform RCT duties on an as-needed basis further added to the cost and time savings realized on the VTF project.

Since the uranium mixture described in Fig. 1 is the sole radiological component of hazards present at the VTF, the bulk of radiological control activities address contamination and airborne radioactivity, rather than external radiation fields. Due to the low enrichment in U235 of the uranium mixture, criticality control was determined to be unnecessary. Industrial hazard monitoring and control, though not addressed in this text, required a substantial manpower commitment from the RCO. All industrial hazard monitoring and control at the VTF, including that for heat stress, noise, chemical hazards, combustible gas, oxygen levels, and confined space entry is conducted by the RCO under the guidance of a Certified Industrial Hygienist. Consolidation of radiological control and industrial safety duties significantly reduced the cost of the VTF health and safety programs.

The RCO facility at the VTF consists of a dedicated mobile office trailer with desk space for three, records files, a single computer and printer, and the radiological count room. The count room is equipped with a stationary phoswich detector and scaler for alpha/beta/gamma counting of airborne and surface contamination samples, a GM detector and rate meter for personnel monitoring, storage space for all field monitoring instruments, and a set of check sources for all radiological instrumentation. Field monitoring instruments include additional "friskers" for personnel monitoring, portable alpha, beta, and gamma contamination and dose rate meters, and stationary and portable air samplers. Instrumentation is calibrated at an offsite calibration facility, also owned and operated by GTSD. Since instrument needs "ramp up" toward the end of sludge transfer and into startup, Instruments are added to inventory on a "just-in-time" basis. Instruments are also provided by the GTSD Instrument Services group, so no downtime is experienced in using the just-in-time approach. Posting materials, protective clothing, and respiratory protection equipment at the VTF is provided by WSRC.

Currently (post-construction/start-up testing), there are several radiologically posted areas at the VTF. The 341-1M Building is posted as a Radiological Buffer Area (RBA) with the area around tanks 100-3, 4, and 5 posted as Contamination Areas (CA). All open tanks which have not been decontaminated internally (100-7,8) are posted CA and Airborne Radioactivity Area (ARA). All remaining tanks have been decontaminated and deposted. Maximum loose surface contamination inside any tank is approximately 10,000 d min-1 100 cm-2 beta-gamma, and 1000 d min-1 100

cm-2 alpha. Airborne radioactivity in any tank has not exceeded 2 DAC total, and respiratory protection is routinely used inside the tanks. Tank 100-10 RCRA clean closure is completed. The radiological count room is currently posted RBA. No area at the VTF currently requires Radiation Area (RA) posting, and the need for this posting is not anticipated for the duration of the project. The maximum external exposure rate recorded at the VTF has been 0.7 mR hr-1 at contact in a sump used for sample storage in Building 341-1M.

RESULTS

To date (11/30/95) one Radiological Incident has been recorded at the facility. The incident consisted of a contaminated system opening for valve maintenance, without appropriate RCO notification. The incident did not result in any release of radioactive contamination or personnel contamination. The incident was promptly discovered by RCO personnel, with appropriate notification of the RSS and WSRC, and follow up actions in accordance with the RPP. No personnel contaminations have yet occurred at the facility. External radiation doses by thermoluminescent dosimeter have not been detected (self-reading dosimeters are not required since no RAs exist at the facility). No bioassay trigger levels have been reached, so no internal radiation dose has been recorded.

CONCLUSIONS

The following items are noted as strengths of this project which resulted in cost and time savings:

Dedicated health and safety staff provided to the project at significantly less cost to WSRC.

Support of the dedicated health and safety staff concept by the WSRC Site Technical Representative (STR).

Abbreviated RPP and Implementing Procedure development time.

Consolidation of radiological control and industrial safety duties.

RCO personnel participating in tasks such as decontamination, housekeeping, and general labor.

Willingness and ability of the RSS to perform RCT duties on an as-needed basis.

Ramp-up and ramp-down of personnel and material resources commensurate with project evolutions.

Defined work scope and RCT familiarity associated with dedicated, assigned duty areas.

RSS and RCT candidates selected from applicants with current training and qualification.

Just-in-time instrument supply and calibration resources.

Continuous assessment of the adequacy of allocated RCO personnel and material resources to VTF operations must be conducted to ensure that the VTF RPP and RCO continues to support the mission of the VTF in a cost and time-efficient manner.

Session 25 -- FIFTY YEARS OF NUCLEAR WASTE MANAGEMENT POLICY AND PRACTICE II

Co-chairs: Ted Gado, Foster Wheeler Environmental Corp.
Jack McElroy, Geosafe

25-1

HIGH-LEVEL WASTE MANAGEMENT: A PERSPECTIVE EXTENDED TO THE 1996 STATUS

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ABSTRACT

A 1974 symposium on High-Level Radioactive Waste Management is revisited. The early year plans for waste management are briefly reviewed. A selected chronology traces progress on programs for commercial, defense, and foreign High-Level Waste Management from that symposium to date. The emergence of a glass waste form produced in vitrification plant as an industry norm is apparent.

INTRODUCTION

This paper revisits the symposium on High-Level Radioactive Waste Management organized by this author in 1974. The symposium was sponsored by the American Chemical Society (ACS) Division of Nuclear Chemistry and Technology at the 167th national ACS meeting. The symposium papers were collected and published in the Advances in Chemistry Series No. 153 (1). Early on, it was recognized in the US Atomic Energy Commission that radioactive waste management covered a broad area of federal government and commercial activities in the nuclear fuel cycle. Two simultaneous sources for radioactive waste were perceived; 1) national defense plans that required plutonium generated radioactive waste of several unique types, and 2) commercial use of nuclear reactors to produce electric power generated quite another set of radioactive waste forms. One common product of both courses was (and is) high-level radioactive waste. However, even within a single generation source, the waste takes multiple forms and has no conveniently simple definition.

By 1974 high-level radioactive waste management had become a topic of discussion in the scientific and engineering communities, a subject for the nation's newspapers to interpret for the lay public, and a major issue for activist groups. The purpose for this symposium was to present a 1974 overview of high-level waste management activities. The invited papers fell into three categories: 1) Commercial High-Level Waste; 2) Defense High-Level Waste; 3) Environmental and Decontamination.

COMMERCIAL HIGH-LEVEL WASTE

In his overview of this category, Dr. Pitman, then director of the Division of Waste Management and Transportation, US Atomic Energy Commission made the following points about solidified high-level commercial waste (2).

> "The AEC's program will provide retrievable storage for these solidified wastes in carefully maintained and monitored engineered facilities for the next several decades."
>
> "During this period investigation and demonstration of waste disposal in deep, stable, geologic formation will continue."
>
> "After completion of an extensive testing and demonstration program establishing the acceptability of permanent disposal, the radioactive waste will be removed from the retrievable surface storage facility and disposed of permanently."

High-level waste management at the first, and only, commercial fuel reprocessing facility in the U.S. (Nuclear Fuel Services at West Valley, N.Y.) was summarized (3). Purex processing type high-level and intermediate level wastes were neutralized and stored in multi-confined mild steel tanks.

An Oak Ridge National Laboratory study (4) projected commercial waste generation through year 2000 indicating that it would, "portend a problem of impressive size and complexity, but one that could be handled within the framework of current and planned investigative programs." Natural salt formations were believed to offer the best prospects for high-level

waste disposal, although other promising geologic formations were being considered.

Solidified waste forms for the immobilization of high-level radioactive wastes from the commercial reprocessing of power reactor fuels and processes for the reliable production of the waste forms were being developed at Battelle Pacific Northwest Laboratories (5). The development program had begun on nonradioactive laboratory and pilot scales and was planned to be carried out through fully radioactive engineering scale demonstrations of the processes. Development emphasized silicate glass or glass-ceramic forms produced in a two step calcination-melting process.

Radioactive waste management in Europe was summarized (6). At that time, Germany, England and France were vigorously developing technology and methodology for incorporating high-level radioactive waste into silicate glass. Germany was concentrating on a spray calcination vitrification system. England had selected a rising level glass process in which evaporation, calcination, and borosilicate glass vitrification all took place in a heated pot. France was operating a small-scale batch pot calcination - batch vitrification system and developing a new continuous system using a rotary calciner and melter with a batchwise draw-off.

DEFENSE HIGH-LEVEL WASTE

Reprocessing defense fuels to recover plutonium values resulted in site specific waste forms closely related to the chemical processes employed. Consequently, high-level waste management strategy varied somewhat from site to site.

Methods were being considered to immobilize Savannah River Plant (SRP) wastes (7) in solid forms such a cement, asphalt, and glass. 137Cs and 90Sr were recognized as the major biological hazards and heat producers in the alkaline wastes prepared at SRP. In the conceptual processes studied, 137Cs removed from the alkaline supernates together with sludges that contained 90Sr were to be incorporated into solid waste forms of high integrity and low volume suitable for storage in a retrievable surface storage facility for about 100 years and for eventual shipment to an off-site repository. Mineralization of 137Cs or its fixation on a zeolite prior to incorporations into solid forms was being studied. Methods for removal of sludge from the waste tanks were under study.

A status report was given on high-level waste management at the irradiated fuel reprocessing plant (ICPP) at the National Reactor Testing Station (INEL) (8). Interim storage of an acidic high-level liquid waste in stainless steel tanks was followed by calcination of the liquid waste in a fluid bed calciner and storage of the calcine in bins. Conversion of the granular calcine to cermet, glass-ceramic, and ceramic forms was being carried out to test for alternative long-term storage requirements.

The Hanford Waste Management Program was to complete evaporation of liquid tank wastes to prepare a solid radioactive salt cake to be stored in mild steel tanks for the interim (9). Additionally, megacuries of 137Cs and 90Sr previously removed from the liquid tank wastes were converted into solid 137CsCl and 90SrF to be stored under water in doubly encapsulated metallic containers. Several alternative modes for long-term storage/disposal of these high-level liquid wastes were being evaluated. For some of these modes, conversion of the solids to immobile silicates of low water solubility would be desirable. Low (10) and high temperature process development for preparation of silicate minerals was in process.

ENVIRONMENTAL AND DECONTAMINATION

Although it was not an exhaustive treatment, concern for radioisotope containment and a clean environment was highlighted by several papers. Studies on migration of plutonium (11) in limestone and basalt indicated that the absorption coefficients were dependent on the types and amounts of other ions present in the solution. Migration coefficients were for flow along the surface of fissures and through the porus stone.

Spent fuel from nuclear reactors is stored under water at the Idaho Chemical Processing Facility. The fuel storage basin becomes contaminated with 137Cs and 90Sr from fuel elements that leak and from cut pieces of fuel. Both 137Cs and 90Sr were removed from the coolant by ion exchange (12).

One requirement for any nuclear facility is to monitor the effluent waste water to show compliance with existing standards. A sequential procedure was described for the separation of the transuranic elements from up to 60 l of water sample (13).

FOLLOW-UP CHRONOLOGIES

This concise view of the approach to Radioactive Waste Management can be opened like a time capsule some twenty years later to validate our vision of the seventies. I'll now address three of the categories adding some interim chronology to see how the predictions fared. It is not possible in the time available to make an in-depth coverage on the chronological progress for each discussion item; rather, the speaker has taken the liberty of selecting his version of key events or status points. Members of the audience should feel free to overlay this presentation with their own perceptions.

Commercial High-Level Waste Chronology

A chronology for the Commercial High-Level Waste category is shown in Fig. 1. The column under 1974 reflects the positions taken in the symposium. Author selected entries are inserted under the appropriate year and opposite the appropriate 1974 entry. The Nuclear Waste Policy Act (NWPA) of 1982 and its subsequent amendments have played an important part in this chronology. For instance, even though an interim storage facility was originally envisioned and the thought was reinforced via NWPA, no such facility is in sight. Permanent geologic storage, seemingly on the fast track in the eighties has been loosing steam this decade much to the dismay of the utilities. Activities to develop commercial high-level waste forms have been virtually discontinued in the U.S. due to the demise of fuel reprocessing. There is more action on that in the foreign high-level waste category, however. Nuclear Fuel Services at West Valley is the only commercial reprocessing waste in the U.S. A vitrification facility has been constructed and demonstrated in the cold mode at that location. Hot start-up for a 30 month campaign is planned in March, 1996. The waste filled canisters will be temporarily stored in the decontaminated process cells.

Fig. 1

Defense High-Level Waste Chronology

Defense High-Level Waste Chronology is displayed if Fig. 2 with a format similar to the previous figure. In this case, the three DOE sites where fuel reprocessing was conducted were considered. With a number of alternative waste forms under consideration in 1974, the first obvious move was to select a single reference waste form. There was not enough space on the figure in the early eighties to record all of the independent studies conducted by independent peer groups, the National Academy of Engineering, etc., etc. Fortunately, these thoughts were

collected by Ray Walton et al in 1982, reporting the decision that borosilicate glass was to be the reference waste form for DOE with crystalline ceramic as the alternative waste form. The Defense Waste Processing Facility, DWPF, was designed and constructed at SRP in the 1980's; it has gone through waste qualification tests with 75 full sized canisters of simulated waste being produced. A number of these canisters were cut open to assess the glass quality and the compositional homogeneity. The results were suitable for qualifying the waste form and vitrification process. Hot feed start-up is anticipated early in 1996. At ICCP, the calcined waste form with a high calcium, alumina, zirconia, borate, and fluoride can be best solidified into a ceramic waste form. At Hanford, the initial plans for a vitrification facility, the Hanford Waste Vitrification Plant, HWVP, that would convert the double-shell tank wastes to borosilicate glass was abandoned in 1992. Since then an RFP has been prepared for early 1996 issue on the subject of privatization of the vitrification process. To summarize, a substantial technical evaluation established borosilicate glass as being the reference waste form for Defense Wastes. Only one vitrification facility has been put on line to date. Addition of future facilities in the DOE complex is a long way off.

Fig. 2

Foreign High-Level Waste Chronology

The Chronology presented in Fig. 3 for Foreign High-Level Waste is impressively far ahead of the U.S. position. First, it is clear that the glass waste form option was widely accepted early on. With vitrification plants that have operated since the 1980's, the high-level waste management programs outside the U.S. are an outstanding reflection of what was envisioned at the 1974 symposium.

Fig. 3

SUGGESTIONS FOR THE FUTURE

High-Level Waste Management in the United States will be best served by staying the course and completing vitrification of Defense High-Level Waste materials as expeditiously as possible. A substantial experience base will be forthcoming from the Savannah River Defense Waste Processing Facility operation and from the West Valley Demonstration Project. Implementation of these experiences with modifications that accommodate feed type variations should assure a safe, timely completion of this challenging task.

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25-2

HLW MANAGEMENT AT THE IDAHO CHEMICAL PROCESSING PLANT - HISTORICAL DEVELOPMENT OF WASTE CALCINATION AND IMMOBILIZATION*

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ABSTRACT

Irradiated nuclear fuel has been reprocessed at the Idaho Chemical Processing Plant (ICPP), which is a part of the Idaho National Engineering Laboratory (INEL); formerly the National Reactor Testing Station (NRTS), since 1953 to recover uranium-235 and krypton-85 for the U.S. Department of Energy (DOE); formerly US Atomic Energy Commission (AEC) and US Energy Research and Development Administration (ERDA). The resulting acidic high-level liquid radioactive waste (HLLW) is stored in stainless-steel, 1,100-m3, single-shell tanks in underground concrete vaults. A solidification process was developed during the 1950s to form a granular calcine solid from the acidic HLLW with a seven-fold volume reduction. An engineering-scale demonstration, the Waste Calcining Facility (WCF) was constructed and operated in 1963. After the successful demonstration of the process, the WCF was continued as a production facility through 1980, calcining 15,000 m3 of HLLW to 2,160 m3 calcine.

The New Waste Calcining Facility was designed and constructed based on the operating experience of the WCF and began operation starting in 1982. The calcined waste is stored near-surface in stainless-steel bins within concrete vaults. The bin sizes are approximately 4-m diameter by 12.5 to 18.5-m high. Some of the bins are cylindrical and others are of an annular configuration. Currently, there is an inventory of 3,800 m3 HLW calcine at ICPP consisting primarily of alumina and zirconia-based calcines, resulting from dissolution of aluminum and zirconium fuels, respectively, and zirconia-sodium blends. The amount of alumina, zirconia, and zirconia-Na calcines is approximately 560, 1250, and 1750 m3, respectively. An additional 240 m3 calcine inventory consists of calcines from processing other minor fuels and start-up bed material. Several technologies have been identified to date that could immobilize calcine; these include vitrification and glass-ceramic processing. Preliminary scoping tests were run in the 1960s, and laboratory testing was started in the 1970s to develop glass formulations for ICPP calcines. In the 1980s, glass ceramic formulations were tested to produce a high waste loading waste form with a nearly three-fold lower volume than the equivalent glass waste forms. For alumina calcine, waste loadings of up to 29 wt% and 24 wt% could be obtained in a borophosphate and borosilicate glass, respectively. For zirconia calcines, waste loadings of 33 wt% were observed using a borosilicate frit. Nonradioactive laboratory- and pilot-scale and radioactive laboratory-scale tests were run using the borosilicate frit 127, and MCC-1 and MCC-2 leach tests indicated that there did not appear to be significant differences in the glasses formed using simulated zirconia calcine at laboratory or pilot scale and using radioactive zirconia calcine at laboratory scale. This paper focuses on some of the history and early experience which has helped the ICPP successfully accomplish its mission safely and with minimal impact on the environment.

INTRODUCTION AND PURPOSE

Irradiated nuclear fuel has been reprocessed at the Idaho Chemical Processing Plant (ICPP), which is a part of the Idaho National Engineering Laboratory (INEL); formerly the National Reactor Testing Station (NRTS), since 1953 to recover uranium-235 and krypton-85 for the U.S. Department of Energy (DOE); formerly US Atomic Energy Commission (AEC) and US Energy Research and Development Administration (ERDA). A decision was made at the start of reprocessing to store the resulting acidic high-level liquid radioactive waste (HLLW) in stainless-steel 1100 m3 single-shell tanks in underground concrete vaults, rather than continue the practice at that time of HLLW neutralization and storage in carbon steel tanks. In another innovation in the waste management practice, a solidification process was developed during the 1950s to form a granular calcine solid from the acidic HLLW with a seven-fold volume reduction. A pilot-scale demonstration, the Waste Calcining Facility (WCF) was constructed and operated in 1963. After the successful demonstration of the process, the WCF was continued as a production facility through 1980, calcining 15,000 m3 of HLLW to 2,160 m3 calcine. The New Waste Calcining Facility was designed and constructed based on the operating experience of the WCF and began operation starting in 1982.The calcined waste is stored near-surface in stainless-steel bins within concrete vaults. The bin sizes are approximately 4-m diameter by 12.5 to 18.5-m high. Some of the bins are cylindrical and others are of an annular configuration. Currently, there is an inventory of 3,800 m3

HLW calcine at ICPP with compositions shown in Table I. Not shown in Table I is zirconia-Na calcine, which has a similar composition to Fluorinel-Na calcine. The amount of alumina, zirconia, zirconia-Na, and fluorinel-Na calcines is approximately 560, 1250, 950, and 800 m3, respectively. The remaining 240 m3 calcine inventory consists of calcines from processing other minor fuels and start-up bed material with compositions shown in Table I.

Table I

Several technologies have been identified to date that could immobilize calcine; these include vitrification and glass-ceramic processing. Preliminary scoping tests were run in the 1960s, and laboratory testing was started in the 1970s to develop glass formulations for ICPP calcines. In the 1980s, glass ceramic formulations were tested to produce a high waste loading waste form with a nearly three-fold lower volume than the equivalent glass waste forms. For alumina calcine, waste loadings of up to 29 wt% and 24 wt% could be obtained in a borophosphate and borosilicate glass, respectively. For zirconia calcines, waste loadings of 33 wt% were observed using a borosilicate frit. Nonradioactive laboratory- and pilot-scale and radioactive laboratory-scale tests were run using the borosilicate frit 127, and MCC-1 and MCC-2 leach tests indicated that there did not appear to be significant differences in the glasses formed using simulated zirconia calcine at laboratory or pilot scale and using radioactive zirconia calcine at laboratory scale.

This paper focuses on some of the history and early experience which has helped the ICPP successfully accomplish its mission safely and with minimal impact on the environment.

EARLY ICPP DEVELOPMENT

The ICPP was purposely located on an arid tract of withdrawn public land previously used by the U. S. Navy for testing guns and munitions. The sparsely populated surrounding area and its remote location were particularly suitable for processing spent and irradiated nuclear fuel , then in its infancy. The ICPP was originally designed to process a widely-varying load of enriched production reactor fuel material as well as uranium and uranium-aluminum alloy fuel elements from a number of experimental reactors. The original design also provided space for the addition of processing facilities for stainless steel and zirconium fuel elements, and for other radio-chemical processes.

The development of the initial plant processes and the initial design scoping of the plant were carried out by Oak Ridge National Laboratory (ORNL). After completion of the design studies the AEC Processing Advisory Committee recommended in April 1950 that the plant be built at the NRTS in Idaho. In June, 1950 the Foster Wheeler Corporation was selected as the architect-engineer with responsibility for all detailed plant design except for the processing equipment. The American Cyanamid Company was selected as the operating contractor in late 1950. About this time Bechtel Corporation was selected as the construction contractor and excavation started.

Construction was essentially complete by mid 1952. After a preliminary test, cold run, and low radiation level run period, the plant was placed in routine production in March 1953. Late in 1953 Phillips Petroleum Company (Phillips) assumed responsibility for plant operation. Extended periods of high efficiency operation demonstrated the worth and dependability of the ICPP for processing a variety of irradiated reactor fuel (1).

Increasing liquid waste volumes were projected as more and more fuel was committed for reprocessing. A process for waste volume reduction was desired and pyrochemical processes were evaluated (2). Eventually development of a fluidized-bed calciner reduced waste volume by 7-fold and converted the liquid waste to a less mobile solid-granular form. This work was initiated in about 1953. The calciner development, based on a concept and testing by ANL, used a 15-cm (6-inch) diameter pilot plant (3). Positive results of the early development tests convinced the AEC, in 1956, to build a demonstration facility, The Demonstration Waste Calcining Facility (DWCF), at the ICPP. Construction began in 1958, at a cost of $6 million, and was completed in 1961. Phillips assigned a special section of its Technical Branch the responsibility to begin testing and to prepare to start the facility. Cold testing began in 1961 by Operations personnel and continued for one and one half years. In November 1962 , operation of the DWCF was demonstrated. The "D" was dropped from the acronym and the facility became known as the WCF. Radioactive feed was introduced on December 8, l963. The first run lasted 11 months, until October 15, 1964, when the first Calcined Solids Storage Facility (CSSF) was filled. More than 1800 m3 (500,000 gallons) of radioactive liquid waste from three waste tanks were converted to 2100 m3 (7,500 ft3) of solids. The net output exceeded the design rate by 15% and the process operated at 99% efficiency. Figure 1 is a photograph of the WCF and CSSFs 1-3 after construction.

Fig. 1

As fuel shipments to ICPP increased additional CSSF's were built and the WCF was modified for higher feed rates. The facility operated until March 1981, calcining about 15,000 m3 (4 million gallon) of liquid radioactive waste. The WCF was then replaced by the New Waste Calcining Facility (NWCF) which began hot operations in September 1982. The ICPP fuel storage and dissolution and remote analytical facilities were also subsequently replaced by state of the art facilities to improve operational safety and allow throughputs consistent with the demand of fuel receipts. Figure 2 is a 1985 photo of the NWCF and additional CSSFs 4-6. Note the relative size of the WCF and location of the subsurface HLW liquid tanks.

Fig. 2

ICPP WASTE MANAGEMENT CONSIDERATIONS AND DESIGN DECISIONS

Basic decisions and considerations had a direct bearing on the future of reprocessing. Specific issues included whether or not to neutralize acid wastes from the process, selection of a calcination method, design of waste tanks, number of waste tanks required, and design of CSSF's.

Acidic Raffinates Storage: A decision was made not to neutralize raffinates with sodium hydroxide as was commonly practiced in those days. This was possible due to implementation of improvements in extraction technology. Neutralization would have significantly increased the volume of high-level liquid waste generated and complicated its use as an isotope feed stock for ORNL's expanding isotope recovery operations. A decision to neutralize would have precluded continuous fluidized-bed waste calcining because of the inability of the process to handle high concentrations of alkali nitrates present in the solid product.

Design and Number of Liquid Storage Vessels: A decision was made to construct a limited number of 1,100 m3 (300,000 gallon) waste tanks of stainless steel. Stainless steel was available after the end of WW II and was necessary because the waste was acidic. The initial tanks were

constructed of type 348 stainless steel. Corrosion testing was used to justify the substitution of more economical 304L stainless steel in future tank construction. Tanks planned for use with high-level waste were equipped with in-tank cooling system to maintain the solution at low temperature which minimized material corrosion. Acidic waste with no precipitates minimized the formation of galvanic corrosion cells between solution precipitates and the tank material. Today the ICPP has only 11 major waste tanks because the tankage space as the tanks have been emptied using the calcining process. Also, no tank failures have been experienced, even for tanks used to store aluminum-complexed-fluoride solutions.

Selection of Calcination Process: Initially the design liquid extraction raffinate was aluminum nitrate based. Candidate solidification processes included pot calcination, spray calcination and rotary kiln calcination (2). Later the fluidized bed process was chosen because it was continuous, was readily adaptable to remote operations, had no moving parts to wear out, had adequate throughput capacity to meet ICPP requirements, and was adequately developed for the demonstration (4).

Design and Construction of Calcined Solids Storage Bin Sets: A decision was made that the calcined solids would be stored in stainless steel bins with extended design life. The vessel design was based on a critical centerline temperature that would prevent the radionuclide migration. The first CSSF was conservatively designed and constructed of annular bins; subsequent CSSF's incorporated observed heat transfer experience and employed a more economical cylindrical bin design . In addition, after CSSF #1, subsequent CSSF's were designed with retrieval ports to accommodate any future desire to retrieve the calcine for subsequent processing.

ANL CALCINATION TECHNOLOGY DEVELOPMENT

The concept of calcining radioactive liquid waste by atomizing the liquid in a heated bed of inert particles was originated by ANL. Initial calcination development was then accomplished jointly between ANL and Phillips Atomic Energy Division through 1961.

Early development by ANL (5) consisted of brief feasibility studies using a 7.5-cm (3-inch) diameter calciner and additional more extensive studies using both unshielded and shielded 15-cm (6-inch) diameter calciners. Tests with the 7.5-cm diameter calciner were all with cold feed. Subsequent tests with the 15-cm diameter calciners were both cold and up to 12% spiked, hot, as-produced feed from the ICPP. Initial development was based on testing acidic aluminum nitrate solution but ANL later extended the work to include Hanford type PUREX waste. ANL development with aluminum nitrate waste consisted primarily of testing the behavior and treatment of fission products in the calcination process using the six-inch-diameter calciner. Radioactive feed consisted of ICPP first-cycle raffinate diluted with simulated cold aluminum nitrate raffinate in a dilution ratio which ranged from 1:1000 to 1:8. Basically ANL pilot plant experience indicated the fluidized bed process was feasible from the operating viewpoint and that ruthenium was the only volatile fission product. The tests showed that virtually all activity except ruthenium remained in the solid product over the temperature range of 350 to 550C. The ruthenium volatility was determined to be very temperature sensitive. At 350 degrees operating temperature about 88% of the ruthenium was volatilized. Whereas at 550C less than l% volatilized. Partial condensation of the off-gas removed 99.8% of the entrained non-volatile

activity. Later use of a venturi scrubber and silica gel bed prior to partial condensation resulted in decontamination factors of 1,000 to 10,000 for ruthenium and 100,000 for non-volatile fission products.
Both bayonet and convoluted filters were tested for calciner off gas cleanup. The bayonet sintered metal filters proved highly efficient for removal of particulate from calciner off-gases. The filters were especially effective when coated with a dust layer. They were so effective that analysis of condensate and gas samples detected no radioactivity other than ruthenium. Failure of the sintered metal filters after being subjected to extreme temperatures (700C) was caused by embrittlement. Additional tests indicated that if operating tests did not exceed 400C no embrittlement would occur. Convoluted metal filters were tested and found to be unsatisfactory. Breakthrough of non volatile activity was 10-fold greater than with bayonet filters.
Since sintered metal filters required much development, venturi scrubbers were studied as an alternative for cleaning the off-gas. The use of silica gel adsorbers were studied for removal of volatile ruthenium. A primary advantage of these devices was low waste volume generation. The scrub solution from the venturi could be recycled to calciner feed and the regenerating solution from the adsorber was very small. Both scrubber and adsorbers were then added to the ANL 15-cm (6-inch) calciner to study particulate removal in parallel with ICPP studies. Venturi scrubbers were very effective in removing particulate from the off-gas, and silica gel adsorbers were very effective in removing volatile ruthenium. Tests also showed that the silica gel could be readily regenerated.

ICPP CALCINATION TECHNOLOGY DEVELOPMENT

Development at ICPP in support of a DWCF was the responsibility of Phillips, the operating contractor at ICPP from the fall of 1953 until 1967. Calcination of acidic aluminum nitrate waste using fluidized bed technology reported by ANL in 1955 was based on conceptual studies. A 15-cm (6-inch) diameter pilot plant calciner patterned after the one used by ANL but with a number of changes to help solve operability problems reported by ANL was then built and operated by Phillips personnel at the NRTS (6). The purpose of this pilot plant testing was to provide equipment and process scale up data and to define process parameters for designing the DWCF. The 15-cm diameter calciner pilot plant as reported by Grimmett in 1957 (7) was fed simulated cold aluminum nitrate waste at rates ranging from 20 to 110 ml/min, at fluidized bed operating temperatures ranging from 180C to 500C, and at operating period durations of 8 to 336 hours.
The 15-cm diameter calciner vessel was 1.7 m (5.5 feet) long and contained a charging tube, a thermowell and off-gas tube at the top of the vessel. The fluidizing air distributor consisted of a flat plate with twelve 1.3-cm (0.5-inch) diameter holes, spaced on a 4.3-cm (1-11/16 inch) triangular pitch with 1.3-cm (0.5-inch) tubes welded to the top of the plate to extend 0.6 cm (1/4 inch) above the plate. The top of the tubes were capped with 1.4-cm (3/4-inch) discs and 0.3-cm (1/8-inch) holes were drilled at 90 degree intervals. Air was introduced to the bottom of the distributor through a 1.4-cm (3/4-inch) pipe. The calcined product was removed continuously by means of a one-inch draw-off line extending 61 cm (2 feet) above the distributor plate. Heat was supplied by 18 internal electric heaters with 15-cm (6-inch) heated lengths located horizontally inside the calciner bed. Liquid feed was introduced under pressure through commercially available pneumatic atomizing spray

nozzles. To provide operating data, pressure taps and thermocouples were located throughout the bed. Calciner auxiliary equipment included a cyclone, a spray scrubber, condenser, and fluidizing gas preheater. Operating variables studied were feed rate, bed temperature, fluidizing velocity, feed nozzle air-to-feed ratio, product draw off, and liquid feed composition. A problem with size distribution not reaching steady state made it difficult to determine various effects of operating variables on the product.

Feed Rate. It was generally determined that feed rate influenced particle size distribution and possibly bulk density. Low feed rates generally produced smaller particles than runs at higher rates. The effect of feed rate on bulk density was not conclusive.

Bed Temperature. The data showed decreasing bulk density with increasing bed temperature throughout the temperature range studied. The bed temperature also had a profound effect on particle-size generation. Low bed temperatures (300C) produced large mass mean particle diameters, whereas, 500C tended to produce fines.

Fluidizing Velocity. Changing fluidizing velocity appeared to change particle size distribution.

Feed Nozzle Air-to-Feed Ratio. Runs were made at constant air to feed rates, but low atomizing air flow rates possibly caused agglomeration tendencies.

Product Draw Off. A relatively small amount of air is passed countercurrent to product overflow. The air is an effective means of controlling particle size withdrawal.

Feed Characterization. The concentration and composition of feed to the calciner was not studied but was expected to have a strong effect on calcine structure and size distribution. Although most runs were made with aluminum fuel, some exploratory runs were made with stainless steel and zirconium type fuel raffinates.

The results of the pilot plant tests verified the feasibility of calcining aluminum type waste. The advantages of the technology include independence from complicated moving parts, excellent control of bed temperature with uniform high heat transfer rates, reduction in corrosion potential of stored product, adaptability to remote operation, and excellent control of particle size and properties.

ICPP CALCINATION DEMONSTRATION AND HOT OPERATIONS

Pilot plant tests established target ranges of principal operating values for the full scale demonstration. The ranges for the variables was 1) calciner bed temperatures of 400 to 500C, 2) fluidizing air velocities of 23 to 40 cm/sec (0.75 to 1.3 feet/second), and 3) air to liquid feed nozzle ratios of 500 to 750. The facility was operated on cold simulated feed for more than 4500 hours to establish that the equipment was adequately reliable and safe for hot remote operations. Prior to "hot" startup of the WCF an extensive safety analysis was conducted to assure no surprise hazard existed. Through 1973, the WCF and ICPP reprocessing facilities were operated alternately with common crews during campaigns that lasted several months each. This provided maximum use of both maintenance and operation personnel. In order to increase ICPP throughput separate dedicated crews were provided to allow concurrent operation of the WCF and reprocessing facility. Ultimately, the WCF operated until March 1981, calcining nearly 15,100 m3 (4.0 million gallons) of liquid radioactive waste.

A simplified flow diagram of the WCF process is given in Fig. 3. A major portion of the equipment, ~70%, is for off-gas cleanup. The cleanup equipment includes a cyclone, a quench tower, a venturi scrubber, entrainment separators, silica gel adsorbers, and HEPA filters.

Fig. 3

The WCF calciner vessel was four-feet in diameter. Radioactive feed was introduced into the bed through pneumatic atomizing nozzles. The feed was sprayed through three nozzles equally spaced in a horizontal plane at a total rate of 320 to 530 L/hr (85 to 140 gallons/hour) including waste, additives, and scrub recycle. During the first three campaigns and until June 1969, heat was supplied by an in-bed heat exchanger bundle using sodium-potassium alloy which operated at a bundle temperature of up to 760C. The NaK was pumped by an electromagnetic pump and heated in an oil fired furnace. Preheated fluidizing air was introduced through 14 capped orifices equally spaced on a distributor plate. Operation at superficial fluidizing velocities between 25 and 40 cm/sec (1.0 and 1.3 feet/second) was satisfactory with average bed particle size ranging from 0.6 to 0.75 mm.

To allow increased WCF capacity and reliability, an in-bed combustion system using oxygen-atomized kerosene was installed in 1970 (8). The use of in-bed combustion resulted in increased heat transfer rates, a change in the behavior of fission product ruthenium, and lower metal wall temperatures. A less volatile ruthenium species reduced concentrations in the off-gas by a factor of 100 to 200. However, decontamination on WCF internal surfaces became more difficult.

Boric acid was added batchwise to aluminum nitrate feed solution to suppress the formation of alpha alumina and insure the product remains in the amorphous rather than crystalline form. The addition of boric acid is essential to provide particle size control, minimize decontamination problems, and reduce erosion of pumps, valves, and piping by making the aluminum specie soluble in the nitric acid scrub solution.

Calcium nitrate is added to fluoride bearing zirconium type waste to complex free fluoride. This is necessary to reduce corrosion on downstream process equipment and piping and to control fluoride volatility.

The granular calcined solids product is pneumatically transferred to and stored in vented stainless steel bins located in concrete vaults.

NWCF DESIGN, CONSTRUCTION, AND OPERATION

The WCF was retired in 1981, to be replaced by a state-of-the-art NWCF. Anticipated increasing fuel receipts coupled with excessive wear and the desire to reduce exposure to personnel during the operation/maintenance cycle called for a replacement calcining facility.

Wear and tear on the WCF equipment after many years of operation result in increased delays for maintenance refurbishment. This coupled with a need to increase throughput to accommodate increased fuel shipments prompted the need for a replacement. Design of the NWCF was initiated in 1974 (9) and construction started in 1976. The facility incorporated the newest fluidized bed calcination, off-gas cleanup, remote operations and maintenance, and decontamination technologies. The design throughput was increased to 11,300 L/day (3,000 gallons/day).

The calcination process didn't change significantly with the new facility. The construct incorporated many improvements. The calciner vessel active bed region was increased to a 1.5-m (5-ft) diameter with a 2.1-m (7-ft) diameter expanded upper section fabricated from 347

stainless steel. Downstream vessel sizes correspondingly sized. Also more corrosion resistant Nitronic 50 was used to handle more hostile (trace fluoride and chloride) chemical environments and higher operating temperatures. WCF experience identified high wear and exposure areas. To greatly reduce or eliminate such problems, high maintenance equipment (pumps, valves, flanges, electrical connections, and off-gas filters) was placed in readily accessible corridors for remote replacement using master-slave and electro-mechanical manipulators and cranes. The remote capabilities installed at NWCF were tested in a full-scale mock up facility at the INEL and positioned for easy removal and replacement without lengthy shut downs or high radiation exposures to personnel. Failed equipment was transferred directly to integral shielded facilities for remote decontamination and subsequent repair or disposal. NWCF operating experience has demonstrated the effectiveness of this concept and design.

The NWCF began hot operations in September 1982, after extensive system operability and cold testing, and calcined 6,000 m3 (1.6 million gallon) of waste before shutting down in March 1984 when all available liquid waste feed was processed. During the down period a new Distributive Control System was installed. The NWCF was restarted on September 30, 1987 and has operated successfully on various feeds and combinations of feed such as Fluorinel, zirconium, and blended Fluorinel-Na waste feeds. To date the NWCF has calcined over 13,600 m3 (3.2 million gallon) of radioactive liquid waste to 1,670 m3 (5,900 ft3) of calcined solids. Successful operation of the NWCF and WCF have combined have to eliminated the need to build twenty-six 1,100-m3 (300,000-gallon) waste tanks,

EARLY HLW IMMOBILIZATION TECHNOLOGY DEVELOPMENT

As calcining of acidic liquid high-level waste became a reality it became readily apparent that methods for final disposal of the calcined waste most be developed. The calcine, although physically and thermally stable, was leachable for several of its radioactive constituents, most notably cesium.

Scoping experiments were therefore started in the 1960s to evaluate potential final chemically-stable waste forms. Many possibilities were examined including glass, sulfur encapsulation, concrete, and ceramics (10). Final disposal of reprocessed nuclear-fuel waste was not of high priority during the sixties in the then current DOE structure and research on these aspects languished to the end of the decade.

By the early 1970s a reawakening occurred and research funds on some scale became available to all of the DOE sites managing high-level waste, but most notably at the Savannah River site because DuPont was committed to solidifying and disposing of its liquid waste prior to passing its operating contract on to someone else.

At the ICPP a small program was started by the mid 1970s to evaluate immobilization of its calcined waste for final disposal. The reprocessed high-level waste at the ICPP consists of varied nuclear-fuel compositions and is also different from that of either the Hanford or Savannah River high-level waste. The major differences are 1) the liquid waste is acidic as opposed to neutralized (basic) and 2) the ICPP waste compositions are totally different. The ICPP calcined waste are mostly cladding material. Wastes from the early 1960s consists mostly of aluminum, whereas later reprocessed waste consists mostly of zirconia, calcium fluoride and alumina and just for good measure stainless-steel fuels and sodium containing calcined waste is also present.

Although early waste immobilization experiments revisited such possibilities as encapsulation of the calcine in concrete, and explored sintered glass-ceramics (11), metal matrices (12), pelletizing (13), it became obvious more practical processes were needed. Hanford had done extensive experiments on conversion of their waste to glass but those glass formulations were not suitable for vitrifying the high-alumina and zirconia calcines at the ICPP.

Glass waste-form development at the ICPP proceeded rather efficiently because of the meager funding allocations available at the time. Calcination was very successful and engineered storage of the calcine was planned for many years. Therefore, there was no apparent urgency to immobilize the ICPP high-level waste for final disposal. Nevertheless an efficient glass composition was demonstrated for the high-alumina waste (14). Another glass composition was developed for the zirconia-calcium fluoride calcine (15). For alumina calcine, waste loadings of up to 29 wt% and 24 wt% could be obtained in a borophosphate and borosilicate glass, respectively. For zirconia calcines, waste loadings of 33 wt% were observed using a borosilicate frit. Glass from this waste has turned out to be of extraordinary leach resistance. In fact current Savannah River glass contains a small percentage of added zirconia to provide improved stability. Laboratory and pilot-scale melter experiments showed that the high zirconia-calcium fluoride glass can be produced at practical melter temperatures, and very durable melter refractories are required because of fluoride corrosiveness (16, 17, 18). Nonradioactive laboratory- and pilot-scale and radioactive laboratory-scale tests were run using the borosilicate frit 127, and MCC-1 and MCC-2 leach tests indicated that there did not appear to be significant differences in the glasses formed using simulated zirconia calcine at laboratory or pilot scale and using radioactive zirconia calcine at laboratory scale.

Because the calcines at the ICPP are of varied composition, a practical and simple computer code was developed to not only predict glass stability but also determine a practical glass composition for a proposed waste composition. The code has worked effectively to determine aspects of permissible waste loading, immobilized waste durability, and amount and type of additives needed.

Waste volume reduction became the byword during the 1980s. The volume of a glass waste form required to immobilize the ICPP calcined waste was considered large mostly because of requirements to add calcium to the waste during calcining to stabilize the fluorides present in the liquid waste. This addition is required even if the liquid is directly vitrified. Direct liquid vitrification does not save on glass volume and is very corrosive to melter electrodes.

Waste volume reduction efforts resulted in the development of a glass-ceramic waste form during the 1980s for the ICPP calcined high-level waste (19, 20). A glass-ceramic can accommodate up to about 70 wt% calcine whereas a glass can only accommodate about 30 wt% calcine. In addition the glass-ceramic is about 50% more dense. Therefore final volume reduction is greatly reduced and the glass-ceramic is equally as durable, if not more so. Processing a ceramic waste form is more sophisticated, ideally requiring an isostatic pressing process and special particle and containment handling technology.

Recent evaluations have proposed dissolution of calcine and separation of a small high-activity waste fraction, followed by a significantly smaller-scale vitrification process (21). The low activity waste would be

disposed on site as a grout. Evaluations are continuing with a full-scale facility projected to process all of the calcine by 2035 in order to meet a legal agreement with the State of Idaho.

CONCLUSIONS

The ICPP fuel reprocessing operation that began in 1953 has resulted in approximately 3800 m3 of solid calcined HLW which is stored safely in stainless steel bins with a long design life and is scheduled for immobilization for shipment of the glass off-site by 2035.

A set of decisions and focused development activities during the early history of the ICPP has resulted in these current relatively safe conditions, with no major waste leaks to the environment compared to some of the other DOE sites. This paper describes some of these developments which illustrate successful application of an innovative technology including development through radioactive pilot scale and pilot demonstration to result in full-scale waste processing that is unique in the DOE complex. As a result, the HLW storage at the ICPP is currently under a relatively minimal environmental risk compared to other comparable DOE sites.

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25-3
RADIOACTIVE WASTE DISPOSAL - THE EARLY YEARS (1943 - 1970)

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ABSTRACT
Though there were early concerns about the fate of the radioactive materials discharged to the environment (Arthur Compton's 1943 letter concerning the likely fate of materials that would be discharged to the Clinch River when the Manhattan Project moved to Oak Ridge), there was no major public disclosure until the control of atomic energy was put under civilian control in 1954 and the first International Conference on Peaceful Uses of Atomic Energy in Geneva, Switzerland by the United Nations. The first International Conference on Radioactive Waste Disposal organized by the International Atomic Energy Agency was held in 1959 and the first congressional hearings on radioactive waste disposal were held in the same year. The end of innocence began in 1970 with the start of the environmental movement triggered by Silent Spring by Rachel Carson but implemented by the National Environmental Policy Act but really gained full momentum with 1984 law suit which put the hazardous wastes of the Department of Energy under the legal jurisdiction of the U.S. Environmental Protection Agency. Disposal practices at the major sites will be reviewed with particular emphasis on practices at Oak Ridge where the author was head of radioactive waste disposal research. International aspects will also be presented and compared to U.S. practices from the perspective of the author who was head of radioactive waste disposal at the International Atomic Energy Agency from 1960-1961. Recent, since perestroica, information about early Soviet practices will also be

discussed. Despite revisionist views of the early days, it is clear that in the U.S. the majority of the practices were the industrial norm or the tools for better information were not available (neither the alpha nor the gamma spectrometer had yet been invented) or the general scientific ignorance (discharge of wastes to the ground was considered standard practice). Speculation about future disposal practices will be presented.

INTRODUCTION

Some highlights of the USA Radioactive Waste Policy - Past, Present and the Future as seen through my own clouded lens will be presented. My views are prejudiced because I have been a participant in many of the activities I will describe. While the present day situation is known to many of you, I am always amazed that in the USA, due to rapid turnover of civil servants engaged in these tasks and the almost as rapid turnover in the operating contractors of the DOE sites and the somewhat slower turnover in the personnel at the DOE National Laboratories that the corporate memory of what took place has faded. This is vividly shown in the revisionist history of Radioactive Waste Disposal in the USA. (1) I hardly recognized some of the events described there. Even the polemical book Nuclear Imperatives and Public Trust by Luther Carter gives a better insight into the program. (2)

The present situation is clouded by the election of a new congress this last year that has little knowledge or interest in the topic unless the budgets can be cut. Nor can we expect much resistance to this juggernaught from the executive and judicial branches because as Finley Peter Dunne, a noted political satirist at the turn of the century, had his cartoon character Mr. Dooley say "No matter whether th' constitution follows th' flag or not, th' supreme coort follows th' iliction return."(3) With the President promising to veto many of the Congress' Budget Bills because of the most egregious parts of the bills, none of the major fights are about nuclear energy or radioactive waste disposal. In fact, the Department of Energy's budget passed the Congress and was signed into law on the first try. Where this will all end, is still uncertain. In any case, it will not be what was before. Finally, I would like to make some guesstimates and recommendations about future policy. This is the safest part of the presentation because none can dispute the factual content and as Abraham Lincoln said, "The world will little note, nor long remember, what we say here" (4) so I do not expect to hear from any of you in the next century to say "Frank, you were dead wrong."

The comments will be divided into three time periods as 1943 to 1990 (past), 1990 to 1995 (present) and post 1995 as the future. Because of the time limitations, only a few topics will be discussed of the major categories of wastes: high level wastes, and low and intermediate level wastes and their environmental impact, and the governmental policy and regulatory framework and public involvement that prevailed during these time periods.

HIGH LEVEL WASTES

Till ownership of nuclear reactors was allowed in the civilian sector in 1954, McMahon Bill (5) and President Eisenhower announced the Atoms for Peace Initiative (6), all U.S. fuel was reprocessed after short irradiation times to maximize plutonium-239 production for bomb purposes. A similar concept, short storage times after removal from reactors, was carried over to irradiated civilian reactor fuel. In fact, this was mandated in the regulations that the solidified waste from reprocessing would be transferred to a federal repository no later than 10 years after

separation. (7) This was the legal policy of the USA for many years, even after President Carter indefinitely suspended reprocessing of spent fuel derived from the production of civilian nuclear power in 1977 because of fears of nuclear proliferation (8) despite the fact that the INFCE studies (9) and the OTA study (10) indicated that spent fuel and reprocessing wastes would not be the preferred routes for terrorists groups to obtain nuclear weapons. This also has relevance today with heavy political pressure to transmute the plutonium in high level wastes to reduce proliferation threats. With the world now awash with surplus plutonium from dismantled weapons, it seems silly to oppose reprocessing on proliferation grounds. Because of this concern, the United States has attempted to maintain a rigid separation of materials from weapons production and those from civilian use. President Reagan weakened this rigid separation when he authorized the burial of spent fuel from civilian reactors in the same repository as high level waste from weapons production. (11) Since the U.S. is the only nation that will not allow reprocessing and commingling of civilian and weapons nuclear material, the net deterrent is not great. This is true, particularly, if it is kept in mind that the USA has only 1/4 of the world's nuclear power reactors and that together these power plants produce 20 tons per year of new plutonium. Though this plutonium is not weapon grade, this does not mean that explosive devices cannot be made from this material. The decision of what to do with the 50 tons of surplus U.S. weapons grade plutonium already available has not yet been decided but some indication of the options available can be had in the recent NAS/NRC report. (12) It may be thought that this is too much emphasis on nuclear weapons and civilian power plutonium problems. It should be remembered from the consequences of Chernobyl, that one terrorist detonation of a nuclear device would doom nuclear power and any rational discussion of radioactive waste disposal for many, many years. One of the first lectures given to new employees at Oak Ridge, contained the sentence, "You can get away with most anything here except a criticality accident." (13)

Discussion of high-level waste disposal had already appeared in government documents as early as 1944, (14) and in reviews by the NAS/NRC as early as 1956. (15) However, it was not until the NAS Committee meeting in 1955 that a recommendation was made for deep geological disposal with emphasis on rock salt as the medium of choice. (16) This decision by a non-governmental group resulted in the start of experimental investigations on deep geological disposal in salt in 1957 at the Oak Ridge National Laboratory with a first publication on the progress in 1958. (17) The initial experiments were simulated high-level liquid wastes in the operating Hutchinson, Kansas Salt Mine of the Carey Salt Company. Despite assurances by geologists that salt was impermeable, it was quickly found that the simulated waste moved rapidly along the interfaces of bedding planes of anhydrite and polyhalite and salt. However, the success of the laboratory work on stability of rock salt under high irradiation and the theoretical studies on the heating of the salt from high level wastes prompted a move to the inactive Carey Salt Mine in Lyons, Kansas. In today's gridlocked bureaucratic environment, it is useful to recall how the Lyons mine was chosen. Four of us, Joe Lieberman, head of the Sanitary Engineering Section of the AEC (see paper by Walter G. Belter this meeting), Ed Struxness, Associate Director of the Health Physics Division at ORNL, myself and Bill Heroy, one of the most distinguished geologists of the time, met in a room at the Cosmos

Club in Washington. The Cosmos Club, a private institution, was then the informal headquarters of the scientific elite in the U.S. and particularly for the earth sciences. Bill Heroy and I had visited all of the operating and accessible salt mines in the U.S., as well as a number of other mines to help determine the location of these first tests of a geological repository for high level wastes. In an afternoon, we did an impromptu, informal multiattribute utility analysis of potential sites with a sensitivity analysis of the weightings of the attributes (note, I have put what we did into present day jargon) and the choice of Lyons, Kansas was robust. Therefore, at the end of the meeting, Lieberman said to me "go ahead and do it Frank." That was it, the authorization and the appropriation. As an aside, I do not believe our budget ever came close to a million dollars per year. For local transportation, we had a shopworn panel truck from the government car pool in Kansas City. Yet despite this, the first set of Engineering Test Reactor fuel assemblies (thermal flux of 1.5 x 1014 n/cm2sec and irradiation times of 23 days with an initial loading of 400 gm of 235U resulting in a curie content of 240,000 curies per canister at time of emplacement) were placed in the test facility during November 17-19, 1965. After having reached the design objectives of a total dose to salt of 5.3 x 108 rads and a peak temperature of 200C, the third and final set of irradiated fuel assemblies were removed on June 26-27, 1967. (18) Basically all of the equipment and procedures being used in test facilities today are based on designs developed then. What has been going on in the last 30 years? The results were so satisfactory that in June 1970, the AEC tentatively selected a full scale high level waste repository site in the bedded salt at Lyons, Kansas, pending confirmation tests, that would be completed in 1978 at a total cost of 25 million dollars. (19) With the political climate prevailing at that time, it would have been possible to do that. However, that site was found to be inadequate. Instead of utilizing an eminently suitable site (Naval Air Station) only 10s of miles from Lyons, there have been a long series of failed attempts to establish a centralized above ground surface interim storage facility for spent fuel and a repository. They have failed for complex reasons, both technical and political. After the abandonment of the Lyons site, the AEC in 1973, while making a centralized surface storage facility the "near-term objective," reaffirmed that the "major effort" was to be a Federal Repository "to be ready in the early 1980s." (20) In 1974, the new Chairperson of AEC, Dixy Lee Ray, downgraded geological disposal to a separate objective. (21) In 1977, the Department of Energy in the course of announcing a new centralized storage facility (away from a reactor) also noted that a geologic repository was still required. (22) The Interagency Review Group on Nuclear Waste Management in March 1979 called for a "stepwise approach to the development of HLW repository." (23) In February 1980, President Carter made the first Presidential Announcement on Nuclear Waste Policy (24) and said that the goal was permanent geologic disposal of nuclear waste. The Nuclear Waste Policy Act of 1982 favored permanently disposing of high-level radioactive waste in a geologic repository and provided a mechanism for narrowing the choices to three sites which would be examined simultaneously. (25) The repository would only be allowed to dispose of 70,000 MTU, less than the expected wastes from presently existing nuclear power plants excluding the vitrified wastes from defense activities, which could potentially be greater than that from civilian wastes. In May 1986 in a blatant

political move DOE announced the cessation of the search for the site for the second repository. (26) The Nuclear Waste Policy Amendments Act of 1987 directed DOE to terminate all site specific activities at all repository sites except Yucca Mountain. (27) Needless to say, no centralized facility for spent fuel storage is available today and defense high level wastes continue to be stored as liquids in tanks at Hanford, Savannah River, Idaho Falls and West Valley (the only commercial reprocessing plants to operate in the U.S.) and as calcined material in bins at Idaho National Laboratory. Civilian spent fuel is stored in pools at nuclear power plants. Where there is insufficient space in pools, dry storage on site in silos and "roach motels" is practiced.

Because of the concern that there would be insufficient space at civilian nuclear power plants for spent fuel and that a geological repository would not be available in time to receive this spent fuel, the Congress established a Monitored Retrievable Storage Review Commission to advise them on what should be done about a centralized storage facility. (27) As can be seen from Table I by 2025 most existing nuclear power plants will have reached the end of their licensed lifetimes and the problem will grow more acute. The MRS Review Commission after an extensive study recommended:

1) Construct Federal Emergency Storage Facility of 2000 MTU
2) Construct a User-Funded Interim Facility of 5000 MTU
3) Congress should reconsider interim storage by the year 2000, taking into account uncertainties that have been resolved in the meantime, new developments and experience with the facilities recommended in 1 and 2. (28)

Table I

Though almost universally declared a superior technical document, it was rejected by those who wanted a centralized storage facility with a much larger storage capacity and those who wanted no centralized facility at all, and the report disappeared into a black hole, but has recently been resurrected. Preparations for a user-funded facility (Mescalero Indians) are proceeding rapidly. Because of delays in the construction of a repository, the DOE is again attempting to site a centralized interim storage facility. Some members of Congress, including the influential Senator Bennett Johnson, formerly Chairman of the Senate Energy Committee, have become so discouraged by the slow pace and high cost of determining whether Yucca Mountain is suitable for a final repository, that he presented a bill to halt further studies of Yucca Mountain in favor of a centralized storage facility at that site. (29) That portion of the bill has been defeated. So where does that leave us with respect to high-level waste? Bluntly, a behind schedule program to vitrify defense wastes for the repository with no agreement in place that the wastes will be acceptable when and if they are vitrified, and no guarantee that the Federal Government will pay its fair share of the repository costs. For the civilian spent fuel program, producers feel betrayed because in 1970 Federal Regulations AEC required high level wastes at reprocessing plants to be converted to solid form within 5 years of its generation and transferred to a Federal Repository within 10 years after the irradiated fuel is reprocessed. (30) In the 1982 Waste Policy Act, the Congress seemed to promise that in exchange for the $0.001/ kwhr of nuclear energy sold, the DOE would take title to spent fuel no later than January 1998 and dispose of it in a geologic repository. Because of the 1970 regulations the utilities did not build

large spent fuel pools as the government would take the spent fuel within a few years after discharge and in 1982 guaranteed a date certain of January 31, 1998. Government lawyers do not agree with that interpretation of the law so the utilities are left with building their own on-site out of reactor spent fuel storage facilities with increasing likelihood that the State Public Utility Commissions will not allow both costs as direct pass through to the consumers. With the increasing competitiveness of the electrical generation and supply business in the United States, even if both costs were allowed, the nuclear utilities may not be able to recoup their costs and still remain competitive. So if there is progress such as starting up the Defense Waste Processing Facility (DWPF), it will be only token. Despite the accelerated schedule for making a decision about the suitability of the Yucca Mountain, it is highly unlikely that those dates will be met, as EPA has not yet issued its final regulations. The new regulations, required by court remand, (31) need to take into account the study of the requirements by the National Academy of Sciences/National Research Council (NAS/NRC) mandated by the Congress. (32) Even if they do adopt the NAS/NRC recommendations, (33) there are likely to be court challenges to the regulations. So despite the US's early successes 30 years ago in showing that it was possible to successfully emplace and retrieve spent fuel from deep geological facilities, we are likely to see a Belgian or Swedish attempt to pass the United States and become the first nation to construct a geological repository and place wastes in it just as the French overcame the early U.S. lead in reprocessing to become the first nation to successfully commercialize the process.

The Waste Isolation Pilot Plant, first authorized in 1980, (34) and completed in 1990 is still waiting to open.

LOW LEVEL SOLID WASTE DISPOSAL

In some ways the saga of low level radioactive waste disposal is even more fascinating than that of high-level waste disposal, because it is so widely dispersed and affects far more individuals and institutions. Consequently, there are a plethora of books dealing with the subject, but I shall only sketch in the historical background and complexities so as to reach some conclusions. From 1942 to 1950, all burial (almost entirely weapons connected wastes) was done nearby onsite, as though they were municipal wastes, which were largely unregulated at the time (I shall deal with liquid wastes as an environmental problem). As the amount of radioactive wastes from the civilian sector, power, industry, research etc., increased and it became apparent that the defense work would continue and grow, it became necessary to centralize and control the disposal of both the defense and civilian wastes at AEC sites during the 1950s. As the 1960s began, it became apparent that though the volume of waste was so substantial that commercial sites should be established. As a consequence, six commercial sites were licensed. Within a few years, three, all in the humid region, had closed. The reason for the closures have been administrative, regulatory, perception, and failure to achieve the containment that was promised even though the public health and environmental impact was low. Since then a fourth has closed and only Barnwell, South Carolina and Hanford, Washington are still open.

Once the shortcomings of municipal waste type disposal were recognized, enhanced containment and shielding of wastes with higher external radiation was practiced both at AEC's and commercial sites. The burial sites were still open to precipitation during disposal operation and

stabilization of wastes was not mandated or practiced. Only after the French developed the surface tumulus method involving stabilization and engineered containment did the U.S. begin to move in that direction. Most proposed compact designs are some variant of the French system though none of the sites presently operating in the U.S. use that system for all disposal at their sites. Some other countries, e.g., Sweden, United Kingdom and Germany have gone to deep underground geological disposal for greater containment.

Without going through the vicissitudes of low level waste compact formation allowed by the Low Level Waste Policy Act of 1980, (35) and the subsequent amendments, (36) no new sites have yet been licensed. Suffice to say that South Carolina has temporarily shattered the system by opening its site to anyone, except North Carolina, who can afford their fancy prices. How long this will continue is entirely problematical. The price of disposal has risen astronomically from $1/ft3 in 1975 (Class A wastes) to more than $300/ft3 at Barnwell at present. The failure to site new facilities has been due to a variety of causes which many English in her book on the topic has labeled Trust, Justice, Risk, Authority and the Quest for Legitimacy.

ENVIRONMENTAL

It would be inappropriate to conclude this retrospective view without some mention of the environmental impact of all of these developments. At one level the cost of remediation is at least on order of magnitude greater than the high level waste disposal program and much greater than that of the low level waste program. (38) More important, the environment in its broadest sense is where this impact on humans and the natural environment (humans, of course, are part of the natural environment) takes place. Though the early practice followed what was standard at the time, utilize the dilution capacity of the environment and the cleansing power of the soils, there was far less fundamental knowledge of these processes than there is today and the environmental standards were far different than they are today. Even as avid a proponent of restriction of radioactive dosages to humans as K.Z. Morgan was willing to accept geologists' views that shale was impermeable and allow low level liquid wastes to be discharged to open basins to evaporate because the potential evaporation was greater than precipitation. We quickly learned from the leakage from these open basins about heterogeneities and anisotrophy and kinetics and the stochastic nature of precipitation and evaporation. We also learned about the "bathtub effect." However, the practice continued with covering of the basins or trenches to reduce the worker dosages from the radiation from the nuclides which had sorbed to the sides of the basins.

Despite claims to the contrary, there was environmental awareness. The Clinch River Study, carried out in 1960-1964, showed that even with practices of that time, no one downstream from the Oak Ridge Reservation was subjected to doses greater than the permissible levels. (39) The objective after that study was to maintain Strontium-90 releases to less than two curies per year. Sadly, this is not true at some sites in the former Soviet Union where greater than permissible doses are received by the downstream population and the releases of Strontium-90 to the river per year range from 20 to 100 curies per year, with a projected increase to as much as 1000 curies per year into the Techa River, which has an average flow one two hundredth that of the Clinch. (40)

CONCLUSIONS

While we have obviously become more sophisticated and knowledgeable since those early days, it has not enabled us to forecast the future any better. Despite that, I will take a flyer on where we will be. For a solution of the high-level waste problem, it is clear that further refining of models to reduce projected doses for time periods up to a million years from the releases from deep geological repositories from their current 10-6 to 10-16 Sr/yr to even lower levels will win no new adherents. If these calculations are robust, then research to establish a repository should be carried out only to the degree necessary to obtain a license. There are still may fundamental scientific questions raised by the repository investigations. Resolution of these questions should compete for funding with other basic research questions.

For low level wastes, where the wastes will decay to innocuous levels in short time periods, less than 300 years, then solidification (vitrification) and engineered containment should be carried out to prove, by extrapolation, that the hazardous substances can be safely kept from the environment for that period of time.

For remediation of the polluted environment, as can already be seen, the funds available will dictate the degree of cleanup. For the most part the risks are not imminent, except to workers. However, there are still unstabilized, potentially mobile wastes, e.g., high-level waste in tanks, that must be immobilized. (41) The degree of cleanup needs to be related to the risk and this will necessitate more attention to future land use. Not mentioned at all in this paper, is perhaps the most important topic of all, i.e., to prevent future contamination of the environment, i.e., pollution prevention. This is, perhaps, a reflection of the lack of emphasis given the topic in the early days till now. All these decisions will be dependent upon public involvement, trust, political will, and good fortune, a tall order.

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25-4

PARADISE LOST

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ABSTRACT

For over 40 years, the country has been pursuing a goal of developing a practical system which would provide for the control and permanent isolation of the quantities of waste generated in the civilian power program. Today, despite the expenditure of hundreds of millions of dollars and the efforts of many skilled scientists, we seem no closer to realizing that goal than we were when the AEC's civilian waste management program began in the early fifties. Reflecting on the history of the development of the AEC's civilian waste management program and the experiences at Lyons, KS it should be quite clear that neither the presentation of data on waste form stability, geologic stability, the verification and validation of environmental transport models or geochemical relationships, bio-chemical interactions or health effects can, over the periods of time in question, be incontrovertibly demonstrated to the satisfaction of serious critics.

We are dealing with a largely political problem which cannot be resolved by purely technical arguments; it is a problem which exists now, which cannot be avoided and one which may be resolved only through the legislative process. If, from a purely political perspective, spent fuel can not be left on the reactor site under the close scrutiny of national, state and local regulators, the DOE must build a monitored, retrievable surface storage facility. Irrespective of the outcome of the Yucca Mt. studies, construction and operation of an RSSF should not be delayed any longer if we wish to keep any semblance of a national nuclear power program alive.

INTRODUCTION

In the early days of the AEC's civilian power program, most papers on the subject of civilian waste management seemed to be introduced with the phrase "as nuclear power assumes an ever increasing role in meeting our nation's energy demand, the quantities of radioactive waste will grow proportionately...." Some growth! In 1970, we were projecting nuclear power production to be on the order of 1000Gwe by the year 2000. There would be, in that year, one reactor brought on line each week and every shipyard in the world would be busy fabricating reactor pressure vessels! Today, we may be on the verge of losing our nuclear power option and, in the near future, we may find ourselves buying from other countries the very technology we developed.

One of the major objections to the increased use of nuclear power in this country has been waste disposal. There is no better example of the politico-technical dilemmas which face our nation in many areas than the issue of radioactive waste disposal. For over 40 years, the country has been pursuing a goal of developing a practical system which would provide for the control and permanent isolation of the quantities of waste generated in the civilian power program. Today, despite the expenditure of hundreds of millions of dollars and the efforts of many skilled scientists, we seem no closer to realizing that goal than we were when the AEC's civilian waste management program began in the early fifties. As early as 1953, in an article in "American Scientist", Dick Hatch of Brookhaven wrote:

"..if the atomic energy program is to grow and become worldwide, as seems highly probable, much will depend on the development and maintenance of a high sense of responsibility to future generations in regard to the disposal of waste from that industry. Meanwhile, if we are to anticipate the need of disposing of large quantities of longer-lived radioactive waste products on a permanent basis for our own benefit, we must face up to a new kind of responsibility, that is, setting up for our own use a system of standards and practices under which this generation would not be the principal beneficiaries."

At the time that article was written, the highly radioactive liquid wastes produced during the recovery of plutonium were being stored in tanks at the AEC's Hanford and Savannah River production sites. Outside of the production groups, the rest of the AEC community had only a limited picture of the scope and magnitude of the waste management practices at these sites. Even then, it was clearly recognized, as Hatch noted in his article, that total reliance the indefinite storage of highly mobile, highly radioactive liquid waste in tanks was not a very good idea.

"..Certainly it would be unimaginative, uneconomic and eventually downright dangerous to continue to pursue the same course in the management of (civilian) waste disposal affairs...Thus it is quite generally recognized that any stored waste should be reduced to the least practical volume, possibly to the solid state..."

By the early 50's, the AEC's civilian waste management research program had already taken shape. Lab-scale studies at Brookhaven, Oak Ridge, Hanford and Idaho were showing great promise in converting liquid waste to stable, solid forms. Perhaps recognizing the potential magnitude of the problem and the need to get advice from the scientific community, the AEC requested the National Academy of Sciences to carry out an examination of the problem. They were to review current research efforts and to make recommendations on the future direction of the AEC's waste

management program. That Committee, made up of some rather eminent personages, met in Princeton NJ in 1955 and arrived at a number of conclusions:

1) Safety is a primary concern, taking precedence over cost and

2) Radioactive wastes should be isolated as permanently as possible from the biosphere

More importantly, the Committee suggested a number of possibilities for future work which included 1) disposal in salt 2) deep-well disposal in permeable formations and 3) conversion of liquid wastes to solids

Supported by these recommendations, work began in earnest to develop improved techniques for converting liquid wastes to stable solids and for examining various methods for their "disposal".

By the early sixties, studies on the fluidized bed solidification of fuel reprocessing waste generated at the Idaho Chemical Processing Plant (ICPP), on the "pot calcination" and rising level glass processes at Oak Ridge, the phosphate glass and rotary kiln processes at Brookhaven and on the spray calcination process at Hanford had all reached the point of successful cold engineering demonstration. Indeed, so successful was the fluid bed calciner project at the ICPP that the pilot plant was converted to a production unit which became fully operational in 1963.

These cold pilot plant engineering studies had clearly demonstrated the feasibility of converting wastes to a variety of stable, solid forms and the advantages of solidification became obvious to all:

- Large reductions in waste volumes
- Rendered waste less soluble and relatively immobile
- Lowered storage costs
- Permitted safe transportation of waste and the development of practical methods of disposal

Accordingly, the AEC decided to take the next logical step in this development effort: a hot pilot plant demonstration. Thus began the Waste Solidification Engineering Prototype (WSEP) demonstration project at PNL. Approved in 1962, this project was designed to demonstrate, using full levels of radioactivity, the solidification of liquid waste with compositions representative of those expected to be produced in several commercial fuel reprocessing flowsheets. Three individual processes were to be tested: the ORNL pot calcination process, the Brookhaven phosphate glass process and the PNL spray solidification process. A follow-on product evaluation was also planned to be carried out to characterize the long-term behavior of the solidified products in simulated storage environments.

Fig. 1

The WSEP program was nothing short of a remarkable success. Highly radioactive strontium and cesium concentrate, graciously donated by the AEC Production Division's B-Plant at Hanford, were shipped over to the WSEP facility to produce synthetic waste solutions with representative levels of self-heating and radiation. Interchangeable modular units for each process were designed, tested, installed and operated without a hitch. Hot operations began in 1966. Fantastic engineering by the boys of Battelle. It should be pointed out that the WSEP program was solid proof-of-principle. WSEP demonstrated the basic feasibility and practicality of several solidification technologies, applicable to a variety of reprocessing flowsheets. With the data from this program, the industry could readily adapt this technology to meet their own specific needs. Remember that, at this time, the industry had some very explicit and

singular responsibilities for demonstrating the safety of their plant designs and processes.

We mentioned `commercial fuel reprocessing'. Now, the story of civilian radioactive waste management cannot be separated from the story of spent fuel reprocessing. In the sixties, you may recall, there was a system that we referred to as the nuclear fuel cycle, wherein the large quantities of unfissioned uranium and the plutonium bred in the fuel during power production would be chemically separated from the highly radioactive fission products and recycled for re-use. That separation process produced a waste stream that used to be called high-level waste. In the AEC/Congressional blueprint for the civilian power program, the reprocessing of spent reactor fuel would be carried out as a commercial operation, since the fuel was, in fact, the property of the individual power reactor operators. The unfissioned uranium and plutonium would be sold back to the AEC. Remember the $10/gm plutonium buy-back policy? Based on the power growth curve, the quantities of spent fuel discharged from the civilian reactors were expected to increase dramatically. By 1966, the AEC was in the process of issuing an operating license for the first commercial fuel processing facility, the 1MT/da. Nuclear Fuel Services (NFS) facility at West Valley, N.Y. That facility design was based on the successful PUREX reprocessing plant which has been in operation at Hanford for some time; like PUREX, that facility had provisions for tank storage of liquid waste. At that time also, GE had submitted its application for their 1 MT/da Midwest Fuel Recovery Plant (MFRP) adjacent to the Dresden Nuclear Power Station. What later became the Allied-Gulf Nuclear Services corporation was also involved in discussions with the licensing arm of the AEC on their plans for the construction of a 5 MT/da plant adjacent to the AEC's Savannah River plant at Barnwell, S.C. Such a high throughput facility, it was believed, would lead to dramatically lower processing costs and thus would AGNS could attract substantial business from the utilities since lower fuel cycle costs would, at that time, increase the utilities profitability. The Atlantic Richfield Hanford Company (ARCHO) also expressed an interest in getting into this business as did National Lead, purveyor of fuel shipping casks. So the commercial fuel reprocessing business was shaping up as expected.

Fig. 2

Meanwhile, back on the waste disposal front, the success of the waste solidification program now allowed for the real possibility of disposing of high level waste at sites other than at the site where the fuel was reprocessed. This made the concept of storage in bedded salt (suggested by the NAS) feasible. Project Salt Vault, which was carried out by the ORNL during the period from 1963 to 1967, followed up on some earlier experiments on the storage of simulated liquid and solidified wastes in an abandoned salt mine at Lyons Kansas. Project Salt Vault was an engineering demonstration of techniques for the handling and emplacement of wastes in bedded salt and for determining the response of salt to the effects of heat and radiation. Fully radioactive spent fuel elements from the MTR (or was it the ETR?) at Idaho were buried in an abandoned mine in Lyons Kansas in 1963 and, over a period of three years, data was obtained on the behavior of salt under such adverse environmental conditions. Following completion of the project, the fuel was successfully removed from the mine and sent safely back to Idaho. Salt Vault clearly demonstrated that there were no serious adverse effects on the salt from

heat and radiation, that conventional mining techniques were adequate for dealing with structural problems in a disposal facility and that wastes could be transferred to and handled safely in a underground environment. The success of the waste solidification program and the results of the Project Salt Vault demonstration provided encouragement that a solution to the civilian high-level waste disposal problem was well in hand.

Again, the AEC looked to the NAS for approbation of its overall waste management program. Another NAS committee was formed, reviewing the progress made in solidification and "disposal" since the last study. That Committee issued its report in 1966 and noted, among other things:

"The containment of fission products in an inert solid is still the manner of waste disposal the Committee most favors"

and:

" Continuation of the studies toward ultimate disposal of high level solid waste in salt was recommended by the Committee...the use of caverns in salt beds as permanent storage sites for high level radioactive solids has promise of being successful and satisfactory".

That report came close on the heels of another study carried out in 1964 by the American Association of Petroleum Geologists which identified, among other sites, the salt beds of Kansas as a promising site for waste disposal.

While all this was going on, the Commission was reviewing developments in the licensing of commercial fuel processing plants. As noted, NFS, GE, AGNS, ARCHO and National Lead were all, in some fashion, getting heavily involved in the fuel processing business. At one Commission briefing by the staff, Commissioner Jim Ramey asked, not altogether rhetorically, whether the Commission, as a matter of policy, should continue to permit the random siting of fuel processing plants with their related waste storage facilities or whether these plants should be restricted to pre-defined sites, more suitable from the waste storage perspective. This question may have been triggered by a comment in the 1966 NAS report that no current Commission site was acceptable from the standpoint of long term waste storage.

This humble but incisive question begat the monumental Fuel Reprocessing Plant Siting Study. That study, which focused exclusively on the civilian sector, considered nuclear power growth, the reactor mix (as defined by the Systems Analysis Task Force, SATF, study for the year 2000), fuel reprocessing technology, the status of solidification and waste disposal technology. The results of that study, supported by the magnificent ORNL-4451 "The Siting of Fuel Reprocessing Plants and Related Waste Management Facilities" (to which many, many experts contributed), led the Commission to a number of conclusions, among which was the belief that fuel reprocessing plants could be located in most places provided the site was not going to be used as a site for a permanent waste disposal and that, importantly, disposal of the high level reprocessing waste should not be permitted on privately owned land with federal (rather than state) ownership indicated.

This study provided the bases for the Commission approving the issuance of the landmark Appendix F to 10CFR50 in August, 1970 which established, for the first time, the responsibility of the Federal Government to provide for the long-term management of high level waste generated in this country. That policy stated that:

> reprocessing plants may be located on privately owned land

high level liquid waste inventories must be limited in volume to the quantity produced in the prior five years

all high level wastes must be transferred in AEC-approved dry, stable, solid form to a federal repository no later than ten years following separation of fission products from irradiated fuel

the Federal government will assume permanent custody for these wastes when it accepts the wastes and a single payment from industry which would cover all costs of disposal

repositories will be on land owned and controlled by the federal government

a design objective for reprocessing plants shall be to facilitate decontamination and removal of wastes from temporary storage at the time of decommissioning

Parenthetically, there were some state officials who took umbrage at Federal preemption of this activity. Today, that's hard to believe but in fact, some officials in Louisiana, South Carolina and New Mexico were all, at one time or another, interested in playing host to the federal repository.

Today, we hear much about acceptance criteria for the form of the waste and the long term performance of the multiple barriers designed to prevent release to the environment. Interestingly, it may be recalled that in formulating this policy, the question of the acceptability of the waste form was given every consideration by the AEC staff since such] criteria would have a direct impact on fuel reprocessing plant designs. Given interment in a geologic repository, AEC staff believed, and the Commission agreed, that the form of the waste was of importance only during the period of interim storage, transportation and emplacement in the repository. The staff held to the belief, correctly I think, that the very long term integrity of either the waste form or its packaging could never be convincingly demonstrated over the periods of time involved. Ultimate reliance was therefore being placed on the stability and long term integrity of the geologic disposal environment to isolate the waste from man's biosphere. Here, the geologic record would, we believed, provide a more convincing and demonstrable argument for assurance of long-term isolation.

At that point in time, everything seemed to be coalescing. The technologies of reprocessing, waste solidification and disposal were coming together and we saw that the nuclear fuel cycle could, and would, we thought, be closed in a way envisioned by the pioneers in the field. After numerous endorsements by the NAS, bedded salt was clearly the disposal environment of choice for the federal repository and, at this point, the Commission felt confident it could proceed with the systematic and disciplined development of the first Federal Waste Repository.

Then we had the fire at the Rocky Flats weapons plant. Without going into details, the Rocky fire produced some rather large volumes of plutonium contaminated wastes which could not, as a practical matter, be stored on the Rocky site. Plans were made to ship these wastes to the NRTS in Idaho. Frank Church, who was then Senator from Idaho, objected and asked the Commission in effect `why Idaho? Why, if these wastes couldn't be stored safely in Colorado, would it be acceptable to store them in Idaho?' The Commission clearly needed a solution to this dilemma. One of the Assistant General Managers asked his program managers whether the salt repository could accept plutonium wastes. When the answer was "yes", the Commission committed to removing the plutonium wastes from Idaho by

some definite future date (20yrs) and put the repository project on the fast track; in hindsight, perhaps too fast.

In 1971, ORNL was charged with developing criteria which would allow us to evaluate potential salt disposal sites and to proceed with the development of a conceptual design and cost estimate for such a repository. That information would be used to support a FY 1972 Congressional budget request for construction of the repository. Based on the criteria developed by ORNL and our knowledge about Kansas geology, the Lyons site was selected. Preparation of the required Environmental Impact Statement was begun. It was anticipated that, if funding was received, the Lyons facility could become operational sometime in 1974, barring unforeseen problems. Incidentally, the estimated costs for the Lyons facility was on the order of $25-30 million. It must be noted that the Commission staff considered the selection of Lyons as tentative. There were clearly many questions that we, and our USGS consultants, felt needed to be answered before we started digging. Lyons was, however, a good bet for many reasons. There was, in the abandoned portion of the old Lyons mine, a place for the disposal of the mined salt as well as facilities for marketing what we didn't put back in the ground. Lyons was located on three railroads, the salt had been stable for several millions of years and its thickness was more than adequate for heat dissipation. Because of their experience with Project Salt Vault, there was, at the local level, general public acceptance of a repository.

Fig. 3

Again, the Commission went back to the NAS to solicit their views on this tentative, but monumental decision. NAS reviewed the plans for the Lyons repository project and gave its blessing to the site subject to the completion of certain additional studies which it felt were necessary to provide assurance of long term geologic and hydrologic stability.

During October of 1970 amid much fanfare, briefings on the AEC plans were provided to state and local officials as well as representatives of the local scientific community . Little did we realize what was in store. New technical issues requiring resolution were raised (e.g., the Wigner effect: energy storage in crystals). There were almost blackmail-like demands from various researchers and other interested parties for AEC support to carry out a variety of geologic and environmental studies in the area. Following publication of the Lyons Environmental Impact Statement and the receipt of many comments, Congressional hearings were held in March of 1971. At those hearings, testimony and serious opposition in principle to the project was expressed formally for the first time. A member of the U.S. House of Representatives, Joe Skubitz by name, was the first, I believe, to raise the NIMBY question at Joint Committee hearing on Lyons. Congressman Skubitz said to us, in effect, he didn't care how safe the facility was, it wasn't going to be in Kansas! He was, as I recall, supported by Senator John Pastore who, years earlier, had objected to federal plans to build a Hoof and Mouth Disease laboratory in his state of Rhode Island.

Subsequently, the anti-Lyons fervor rose and the repository became a hot political issue. Many technical issues, real and imagined were raised. Ultimately, as a result of many meetings with interested and affected parties, the many technical issues and red-herrings raised during the EIS review process had been boiled down to a few reasonable questions upon which all parties agreed: 1) the ability to find and plug boreholes left over from earlier oil exploration activities in the area, 2) the ability

to quantify the rock mechanics properties of overlying formations (which protected the salt from dissolution) to ensure that, under conditions of repository operation, their integrity would be maintained and 3) that retrievability of wastes be demonstrated and maintained until some period of confirmation had been completed. A proposed plan was developed to provide this information jointly by the US Geological Survey, the Kansas Geological Survey and the ORNL. These studies followed closely the NAS recommendations.

Subsequent to, or almost concurrent with, the 1971 Congressional hearings, rather serious questions on site suitability were raised following the disclosure of solution mining activities in a salt mine adjacent to the Lyons mine. These questions ultimately led to the abandonment of the Lyons project, even though the Kansas Geologic Survey indicated its support for a disposal project in the Kansas salt beds at a site other than Lyons. We were now on the run. Then-AEC Chairman Schlesinger wanted to withhold announcement of the decision to leave Lyons until Democratic Governor Docking could be replaced by a Republican in the election of 1972, This would clearly show the clout a Republican governor had with the White House. I don't believe the AEC ever openly acknowledged abandoning Lyons, they sort of walked away from it, focusing attention elsewhere. Although exploratory work was initiated at a few other sites in Kansas, ultimately, none of these proved to be as good as Lyons.

The Commission management was finally beginning to see the need to develop contingency plans. The USGS had earlier identified a small area in New Mexico which appeared to have many features which would make it a suitable repository site. The Commission decided to follow up on this suggestion and do some exploratory work while at the same time, identifying other bedded salt sites for further examination.

Sensing the obvious weaknesses in the Commission's waste program, the flood gates were opened for new suggestions and it seemed everyone and his brother joined what became a feeding frenzy. A new cast of characters entered onto the waste management stage, the Joint Committee was neutered, the NRC was split off from the Commission, the residue became ERDA and nuclear power relegated to a lesser role in the national energy plan. There was renewed interest by these newcomers in old suggestions for disposal,e.g., space disposal, disposal in antarctic icecaps, tossing wastes into the subduction zone of drifting continental plates, disposal of wastes into cavities specially created by nuclear weapons, transmutation, sealed casks at Hanford, seabed disposal. Much time and not an insignificant amount of money was spent chasing these old ideas and, aside from the sealed cask concept, none went anywhere. I often wonder how much money was spent on these pipe-dreams. Certainly, their pursuit resulted in deferring needed decisions on the development of more practical waste management solutions based on our Lyons experiences.

Because of the favorable initial findings at the New Mexico site and reflecting on the continued problems of public acceptance in Kansas (which appeared to be due, in large measure, to the precipitous action on our part), the Commission made a decision in December of 1971 to proceed with the development of a "Bedded Salt Pilot Plant" wherein all civilian wastes stored would be easily retrievable by proven means and in which an experimental program to demonstrate the analytical techniques used to assure safety could be carried out concurrently. Later, that facility would be expanded to accept plutonium wastes which were being safely

stored in "20-year retrievable" storage facilities in Idaho. In 1973, a `Governor's Committee' was formed in New Mexico to ensure competent, objective evaluations of the complex technical problems involved and to preclude the political grandstanding and research blackmail evident in the Kansas exercise.

Since the delay in the development of a full scale repository would require extended storage of waste at the fuel reprocessing plants, intervenors were then beginning to use the absence of a repository as a reason for blocking the construction of nuclear power plants. For this reason, the Commission, in 1972, made a decision to proceed with plans to store high-level reprocessing plant waste in a retrievable surface storage facility (RSSF) for a period not to exceed 100 years, thus allowing it to meet its commitment to the nuclear industry to accept 10 year old solidified fuel reprocessing waste. This activity would be carried out concurrently with the BSPP development and other work to identify and evaluate alternate repository sites.

Once again, the NAS was asked to form a panel to comment on the plans for the interim storage of the high-level reprocessing plant wastes. That Committee issued a report in 1975 which stated, among other things, the following:

1) retrievable surface storage is an acceptable interim stage in a comprehensive system for managing high level radioactive wastes

2) the Panel recommended an optimized version of the sealed storage cask concept

3) recommended a number of additional research tasks which needed to be carried out, including optimization of the form and containerization of the waste (providing for multiple barriers)

4) precautions should be taken in siting and design of a sealed storage cask facility to minimize the risks of earthquakes, tornados, "sabotage, act of war, accidental air crash, etc"

5) the facility should be made unattractive to plants and animals

6) noted that retrievable surface storage is not a substitute for ultimate disposal and recommended an immediate and comprehensive assessment of sites in "salt, shale, or other suitable geologic media" for the disposal of high-level waste followed by a demonstration repository at the earliest possible time, and

7) increased research on processes for economically separating and disposing of, or using, long-lived actinides.

Whatever happened to the RSSF is not altogether clear to me. It seems to have evaporated. The NRC staff even went as far as developing regulatory guides for licensing as "ISFSI" (independent spent fuel storage installation), but then nothing happened. Somewhere along the line, I think the Bedded Salt Pilot Plant turned into the Waste Isolation Pilot Plant (WIPP) designed solely for the receipt of low activity plutonium waste, a facility which has yet to accept any real waste.

In retrospect, we have, since the mid-seventies, created an Office of Nuclear Waste Isolation, a Center for Nuclear Waste Regulatory Analysis; we have asked the American Physical Society and National Academy of Sciences to carry out thorough and independent reviews of the nuclear waste program; we have prepared plans, schedules, PERT charts, established time limits, issued extensive regulations, conducted site examinations (Hanford, Deaf Smith e.g.) and held "waste confidence" hearings. We have revisited already demonstrated waste solidification technology. Hundreds, if not thousands of millions of dollars have been

spent on testing, model development, lab experiments, field work on natural analogs, all of which seem to have helped the decision-makers not at all. We are perhaps worse off than we were in 1972 since the public and the Congress have been witness to paralysis in the waste management area. In its frustration, the Congress has enacted various pieces of legislation to try to move the process along. While well intentioned, some elements of the legislation were clearly misguided. Perhaps the most visible example were some elements of the Waste Management Act of 1982. This act not only provided a golden trough (through mandatory utility contributions to a waste management superfund) at which all waste management practitioners could feed but also provided essentially veto power to a wide variety of diverse interests: states, localities, Indian tribes, all of whom, while acting with the purest of motives, could effectively block any attempt to establish any sort of waste management facility anywhere.

As we know, the Carter administration killed the fuel cycle as we once knew it in the interests of non-proliferation. We no longer have the problem of disposing of wastes from the reprocessing of civilian reactor fuels. Now we have spent fuel piling up at reactor sites around the country. Whether solely to continue the feeding frenzy at the trough of the waste fund or for other reasons, spent fuel is now considered as waste, despite the tremendous economic value of the resources contained therein.

Many of those who labored so brilliantly in the early days of the waste management program feel this whole situation is regrettable, to say the least. Fifty years should have been enough time to bring this matter to some form of closure. No one in authority seems to be able to make a decision, probably because any decision in this area is considered potentially damaging politically. It is almost embarrassing to look at the progress of the European nuclear programs, all developed as a result of pioneering work done in the U.S.: Granite disposal in Sweden, salt disposal in Germany, glassification in Belgium and France. Why we, in this country, have not proceeded more expeditiously is truly puzzling to the international community.

The lessons of Lyons and subsequent happenings should be clear if anyone chooses to look back. Years ago, Alvin Weinberg wrote an article titled "Science and Transcience" in which he identified part of our current dilemma. We are still trying to make political decisions with technical information. We wrap a political issue in a white lab coat and say "look at this data, the answer is crystal clear". The data we have developed o'er these many years doesn't seem to convince anyone who feels potentially adversely impacted; our pure scientists will never be convinced we can, with acceptable levels of uncertainty, predict very far into the future. So we, like Sisyphus, may be doomed to pushing that rock up the hill, only to have it roll back to the bottom each time we get near the top.

As a recent example, the NAS has again reviewed the waste disposal problem from the standpoint of our regulatory posture and has come up with some recommendations, among which, if I understand it correctly, is that the time frame over which we need to demonstrate waste isolation should be during the period of maximum hazard and this examination should consider a period on the order of 1 million years; further, that during the period of maximum hazard the radiation dose to a maximally exposed individual should be on the order of 15 mrem. That study has, it would

appear, raised some other virtually intractable issues (such as future population distribution, land use, maintenance of human control) which, in combination, can only serve to further delay the decisions that need to be made.

If history is any teacher, it should be quite clear that neither data on waste form stability, geologic stability, the verification and validation of environmental transport models or geochemical relationships, bio-chemical interactions or health effects can, over the periods of time in question, be incontrovertibly demonstrated to the satisfaction of serious critics. One needs to question the continued pursuit of these ghosts. Interesting science but of little help to those who need to make the decisions and those who have waste which requires "disposal".

It may be that the only way out of this maze may be for the Congress or the courts, to force the issue through legislation or court order, forcing implementation of some plan, any plan, thereby taking the political heat off those responsible. I would note that a decision in the current court fight between the DOE and a group of utilities on DOE's contractual obligations to take responsibility for spent fuel by January, 1998 may do more to get this program back on track than anything our poor scientists can do.

It is clearly time for the country to get on with the job of moving fuel away from reactor sites where it was never meant to be for very long. But what to do with it? We all know really! If, from a purely political perspective, spent fuel can not be left on the reactor site under the close scrutiny of national, state and local regulators, the DOE must build a monitored, retrievable surface storage facility. The next question is "where". The most likely prospect is at the Nevada Test Site. Since weapons testing is in a period of suspension, the impact of an RSSF should not be a problem from the standpoint of our national security. Since retrievability is assured, public acceptance may be easier to secure. Moreover, it is unlikely that that site, given its past history, could be used for other purposes. Assuring the long term protection of that site from human intrusion is not a rhetorical question to be endlessly debated. At that site, among others, long term site control is a fact of life which can not be avoided. There are other sites which may be just as acceptable: Hanford for one, the INEL for another. With the cancellation of the new production reactor and potential reductions in almost all of DOE's facilities and operations, the presence of an interim repository for spent fuel might prove quite attractive from the point of view of the local economy.

While the RSSF is being built and operated, efforts could continue at a more leisurely pace (if that's possible) on resolving the transcientific problem of how to demonstrate long-term isolation. The issue seems to be one of HOW to demonstrate, not the demonstration of a specific site itself. With regard to current attempts to demonstrate the acceptability of Yucca Mt., one ought to review the 1966 NAS reports and the EPA's comments on the bedrock storage concept at Savannah River to see what may lie in store for the unsuspecting at a public hearing.

If construction of one or more RSSF's is to be pursued, one of the major considerations that must be dealt with in terms of public acceptance is the issue of transportation. The Lyons exercise demonstrated that this topic is not, in the eyes of the public, an insignificant issue. The numbers of shipments, the condition of rights-of-way, container testing, inspections, notification of routings and emergency action plans are all

elements of a system that must be developed and documented before an RSSF can get off the ground.

In an altogether different vein, I have often wondered if this waste management problem might be viewed from another perspective. It would seem that, other than heavy metal poisoning, the only demonstrable hazard presented by either spent fuel or high level reprocessing waste is the induction of some form of cancer, production of genetic abnormalities, or irreversible damage to sensitive biota. Ought we not, once again, go back to the NAS and ask that they conduct a study or survey to estimate the likelihood of developing a cure for cancers and genetic abnormalities over the next 100, 200, or 500 yrs. Such a survey might prove enlightening in terms of helping us view the problems of radioactive waste storage in a more balanced light. If the NAS really believes that it is possible to predict, with acceptable levels of uncertainty, events 1,000,000 years into the future, or to confidently assess radiological doses over comparable periods, it should certainly be possible, with even greater confidence, to project when a cancer cure will be developed or when genetic engineering will advance to the point that genetic abnormalities can be reversed. From that perspective, the waste management problem might degenerate to a 1000 year problem, one for which there is a much more reliable basis for predicting future impacts and thus a more solid base upon which to secure public acceptance. That acceptance is, in the final analysis, the key ingredient in this stew.

To summarize, the lesson of Lyons and subsequent events suggest that we are dealing with a largely political problem which cannot be resolved by purely technical arguments; it is a problem which exists now, which cannot be avoided and which must be addressed through the legislative process. The Congress came very close to legislating what would seem a prudent solution: i.e., to mandate the construction of an RSSF. Surprisingly, last October the current administration came out against such a proposal. However, the Congress and the courts may yet force this issue. While a decision to continue work on the Yucca Mt. repository can be debated, it seems unlikely that, based on past experience, the results of this work could ever develop the sort of incontrovertible proof of long term integrity that the various opponents seems to require. In any event, the development of an RSSF can not and should not await the outcome of the Yucca Mt. studies.

25-5

A HISTORICAL LOOK AT WASTE MANAGEMENT IN SELECTED PARTS OF THE NUCLEAR FUEL CYCLE--THE PRACTICES, DECISION POINTS AND THEIR EFFECTS

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ABSTRACT

A historical look at the waste management practices in three segments of the nuclear fuel cycle--uranium ore mining and milling, reprocessing of spent fuel from power reactors and high level waste handling, and disposal methods is presented. The technical and political decisions that affected these practices are also presented. The initial practices of the mining and milling segment came from the mining industry. These practices were changed because of studies showing adverse health effects leading to public pressure to change or through legislation. Reprocessing technology was based on technology developed under U. S. government defense programs. While safely containing the reprocessing waste streams,

improvements have been made to reduce the volume of waste generated and reduce plant operating costs and personnel exposures. The decision of the U. S. not to reprocess spent fuel while many other countries continue to do so has led to quite different approaches to high level waste handling and disposal between these nations. The U. S. is trying to find a permanent repository for storage of the spent fuel while the other countries are reprocessing and thus significantly reducing the volume of waste, vitrifying the waste and temporarily storing it for up to 50 years. At the same time these countries are studying the best underground geological formations for a permanent repository. Which approach will lead to the most economical and best solution to problem is yet to be determined.

INTRODUCTION

This paper will present a historical review of the practices, significant decision points and the effects of these decisions on selected parts of the nuclear fuel cycle. The elements of the fuel cycle presented will be uranium ore mining and milling, reprocessing of spent fuel from commercial nuclear reactors, and high level waste handling and disposal. Eight segments constitute the nuclear fuel cycle. The three selected segments represent the beginning, one of the intermediate segments, and one of final elements of the cycle. While I have worked in all parts of the fuel cycle and there is value in examining the entire fuel cycle, time restraints and my much greater personal involvement in the selected segments led to restricting the paper to the above subjects.

The objective of the paper is to preserve some of the history and the lessons learned so we can continue to improve and hopefully expand the use of nuclear energy.For each of the fuel cycle segments, the paper will describe the process and the waste streams generated, give some background on their history, state any decision points that caused a change in practices, along with some of the factors that influenced the decision, reflect on the actual effects of the decision, and if applicable, describe what may have happened if alternatives under consideration at the time had been chosen.

URANIUM MINING AND MILLING

The Atomic Energy Act of 1947 authorized the Atomic Energy Commission (AEC) to purchase uranium ores and concentrates. The carnotite and roscolite ores of the Colorado Plateau had previously been mined and concentrated for the recovery of radium, uranium and vanadium. The radium was used for painting watch dials so they would be visible in the dark and for some limited use in medicine. The vanadium was used to increase the strength of steel, and the uranium was used as a coloring agent in a glaze on crockery. Most of material for the Manhattan Engineering Project came from the Belgium Congo's very high grade pitchblende deposits assaying 30-70% U_3O_8. Some came from the Vanadium Corporation of America's government funded mill at Monticello, Utah, which produced a uranium- vanadium sludge. These Colorado Plateau ores were much lower grade, normally 0.2 to 0.3% U_3O_8 (1). After World War II, the U. S. almost immediately entered the cold war with the USSR and wanted to develop an indigenous supply of uranium; therefore, the U. S. began an aggressive effort under the AEC's Division of Raw Materials to explore for uranium , develop mines , build uranium processing mills, and encourage companies to do likewise. The AEC reopened and operated the war time built uranium-vanadium mill at Monticello, Utah from 1948 until 1960 (1). Its purpose was to produce needed products but was also used to

determine actual milling costs. This information was used in contract negotiations with private companies such as Union Carbide, Vanadium Corporation of America and later with Anaconda and Kerr McGee to buy their production. The practices adopted in this new industry were those of the mining industry. Some had been in use since the gold rush days of late 1870's. Although a government agency, the AEC's Division of Raw Materials personnel were largely from the mining industry so their technology and their management policies were akin to the mining industry rather than other parts of AEC or the Government in general.

Fig. 1

Figure 1 shows the main elements of mining and milling and the various waste streams produced. Many of the early mines were small underground mines. Most were dry mines, so there was not the problem of acidic water discharges that are prevalent in the eastern U.S. coal mining regions. Relatively little of the material removed from this type of mine was dumped at the site. By the late 1960s, 70 percent of uranium production came from underground mines that were larger in size and produced more waste rock. Some of these mines contained water (3). In wet mines, mining costs were about 30 percent higher (3). At about the same time, the much larger open pit mining operations, ranging in depth up to 400 feet, were begun in New Mexico and Wyoming. Large amounts of overburden and waste mixed with ore were removed (15-150 percent more) and left at the mine site. Some of this was voluntarily returned to old open pits but most was not. Still,because of the arid conditions in the western U. S., these open pit mines did not pose the same ecological problems of erosion and acid water pollution that similar mining practices caused in Appalachia. The other major operating problem for the underground mines was providing adequate ventilation. Inadequate ventilation could adversely affect personnel in the form of radon and dust exposures, which could result in silicosis to the lungs of the miners.

The milling operation required dust collection equipment to contain the dust from crushing the ore. This was particularly true of the limestone ores that fed the Anaconda plant. The dust from these systems was captured and in most cases injected back into the system to recover any values. In the early 1950s, five of seven plants recovered vanadium and uranium. This was done by mixing the ore with 7-15 percent salt (NaCl) in a hearth roaster at a temperatures of ~850F. The fumes and any dust rising from the operation were passed through scrubbers and/or a "bag-house" of filters before exiting up a stack. This operation improved over the years. The roasted product was then dissolved in sulfuric acid (H_2SO_4) by most of the mills; but a few mills used sodium carbonate (Na_2CO_3) as a dissolvent. Ores not containing vanadium could be dissolved directly. The mills separated the solids from the pregnant liquors by thickeners and filtration. Later the mills added a step to increase the concentration of the uranium and remove more impurities by using either solvent extraction or ion exchange processes to recover the uranium (4). Two of the primary waste streams from the milling operation were: 1) A fine material left after the uranium and vanadium were extracted, called tailings. Tailings were usually impounded and stacked in tailing piles; and 2) waste liquors streams, which consist of contained liquors in the tailings and other plant discharges. The chemical composition of these liquors was dependent on the treatment rendered and could be either acidic or basic in nature and could contain varying quantities of H_2SO_4, $NaClO_3$, NH_4SO_4, and small amounts of cyanide.

Significant Decisions

There have been several significant decisions that have effected the waste management practices of uranium mining and milling industry. These were: 1) a series of laws and regulations which forced further treatment of the waste and effluents from the mines and plants; and 2) public pressure to change practices.

These laws included the establishment of the Environmental Protection Agency (EPA) in 1970 but more importantly, the Resource Conservation and Recovery Act (RCRA) of 1976 and the Comprehensive Environmental Response Compensation and Liability Act (CERCLA) in 1980 which implemented hazardous and waste management practices in this and other industries. The Uranium Mill Tailings Radiation Control Act (UMTRA) in 1978 was specifically directed to the industry. These laws plus the Clean Air Act of 1970 and the Clean Water Act of 1972 imposed further restrictions (5). Examples of these effects are as follows: Efforts were made to impound the liquids and allow them to evaporate in the tailings ponds. However, the ponds would eventually dry out. These were the same methods used in other parts of the industry and produce the same results. When the piles dried out, the winds would blow the fine tailings as dust clouds over the surrounding landscapes. This was not considered a problem and only if the tailings were near populated areas, (i.e.,the old tailings pond from gold milling operations at Colorado Springs, Colorado) was it even considered a nuisance. For many years nothing was done about it but problems arose in the uranium industry when the residents around Grand Junction began using the tailings from the Climax Uranium Mill. The first use was for mulch and fertilizer for rose beds and gardens. The retained chemicals made things grow wonderfully. It wasn't until they decided that tailings would also make good building material and cement and concrete blocks that were used to build houses, buildings, sidewalks etc. that problems developed. Radiation levels began to climb from this use and under UMTRA and CERCLA legislation a massive program to remove the material and reconstruct the buildings was instituted. The program also included the remediation of the milling sites and contaminated private properties in the vicinity. These programs returned the overburden back into the mines, moved certain tailings to more remote locations, placed multilayer coverings over the tailings piles, and treated any aqueous waste residues or soils, not only at the mill site, but in private properties surrounding the mills (5).

Liquid Waste Streams

In regard to the liquid waste streams, the State of Colorado accused Union Carbide of polluting the Colorado River from their Rifle, Colorado Plant. However, Carbide won the argument when they showed that the river was cleaner after it left the plant than upstream. It seems that the hot springs at Glenwood Springs about 10 miles upstream had considerable radioactivity. Rifle Creek, a pure mountain stream, also ran through the plant and discharged into the Colorado River. Dilution was sufficient to mask any discharges from the plant and even lowered the initial radioactivity content of the Colorado River

In the Grants, New Mexico area the results were different when the uranium mills decided to inject their waste solutions into deep wells in the desert near the mills. This was practiced for several years but then the Indians living on the nearby desert detected these chemicals in their well water. The practice was halted because of strong public pressure.

REPROCESSING OF SPENT NUCLEAR FUEL

This section addresses the reprocessing of spent fuel from nuclear power plants and the disposition of waste streams it generates. This will include the early reprocessing at Nuclear Fuels Services West Valley plant, the only plant in the U. S. to actually operate in a commercial mode (6,7). It will also discuss the efforts of General Electric at Morris, Illinois (8,9) and Allied General Nuclear Service at Barnwell, South Carolina (10) to operate their plants. Both plants were built and began pretesting, but neither were actually ever in production mode. Exxon Nuclear at Oak Ridge, Tennessee (11), only got to the detailed design stage before it was stopped. The paper will also discuss practices outside the U. S., where countries such as France, the UK, and Japan consider reprocessing an integral part of the fuel cycle. In many cases the technology is the same as reprocessing fuel from the defense program reactors. However, each industry faced some unique problems. I refer readers to Mr. William McVey's WM-96 paper "Evolution of Spent Nuclear Fuel Technology and High Level Waste Management" for a discussion of the technology in the defense plants.

Fig. 2

Figure 2 shows the main elements of reprocessing and the waste streams it generates. These waste streams have been numbered for easy identification.

The reprocessing of light water reactor (LWR) spent fuels throughout the world has the same basic flowsheet. However, there are a number of gas cooled reactors, particularly in the UK for which the head end of the process must be altered. The spent fuel from the reactors is generally placed in fuel storage pools. The pools are filled with very high purity water. The French at their UP-3 plant have now introduced a new dry storage facility which does not require the extensive water treatment required in pond storage and eliminates waste stream-1 (WS-1). The removal of the fuel element-end pieces generates WS-2. These pieces are compacted and packaged. Shearing the fuel bundles into 1.27 cm to 7.62 cm (0.5" to 3.0") segments produces WS-3 from the release of off gases from the fuel elements plenum section and dust fragments which are captured in a dust collection system. The ceramic material containing uranium, plutonium and fission products in the sheared pieces are dissolved away with nitric acid from the zircaloy or stainless steel metal casing. The radioactive metal hulls, WS-4, are washed and placed in containers. The resultant nitrate solutions are clarified and the solids, WS-5, are combined with other streams and treated. The uranium, plutonium and americium, if present, are separated from the fission products using a solvent, tributlyl phosphate (TBP) in a diluent (i.e. kerosene) in a process called PUREX. These fission products become WS-6, a high-level waste (HLW). NFS only evaporated, sampled, and stored this WS-6 (HLW) as a solution. The other U. S. mills had plans to vitrify these waste. The French and the UK vitrify the HLW and Belgium has successfully vitrified HLW from Germany. The second stage of the PUREX process then separates the uranium from the plutonium. These products are then solidified as oxides. The second stage of solvent extraction and the product purification and solidification steps waste streams produces two waste streams that have always been processed to recover the nitric acid and clean the solvent. WS-7 resulted from acid recycle. This system uses evaporation and acid fractionating towers to recover the acid and remove the contaminants. These processes are primarily to reduce waste volumes rather than recover the monetary value of the acid. These acid wastes

require neutralization. It was common practice in these early reprocessing plants to use ferrous sulfamate as a plutonium reduction agent. However, this created a large amount of waste when it was neutralized. As a result, current plants use combinations of N_2O_4, hydrazine and hydroxyl ammonium nitrate (HAN) for this purpose which significantly reduces the waste. The Barnwell plant planned to use an electropulse column for this reduction (10). WS-8 is created from the solvent recovery operation. This process involves removal of heavy metal and fission products in chemical complexes and solvent degradation products. The cleanup consists of alternate carbonate and acid washes with the waste products either being sent to an intermediate level waste storage or to rework. NFS stored most of these waste as liquids but the later plants concentrated and solidified these waste.

These operations also generate off-gases. The usual practice was to treat vessel off gases separate from the dissolver off-gases. The off-gas from the plants contain iodine, krypton, and tritium. From the earliest days, processes were in place to contain and remove the iodine through a series of iodine strippers, absorbers and passage through silver zeolite beds. These off-gas systems also employed high efficiency particulate filters (HEPA) before the off-gas exited through a stack. The replacement of HEPA filters add to the HLW waste volume. The French and UK plants also recover krypton. Although considerable R&D has been performed and pilot operations conducted, no reprocessing plant is recovering the small quantities of tritium in these off-gases. Failed equipment will be another increasing waste stream as plants age.

Significant Decisions

The first significant decision effecting reprocessing occurred in 1973 when the AEC's Regulatory Division [now the Nuclear Regulatory Commission (NRC)] banned shipments of liquid plutonium. This caused U. S. plants to include plutonium solidification circuits to their flowsheets. The imposition of much stricter seismic criteria by this same regulatory body led NFS to abandoned plans to enlarge their plant. Since they believed their 1 MTU/day plant could not compete with the larger plants being built at Barnwell and being designed by Exxon at Oak Ridge, the existing plant was closed. General Electric also ceased trying to operate the Midwest Plant in Illinois. However, their reasons were due to technical difficulties. The plant's different flowsheet, using a uranium denitration-fluorination purification circuit (9), did not scale up from pilot runs. This coupled with difficulties in maintaining certain equipment led them to close the plant before it got into production. The most significant decision was President Carter's declaration of a moratorium on reprocessing in the U. S. and to simply store the spent fuel. This was done because of the fear of nuclear proliferation and to set an example that we hoped other countries would follow. The Carter decision forced the cessation of operation of Barnwell and Exxon to stop all work at the Oak Ridge facility. At about the same time, the NRC stopped any further consideration of a plan to utilize mixed uranium and plutonium oxides in either breeders or LWRs. This decision led to intense studies of whether some modification of the fuel cycle would have greater proliferation protection than the conventional PUREX process. Both international and domestic teams made these studies and the results showed that there were just marginal differences in the proliferation risks of the processes. None of the foreign countries followed our lead except, several years later, the West Germans because of strong pressure

from their "Green Party" abandoned plans to build the Wackersdorf Reprocessing Plant. President Reagan lifted the U. S. moratorium in 1981, stating that the policy was to support private investment in reprocessing of LWR spent fuel by removing unnecessary regulations and creating stable long-term conditions under which commercial reprocessing would be viable. As a result, interest was shown by Bechtel in buying and operating the Barnwell plant. However, because of decreased demand for uranium and no market for plutonium, the sale did not proceed. Meanwhile, the French have built and are operating major additions to their facilities at La Hague (UP-3 & UP-2-800) and the UK has completed and are operating the THORP Plant at Sellafield. The Japanese are building the JNFL plant at Rokkasho-Mura, and the Russians, other Eastern Bloc countries and India continue reprocessing. These countries lack the natural resources the U. S. has and as result of the 1974 oil embargo, have a strong commitment to ensure an indigenous energy supply which they can control.

U. S. Government sponsored R&D, as well as R&D sponsored by other nations and through cooperative programs between nations, was motivated in early years to develop processes that would increase recovery of uranium and plutonium and to reduce personnel exposures. There was fear in those days that we would run out of uranium and it would become so costly that nuclear energy growth would end. There was also early recognition that reprocessing dealt with very high radiation levels so worker protection was essential. While little R&D is being done in the U. S., it is still being done abroad. Today the driving force is to reduce operating and maintenance costs and the above objectives are still valid as they are viewed as a means to that end. This has meant that waste quantities are being reduced and improvements made in the vitrification process for high level waste.

HIGH LEVEL WASTE HANDLING AND DISPOSAL

The handling and disposal of high level waste (HLW) has been studied, practiced and argued about for the past 50 years. The U. S. has always followed the path of proceeding as rapidly as possible to entomb the HLW in a permanent storage repository. The Europeans, United Kingdom and Japanese have a different strategy. They store the HLW in a temporary repository for 50 years while they decide which is the best permanent disposal method to use and where these repositories should be. The first U. S. attempt to build a permanent repository was at Lyons, Kansas (13). In the 1960s it was planned that the LWR spent fuel would be reprocessed and vitrified waste canisters would be the principal item to be stored. Studies by the National Academy of Science indicated that the salt beds underlying Lyons would make a safe storage media. The Oak Ridge National Laboratory (ORNL) had conducted many measurements using fully irradiated fuel elements from a test reactor to simulate waste canisters in an abandoned Carey Salt Company mine. By ORNL moving employees and their families to Lyons, the local townspeople accepted such a facility. They believed that ORNL engineers would not endanger their own families by placing them in an dangerous location. Therefore, the operation would be a safe one and they welcomed the jobs it would produce.

Significant Decisions

The support for the Lyons repository was not shared by people in the rest of the state, especially not by Congressman Skubitz. His first objection was to the trucks and railcars which would cross Kansas and the fear they would have accidents and contaminate the area. Even though these casks which would contain the waste had been subjected to extensive and

demanding test conditions, I doubt if he and his supporters ever changed their minds. However, that was not a deciding factor in the fate of the Lyons repository. Adjacent to the Lyons mine was the American Salt Company's mine. The mine had employed solution mining techniques in an effort to reduce costs. This practice was perceived to endanger the Lyons mine and probably was the final factor that caused AEC to abandon the Lyons site and became the first significant decision affecting the HLW program. I do not know if the AEC ever seriously considered buying the American Salt Mine and closing it. It was reported that the American Company's salt was uncompetitive in nearby Kansas City with salt from the Gulf of Mexico barged up the Mississippi and Missouri Rivers so the sale price of the mine would be low. Had that occurred it would have removed a serious obstacle in locating a repository at that time. Additional details and perspectives of the operations at Lyons can be found in C. R. Barlett's, Dr. Frank Parker's and W. G. Belter's papers given at this conference.

As discussed in the previous section, the second major decision was that U. S. chose to store spent fuel elements rather than vitrified waste from reprocessing. Opponents of reprocessing cite proliferation risks and claim that reprocessing is not economical compared to disposing of spent fuel. Proponents of reprocessing state that reprocessing is not that great a proliferation risk, that storing fuel elements in effect creates a plutonium mine because the fuel elements can be reclaimed and converted to plutonium if a nation desired. Furthermore, with the collapse of the Soviet Union, the far greater threat is from some terrorist group or nation stealing a bomb from one of countries that have been created since the Soviet break-up. Proponents also state it is impossible at this time to determine which route is more economical because while reprocessing costs are known and some information is available on temporary storage of both spent fuel and vitrified waste, there is no information available on the permanent storage of either spent fuel or vitrified waste.

With the decision to store spent fuel and the abandonment of Lyons, the U. S. turned its search elsewhere, (i.e.,. Yucca Mountain, Nevada, Hanford, Washington and Deaf Smith, Texas). The strategy at that time foresaw the possibility that a permanent repository may not be ready before the reactor spent fuel pools were filled to overflowing, so provisions were made to build a monitored retrievable storage (MRS) facility at Oak Ridge. Again the host location was receptive to this facility but the Tennessee state officials waged a successful campaign to kill the project at Oak Ridge based on perceived threats of major contamination from transportation accidents and the fear that a permanent repository would never be built and the waste would remain permanently in Oak Ridge. To pay for the storage, the utilities with nuclear plants were accessed a charge of 1 mill/kwhr of electricity sold. Congress then decided to only consider the Yucca Mountain site so all efforts have been focused there. With no reprocessing, the site will store spent fuel elements and is trying to decide if they can be retrievable.

Despite the many millions of dollars spent on the Nevada repository, the U. S. is still a long ways from an operating storage facility. Instead most utilities have been required to build additional on-site storage for their spent fuel besides paying 1 mill/kwhr of electricity generated for the government to store the fuel.

In contrast, reprocessing has significantly reduced the quantities to be stored in the UK and Europe and temporary HLW storage vaults are being

built (14). The plan is for the containers of vitrified waste to be stored for up to 50 years. Concurrently, an intensive program of investigating three different underground host rock formations for a permanent storage facility is underway. This program includes underground laboratories at Mol, Belgium to study clay formations, at Gorleben, Germany to study salt formations, and near Marcoule, France to study granite formations.

OVERALL SUMMARY AND CONCLUSIONS

The waste management practices of uranium mining and milling industry in the early 1950s through the 1960s were based on those of the mining industry and were accepted by the public. Health studies, changing public attitudes and legislation resulting from conditions in Grand Junction, Colorado and in the Grants, New Mexico area resulted in major changes being made. This includes returning overburden and waste rock back into the mines, the covering of mill tailings ponds, remediation of contaminated soils both at the mills and at surrounding properties and the cessation of injecting waste solutions deep into the ground. This again was another example of the failure of waste injection into the ground as being a safe method of waste disposal.

The technology used in reprocessing power reactor spent fuels is based on the PUREX technology developed at ORNL and used in U. S. government-owned plants. Although the early U. S. plants did a reasonable job in containing the waste streams generated particularly for iodine, the major shortcoming was that many of the waste streams had large volumes and were stored in the liquid state. Efforts through the years have focused on reducing the volume of these wastes and then solidifying the high level waste. This is now done routinely in France and the UK and has been done in Belgium. The U. S. decision to forego reprocessing and curtail practically all R &D has left this development to other countries. Their programs have resulted in lowering reprocessing costs, reducing waste volumes and lowering worker and the public's radiation exposure.

There are marked differences in the approach of the U. S. from other countries such as the UK, France and Japan in handling and disposal of high level waste. The U. S. continues to follow the path of entombing the spent fuel and other HLW in a permanent storage repository. In contrast, the above named countries, vitrify the reprocessed liquid HLW, place them in temporary storage for up to 50 years and are conducting an intensive study of the best way to permanently store the waste through testing in underground laboratories in different geological formations. Arguments are made for each approach that revolve on whether one is a greater proliferation risk than another and whether it is more economical to permanently store vitrified reprocessing waste or spent fuel. Neither approach addresses the problem that the most serious proliferation risk may come from the stealing of nuclear weapons. While we have obtained costs on reprocessing and some initial costs on the temporary storage of spent fuel and vitrified waste it is still impossible at this time to determine which route is more economical since no one knows what the respective costs are for permanently storing either product will be. While Europe and Japan have allocated up to 50 years to find a solution, the U. S does not have a specific time table but is trying to do it as soon as soon as possible. The Yucca Mountain site has not as yet been proven to be an acceptable site. There is no centralized temporary storage location so the spent fuel is being stored at the over 50 reactor locations in the U. S. and the rate payers continue to pay the 1

mill/kwhr charge for permanent storage. Who will arrive at the final solution to problem first and at what cost is still to be decided.

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Session 26 -- MANAGEMENT OF WASTE AND REGULATORY COMPLIANCE
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26-1

AN OVERVIEW OF THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

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ABSTRACT

The Waste Management Programmatic Environmental Impact Statement (WM PEIS) is a nationwide study examining the environmental impacts of managing five types of radioactive and hazardous wastes that result primarily from nuclear defense activitiesthe development, production, and testing of nuclear weapons at a variety of U.S. Department of Energy (DOE) sites located around the United States. The five waste types are low-level mixed waste (LLMW), low-level waste (LLW), transuranic waste (TRUW), high-level waste (HLW), and hazardous waste (HW). The waste types are described, and the major sites are identified. The range of decisions and various alternatives for treating, storing, and disposing of wastes are discussed. The analysis process and major results and insights are summarized. Finally, preferred alternatives are discussed.

INTRODUCTION AND OVERVIEW

The management of both current and anticipated volumes of low-level mixed waste (LLMW), low-level waste (LLW), transuranic waste (TRUW), high-level waste (HLW), and hazardous waste (HW) require safe and efficient handling and disposition of these wastes, compliance with all applicable Federal and State laws, and protection of public health and safety. Each waste type has unique physical and regulatory requirements (see Table I) and accordingly is managed separately. For each waste-type system, facilities are needed to treat, store, and dispose of the waste. For the first time, the U.S. Department of Energy (DOE) has attempted not only to examine in an integrated fashion the impacts of complex-wide waste management decisions for each waste type but also the specific cumulative impacts for all the waste facilities at a given site. In this context, management of these wastes includes:

- Modifying existing waste management facilities or constructing new facilities at particular sites;
- Operating modified or new waste management facilities at those sites;
- Transporting waste among waste management facilities, as necessary; and
- Sampling and analyzing waste constituents as necessary.

Table I

The Waste Management Programmatic Environmental Impact Statement (WMPEIS) provides information on the impacts of various siting alternatives that DOE will use in deciding where to locate additional treatment, storage, and disposal facilities for each waste type. However, the location of a facility at a selected site will not be decided until completion of a subsequent sitewide or project-specific environmental impact analysis. To assist DOE in making decisions regarding the sites at which it should locate waste management facilities, the WMPEIS considers four categories of alternatives for each waste type: a no action alternative that is generally consistent with current practice; a decentralized alternative that would, in general, result in wastes being managed where they are generated or stored currently; a regionalized alternative that would locate waste management facilities at a lesser number of sites throughout the nation; and a centralized alternative that would locate large waste management facilities at only one or two sites. For certain waste types, DOE considers more than one regionalized or centralized alternative to present a wide variety of options on the number and location of sites having major waste management facilities and the sites at which the facilities could be located.

Figure 1 identifies the sites where wastes are generated or stored for one or more of the types of waste evaluated in the WM PEIS. "Major" sites, as shown in the figure, are those candidate locations identified

under the WMPEIS alternatives that may receive wastes generated off-site and/or host disposal facilities. These sites received detailed evaluation in the WM PEIS.

Fig. 1

Table II summarizes the range of decisions that DOE needs to make with respect to the treatment, storage, and disposal of the waste types discussed. The location of waste management facilities to implement these decisions is addressed in the alternatives.

Finally, DOE has conducted a significant outreach program to obtain public input on the Draft WMPEIS. The public comment period was extended from its initial duration of 90 days beginning September 21, 1995 to 150 days, now ending February 19, 1996, several times the normal time provided by DOE for other major impact statements. In addition to receiving written comments for this period, DOE conducted 14 public hearings at 18 locations throughout the nation, selected in accordance with the management and siting alternatives analyzed in the WM PEIS. All of these hearings, which were attended by approximately 600 people, were conducted via interactive televideo conferencing. This innovative method saved DOE more than $400,000 compared with the normal procedure for hearings.

Table II

DOE will consider the public comments as part of an evaluation of the alternatives during the course of the decision process. Table III illustrates examples of the factors and criteria DOE may use to screen, evaluate, and narrow the current alternatives to select a preferred alternative for each waste type considered in the WM PEIS.

Table III

DOE will identify all of its preferred management alternatives by waste type in the Final WMPEIS, which is expected to be issued by late summer 1996. After issuance of the Final WMPEIS, DOE is considering holding another public comment period to receive input on the Department's preferred alternatives before Records of Decision are issued. DOE is also working with site-specific advisory boards and the DOE Environmental Management Advisory Board to develop appropriate processes for reaching consensus. Specific decisions regarding Department wide management of each waste type may be staggered over the coming years upon consideration of all appropriate information, including but not necessarily limited to the findings of the Final WMPEIS.

ALTERNATIVES

In the WM PEIS, an alternative is defined as a configuration of sites for treating, storing, or disposing of a specific waste type. The alternatives for each waste type fall within the aforementioned categories: the no action alternative, and decentralized, regionalized, and centralized alternatives. These four broad categories of alternatives encompass the range of reasonable alternatives available to DOE for siting of facilities for the management of the five waste types that are considered in the WM PEIS. However, under each category of alternatives, there are many possible combinations for the number and location of DOE sites for treatment, storage, and disposal facilities. In addition, there is more than one option for treatment of some waste types. To narrow these combinations to a level permitting meaningful analysis, DOE selected a total of 36 representative configurations to be ultimately analyzed over all the waste types considered in the WM PEIS.

To identify reasonable proposed sites for waste management facilities, DOE determined where the largest waste volumes were located and where transportation requirements would be minimized. The characteristics of the waste, specialized treatment requirements, and existing facilities were also taken into consideration in site selection. For example, some wastes that require special treatment were analyzed separately, and treatment sites were selected for analysis based on the volumes requiring special treatment rather than on total volumes. In some cases, treatment facilities could be used for more than one waste type. Therefore, some sites were evaluated as candidate sites even where the volume of a particular waste type was not among the largest. Table IV presents details by site for the alternatives evaluated in the WM PEIS. On-site LLMW wastewater treatment and minimum LLW treatment facilities would occur under all the alternatives.

Table IV

Table IV con't.

To illustrate the alternative selection process, consider the case of LLMW. The no-action alternative is to continue treatment at existing facilities. However, this alternative does not address disposal or comply with the Resource Conservation and Recovery Act. The decentralized alternative considers treatment at the 37 sites now generating or storing LLMW and considers disposal at 16 sites. Four regionalized alternatives were defined with treatment considered at from 4 to 11 sites and disposal considered at from 1 to 12 sites. The centralized alternative only considers 1 site, Hanford, for treatment and disposal of all LLMW.

ANALYSIS AND KEY RESULTS

To evaluate the potential environmental impacts of the alternatives, DOE first identified the type, characteristics, quantity, and special requirements (e.g., handling requirements) of each waste type. To frame the analysis within reasonable bounds and to make the analytical process more manageable, DOE developed and applied specific assumptions. DOE then determined the health risks, environmental impacts, and costs of waste treatment, transport, storage, and disposal as applicable for each waste type. Figure 2 graphically depicts this framework. Key results will be discussed in four areas: human health effects, transportation, the environment, and program costs and benefits.

Fig. 2

The human health analysis yielded four major results: First, in general, the human health impacts from constructing and operating waste management facilities would be low. Second, waste management activities would generally result in greater risks to waste management workers than to the off-site public. These risks would be primarily due to physical hazards, which could include construction accidents or other accidents related to typical industrial activity. Third, the incineration of transuranic waste requires special technologies or considerations to avoid adverse effects in the off-site population. Fourth, safe disposal of low-level waste and low-level mixed waste at certain facilities may require limitations on the type or quantity of radionuclides. Without imposing such limitations as needed on a site-specific basis, groundwater contamination could exceed standards for drinking water. The analysis of transporting wastes between sites yielded three major insights: First, transportation poses a greater risk to the general public than the construction and operation of waste management facilities. This risk is primarily related to traffic accidents rather than exposure to radiation or chemicals. Second,

transporting waste by rail generally poses less risk to the public than transporting by truck. Based on highway accident statistics, shipping high-level waste over 40 years and other wastes over 20 years by truck could potentially result in from 12 to 69 fatalities. The majority of these fatalities would be from traffic accidents. Shipping all wastes except hazardous by rail over this same time period could result in 2 to 6 fatalities. Third, transporting wastes to consolidation facilities would generally be less expensive than building facilities at many DOE sites, but could result in increased air emissions and traffic fatalities. This situation is a very clear example of the tradeoffs that DOE must consider and the decisions it must make.

The primary environmental impact areas examined were air and water quality. The following sites, RFETS, Hanford, NTS, LANL, WIPP, and ORR could have air quality impacts under some of the regionalized and centralized alternatives. (See Table IV for full site names.) Regulatory standards for air may be exceeded as a result of increases in emissions from on-site construction equipment and vehicles, and from thermal treatment of transuranic or hazardous waste. In addition, the groundwater analysis shows that using a generic disposal facility, Hanford, SRS, FEMP, SNL-NM, and PGDP could experience groundwater impacts under one or more of the PEIS alternatives. However, steps can be taken to avoid air and water impacts. The potential impacts identified can be reduced or eliminated by carefully choosing the technology to be used, facility location, method of vehicular transport, or other mitigating measures. For example, predicted impacts could be avoided or reduced by the use of rail instead of truck transport for some types of waste, or by avoiding certain types of treatment for particularly toxic wastes.

The social and economic impacts of these decisions will be very important to DOE, as illustrated in three major findings. First, using commercial facilities to treat DOE hazardous waste could result in lower costs than building new DOE facilities. Generally, DOE treats its wastewater at the DOE sites and uses commercial facilities to treat and dispose of its remaining hazardous waste. According to the WMPEIS analysis, this approach may be more cost effective and result in fewer impacts than constructing new DOE facilities. Second, the cost to manage all five waste types over 20 years could range from $20 to $45 billion. Costs depend on which alternatives are chosen and tend to decrease as the number of treatment and disposal sites decrease (for example, under the centralized alternatives). Third, socioeconomic impacts would be relatively small, with the maximum increase in jobs generated in the economies around any site being in the range of one to three percent. On a national basis, the largest economic benefits would result from decentralized alternatives and generally would decrease as alternatives become more centralized. Of course, a site chosen for a centralized alternative would experience the maximum economic benefit.

It should be emphasized that the WMPEIS analyses were based on generic assumptions appropriate for a programmatic document. Before any new large waste management facilities are built at a site, more detailed and site-specific analyses would be conducted. Therefore, the results presented provide early indications of where problem areas may be encountered and are not the basis for construction and operation of new facilities.

SUMMARY AND PRELIMINARY CONCLUSIONS

Many lessons have been learned from the evaluations of the treatment, storage, and disposal of the waste types. The key points for each waste

type are as follows. For LLMW, the highest risks are to waste management workers associated with construction activities, although some individual site environmental impacts occur under the Centralized Alternative, which utilizes the lowest number of workers. Costs range from $8 billion for the Centralized Alternative to $13 billion for the Decentralized Alternative. Radionuclide- and/or chemical-specific limits will be required for disposal at most sites.

For LLW at the national level, costs, risks, and impacts are greater for volume reduction than minimum treatment. Transportation poses the highest risks to the public. Centralized disposal results in a large transportation volume with commensurately greater transport risk from both traffic accidents and radiation exposure. Rail transport has significantly lower risks than truck transport. As for LLMW, radionuclide-specific limits will be required for disposal at most sites.

For TRUW, the LDR treatment of TRUW poses greater risks, air quality impacts, and costs than lesser levels of treatment. Transportation risks and costs were roughly equivalent for all alternatives involving shipping to WIPP. For HLW, costs and risks are slightly higher for centralized storage at Hanford, but the differences are not significant. Alternatives are roughly equivalent from the standpoint of environmental impacts and costs. The acceptance rate at the candidate repository controls length of storage time. For HW, risks and impacts are similar for each alternative and the costs favor commercial treatment.

As part of the preparation of the WM PEIS, DOE has identified the following preferred alternatives for treatment or storage of the five waste types. These preferences are based on the lessons learned as cited above, on continued use of effective or proven DOE practices, and in part on the DOE's proposed site treatment plans for LLMW:

The No Action (status quo) Alternative is preferred for treatment of non-wastewater hazardous wastes, which continues use of commercial facilities. Treatment of hazardous wastewater would continue at DOE sites.

DOE prefers to continue to store HLW on-site at the Hanford Site, INEL, and SRS pending disposal in a geologic repository. This arrangement can be accommodated under the No Action, Decentralized, or Regionalized alternatives. DOE does not yet have a preference on where to store West Valley HLW pending disposal in a geologic repository.

The Regionalized Alternatives are preferred for LLMW treatment because they most closely approximate DOE's Proposed Site Treatment plans developed under the Federal Facilities Compliance Act. However, negotiations are underway with regulatory authorities regarding the proposed plans, and DOE's preference for LLMW treatment may be affected by these negotiations.

At the time of this report, DOE had not yet identified a preferred alternative for management of LLW and TRUW.

Finally, DOE is now reviewing the public comments. The most significant issues raised were the relationship and integration of the WMPEIS with other significant DOE proposed actions; currency of waste inventory data used in the analysis; long-term storage of wastes versus disposal; the effect of the analysis if the WIPP and Yucca Mountain facilities do not become available; privatization of WM facilities versus DOE ownership and operation; relationship of environmental restoration activities to WM; transportation planning, risks and emergency preparedness; and "not-in-my-backyard" concerns, particularly for disposal of wastes. Other issues

frequently mentioned involved environmental justice, groundwater contamination in sensitive areas, and adverse economic impacts such as decreased land value near sites chosen for WM facilities. All these issues will be addressed in the preparation of the Final WMPEIS.

26-2

POTENTIAL WASTE-CLEARANCE STRATEGY FOR U.S. DEPARTMENT OF ENERGY WASTE PROCESSED AT TREATMENT, STORAGE, AND DISPOSAL FACILITIES

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ABSTRACT

Past practices at U.S.Department of Energy (DOE) field facilities may have resulted in the presence of minute amounts of radioactive contamination in some hazardous wastes shipped from these facilities. In May 1991, the DOE Office of Waste Operations issued a nationwide moratorium on shipping potentially mixed waste from DOE facilities to commercial treatment, storage, and disposal (TSD) facilities. A potential waste-clearance strategy was developed to address the DOE mixed-waste moratorium issues, which had resulted from a lack of existing regulations regarding volume contamination. A radiological assessment model was developed on the basis of the detailed radiological assessment performed for eight commercial hazardous waste TSD facilities. The model incorporates waste- and site-specific data to estimate potential radiological doses to on-site workers and the off-site public from waste-handling operations at a TSD facility. The described waste-clearance strategy would provide both DOE and commercial TSD facilities with a rapid and cost-effective methodology for assessing potential human exposures from the processing of chemical wastes contaminated with trace amounts of radionuclides. This strategy also has important potential applications for establishing site clearance limits to ensure that worker and public risks would remain well below regulatory limits. The clearance strategy issues pertaining to current free-release practice, dose limits, data requirements, and conservatism are discussed.

INTRODUCTION

U.S.Department of Energy (DOE) facilities have historically disposed of their radioactive waste at DOE-owned radioactive waste disposal sites. The nonradioactive but hazardous chemical component of the DOE wastes has been treated and disposed of at permitted commercial treatment, storage, and disposal (TSD) facilities. However, past practices at DOE field

facilities may have resulted in the presence of minute amounts of radioactive contamination in some hazardous wastes shipped from DOE facilities to commercial TSD facilities.

In May 1991, the DOE Office of Waste Operations issued a nationwide moratorium on shipping potentially mixed waste from DOE facilities to commercial facilities. The moratorium is to remain in effect until procedures are approved and implemented to ensure that hazardous wastes contain no radioactive materials added by DOE operations (i.e., the "no-rad added" policy). The "no-rad added" policy was imposed because of the lack of existing policy and regulations on the release of slightly contaminated bulk materials. Assessment of previous DOE wastes at TSD processing facilities revealed extremely low radiological risks to both TSD workers and members of the public (1).

Strategies for clearance of radioactive materials based on low-risk criteria have been prescribed by international organizations (2). A potential waste-clearance strategy is discussed in this paper for addressing the DOE mixed-waste moratorium issues, which resulted from a lack of existing regulations regarding volume contamination. The radiological doses evaluated from waste processing at eight TSD facilities are orders of magnitude below federal limits and guidelines, as well as the average dose from natural background radiation in the United States (discussed in the section "Comparison of Doses and Risks"). This leads to the potential establishment of "authorized limits for release" as an interim measure, according to current DOE guidance, and ultimately leads to waste clearance based on risk (the section "Strategy towards Waste Clearance" describes this process). A radiological assessment methodology was developed as part of the clearance strategy on the basis of detailed radiological assessment of these TSD facilities (described in the "Development of Dose Assessment Model" section). This methodology provides a simplified physical concept of the potential human exposure associated with the radioactive contents of hazardous chemical wastes.

COMPARISON OF DOSES AND RISKS

On-site worker and off-site public doses were calculated for the eight commercial TSD facilities. The preliminary results are discussed below.

Estimated Doses to Workers and the General Public from Waste-Handling Operations at TSD Facilities

The worker doses include the external exposure as well as the inhalation dose. Maximum annual worker doses were extremely low, ranging from 210-5 to 710-2 mrem. In the absence of more specific data, cumulative worker doses were calculated on a conservative basis, i.e.,assuming that each worker was engaged in the same operations for every year the DOE wastes were received on-site. These cumulative doses ranged from 210-5 to 810-2 mrem.

Two types of workers, inspection/sampling workers and incinerator workers, received the highest annual and cumulative doses. Doses to these workers were dominated by internal doses. Doses to inspection/sampling workers were mostly due to sampling of solid materials (i.e., dirt), which can result in the generation of airborne respirable particulates. Doses to incinerator workers were mostly due to incinerator maintenance. Because of the greater uncertainty, the estimation of internal doses was based on more conservative assumptions than that of external doses, resulting in relatively higher internal doses. For example, resulting data were not available to directly evaluate airborne concentrations

during incinerator maintenance and sampling of solid materials; therefore, a relatively conservative airborne dust concentration of 10 mg/m3 was employed to model these doses. Based on both the likely particle distribution and our experience, it is probable that these values exceeded the actual respirable concentrations by 10- to 100-fold. The potential radiological doses to the general public residing in the vicinity (within 50 mi [80 km]) of the TSD facilities were assessed. Based on an examination of the operations and disposition of the wastes, no credible off-site release of radioactive material to the atmosphere were identified at two facilities (1). At the remaining six facilities, for which population doses were evaluated, maximum annual individual doses ranged from 110-7mrem to 610-3 mrem. Cumulative maximum individual doses, calculated for the entire period that DOE wastes were shipped to commercial TSD facilities, ranged from 210-7 to 610-3 mrem (1). The maximum annual collective population doses ranged from 410-7 to 310-1 person-rem. The total population dose was 0.4 person-rem for all DOE wastes sent to TSD facilities.

Comparison with Federal Standards and Regulations

Federal radiation protection limits differ for members of the public and those exposed as a result of their occupation. The calculated maximum individual public radiation doses are less than 0.1% of the air pathway limit of 10 mrem/yr established by the U.S.Environmental Protection Agency (EPA) for protection of the public (3). The calculated public radiation doses are less than 0.01% of 100 mrem/yr, which is the DOE limit (4)for protection of the public from all sources and all pathways combined. Maximum annual doses to workers are less than 0.1% of the 100-mrem/yr DOE guideline for the general public, such as the unmonitored workers who might have been exposed at the TSD facilities. In addition, DOE has issued guidance establishing a collective dose limit of 10 person-rem/yr (5). The calculated total population dose from all pathways combined is less than 5% of the established guidance. Figure 1 lists some of these standards, their applications, and sources (3-4, 6-12).

Fig. 1

Comparison with Background Radiation

To provide some perspective, the calculated doses to workers and the public were compared with those derived from natural background radiation. In the United States, the average annual dose from natural background radiation is about 300 mrem to an individual (9). In comparison, the estimated dose in any one year to the maximally exposed worker does not exceed 0.03% of the average annual dose from background radiation. The estimated dose in any one year to a maximally exposed member of the off-site public does not exceed 0.002% of the average annual dose from background radiation. For the entire population within 80 km (50 mi) of the various sites, the estimated maximum population dose in one particular year does not exceed 0.00001% of the average dose from natural background radiation received by a similar population.

Comparison with Potential Release Levels

Although no international or U.S.standard exists for the release of bulk materials contaminated with radioactivity, there are principles or examples with which the derived TSD dose (or risk) levels can be compared. In Safety Series No.89 (2), the International Atomic Energy Agency (IAEA) states the principles for exemption of radiation sources and practices from regulatory control. The IAEA recommends that an individual dose of 1 mrem/yr (annual risk level of 10-6 to 10-7) to a

"critical group" of the general public would offer sufficient protection for exemption purposes. The level of 1 mrem/yr represents only 1% of the current recommended limit of 100 mrem/yr to the general public. In comparison with the IAEA exemption level of 1 mrem/yr, the calculated TSD doses are lower by a factor of 10 or more.

Comparison was also made with environmental assessments (11,12) of specific cases that have been (or are in the process of being) prepared for the release of radioactively contaminated metals. The calculated TSD dose levels in these cases are quite comparable with the other dose values.

STRATEGY TOWARD WASTE CLEARANCE

Numerical guidance exists for the unrestricted release of surface-contaminated materials, such as that found in Regulatory Guide 1.86 (13) of the U.S.Nuclear Regulatory Commission (NRC) and equivalent guidance in DOE Order 5400.5 (4). However, equivalent guidance for the unrestricted release of volume-contaminated materials does not exist. Generic guidance on release of bulk contamination is contained in DOE Order 5400.5 (and also in the proposed DOE 10 CFR Part 834 ruling), which allows establishment of "authorized limits for release" based on as low as reasonably achievable (ALARA) analysis.

Interim guidance on ALARA has been issued by DOE (14). Essentially, compliance with ALARA incorporates the following factors: objective, options, cost-benefit analysis, and control. Although the objective for hazardous waste is release, the options may consist of disposal as low-level radioactive waste or off-site shipment for treatment and/or disposal. Costs would be incurred for the activities associated with the options. Release for off-site treatment or disposal would require further activities of surveying, monitoring, and control prior to the release. Weighing costs against risks is certainly the most important element of ALARA and, therefore, for the entire release strategy. Estimating costs, however, is not the subject of this paper. This paper focuses instead on the potential risks incurred in the release of waste for off-site treatment or disposal. As discussed earlier, potential individual risks from release of DOE wastes have been found to be very low, generally below 510-8. For this reason, it would be feasible to establish a waste-clearance process based on risks. This can be accomplished by establishing the "authorized limits for release." How to characterize risks, therefore, becomes a crucial element in this process.

Although it would be feasible to establish a clearance approach for the release of DOE wastes to the commercial TSD facilities, it is important that such releases would also meet state or local regulations. In general, state regulations are tied to existing federal statutes. However, because no federal statutes exist regarding the clearance (unrestricted release) of radioactively contaminated wastes, most states do not have regulations, although some have established release limits of their own. As a result, the stakeholder's acceptance of DOE wastes becomes an integral component of the waste-clearance strategy. In this regard, DOE needs to establish a clear communication and consensus with state regulators, as well as with commercial TSD site operators. Also, as an input to the ALARA analysis, a baseline cost estimate should be established. Achieving such consensus is not a trivial matter; past experience, however, supports the feasibility of such an approach (11).

Aside from the licensed radioactive disposal cost, cost elements associated with waste characterization, surveying, instrumentation,

decontamination, packaging, interim storage, transportation, and so forth are also important in reaching an ALARA decision.

The authorized limits are intended as an interim measure before a universal unrestricted standard is developed. The authorized limits are basically intended for site- or region-specific concerns, with each limit based on specific cost and risk information as input to ALARA practice. It is possible that as time evolves, commonalities can be identified among sites and the wastes generated such that a complex-wide standard can be developed. Furthermore, international agencies such as the IAEA and the Commission of the European Communities are in the process of developing an unconditional clearance standard for slightly contaminated radioactive materials.

DEVELOPMENT OF DOSE ASSESSMENT MODEL

A computer dose model has been developed to support the waste-clearance strategy for DOE wastes. The dose assessment model considers exposure to two groups: on-site workers and off-site members of the public. This model, depicted in Fig. 2, was developed in the following steps on the basis of previous detailed analysis of the TSD facilities studied.

Fig. 2

Identification of Major Receptors and Major Operations

The receptors identified for the TSD facilities include four classifications of on-site workers, an off-site individual, and the general population. Operations are carried out in the same sequence as those for waste processing, including the following:

- Transport of wastes from DOE generator sites and transport of incineration residues off-site;
- Acceptance of waste at TSD facility including unloading, sampling, check-in, and storage;
- Waste incineration, which involves incineration, transport of incineration residues on-site, and incinerator maintenance;
- On-site landfilling of incineration residues.

Incineration and landfilling operations contribute to the off-site public dose.

Selection of Isotopes

The radionuclides incorporated in the model were based on previous DOE waste inventories at TSD sites. Additionally, the model also incorporates the radionuclides regulated under 10 CFR Part 61. The model currently can be applied to wastes containing uranium (U-232, U-234, U-235, U-236, U-238), plutonium (Pu-238, Pu-239, Pu-240, Pu-241, Pu-242), americium (Am-241), carbon (C-14), cesium (Cs-137), cobalt (Co-60), curium (Cm-242), iodine (I-129), nickel (Ni-59, Ni-63), niobium (Nb-94), strontium (Sr-90), technetium (Tc-99), and tritium (H-3).

Selection of Pathways

External exposure, inhalation, and ingestion were considered for all receptors. To extend the model to a broader range of radionuclides and site conditions, simplified groundshine and ingestion pathways were added to the plume-based inhalation.

Calculation of Doses

Simplified formulas were derived to calculate doses to on-site workers and off-site members of the public for each reference operation. By using conservative parameter values derived from the eight TSD facilities, bounding dose calculations were performed on the basis of unit activity. On-site worker external doses were based on scaling of microshield calculations. On-site worker internal doses were dependent on mass

loading factors to determine the quantity inhaled and ingested. Off-site receptor doses were based on the CAP88-PC gaussian plume exposure model (15).

Development of Graphical User Interface

A Windows graphical user interface was developed to estimate the worker and off-site public doses (both external and internal) for a specific TSD. The model enables the user to:

- Easily adjust key parameters for a site-specific assessment;
- View context-specific help;
- View results in tabular or graphical form;
- Save site-specific data to retrievable files;
- Access data forms through multiple means: menu, icons, and command buttons;
- Easily select applicable radionuclides;
- Discriminate site-specific values from default values;
- Easily identify maximally exposed receptors; and
- Compare resultant doses with regulatory limits.

SUMMARY

In summary, it is feasible to establish a DOE waste-clearance strategy for TSD facilities on the basis of existing guidance on ALARA. A risk tool in dose assessment is being developed as an important component of the clearance strategy. Other requirements for successful implementation of such a strategy include acceptance by states and stakeholders, as well as other logistics that include cost estimates and implementation of details to be identified later.

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26-4

SCOPING ANALYSIS OF TOXIC METAL PERFORMANCE IN DOE LOW-LEVEL WASTE DISPOSAL FACILITIES*

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ABSTRACT

This study provides a scoping safety assessment for disposal of toxic metals contained in Department of Energy (DOE) mixed low-level waste (MLLW) at six DOE sites that currently have low-level waste (LLW) disposal facilities - Savannah River Site, Oak Ridge Reservation, Los Alamos National Laboratory, Hanford Reservation, Nevada Test Site, and Idaho National Engineering Laboratory. The study has focused on the groundwater contaminant pathway, which is considered to be the dominant human exposure pathway from shallow land MLLW disposal. A simple and conservative transport analysis has been performed using site hydrological data to calculate site-specific "permissible" concentrations of toxic metals in grout-immobilized waste. These concentrations are calculated such that, when toxic metals are leached from the disposal facility by infiltrating water and attenuated in local ground-water system, the toxic metal concentrations in groundwater below the disposal facility do not exceed the Maximum Contaminant Levels as stated in the National Primary Drinking Water Regulation. The analysis shows that arid sites allow about 100 times higher toxic metal concentrations in stabilized waste leachate than humid sites. From the limited available data on toxic metal concentrations in DOE MLLW, a margin of protection appears to exist in most cases when stabilized wastes containing toxic metals are disposed of at the DOE sites under analysis. Possible exceptions to this conclusion are arsenic, chromium, selenium, and mercury when disposed of at some humid sites such as the Oak Ridge Reservation. This analysis also demonstrates that the U.S. Environmental Protection Agency's prescriptive regulatory approach that defines rigid waste treatment standards does not inherently account for the variety of disposal environments encountered nationwide and may result in either underprotection of groundwater resources (at humid sites) or an excessive margin of protection (at arid sites).

INTRODUCTION

Six DOE sites - Savannah River Site (SRS), Oak Ridge Reservation (ORR), Los Alamos National Laboratory (LANL), Hanford Reservation (HR), Nevada Test Site (NTS), and Idaho National Engineering Laboratory (INEL) currently dispose of low-level waste (LLW), and each is planning to dispose of mixed low-level waste (MLLW) (1). This study has focused on the groundwater contaminant pathway, which is generally considered to be the dominant human exposure pathway from shallow land disposal facilities such as these (2). A simple and conservative transport analysis has been performed to estimate site-specific "permissible" toxic metal concentrations in stabilized waste. Toxic metals present in MLLW were identified and the permissible toxic metal concentrations in groundwater at the site performance boundary (MCLs)were defined. From these concentrations the disposal facility "permissible" toxic metal concentrations in stabilized waste for the disposal facility were derived based on the estimated concentration attenuation provided by the waste form and on environmental transport from the disposal facility to the performance boundary using the conceptual groundwater pathway model described below. These "permissible" stabilized waste concentrations were then compared with data on toxic metal concentrations in DOE MLLW to assess the feasibility of disposal of toxic metal contaminated wastes in DOE LLW disposal facilities. The results of the analysis are used to discuss the U.S. Environmental Protection Agency (EPA) regulation related to disposal of chemotoxic wastes.

Requirements for managing DOE LLW are established in DOE Order 5820.2A (3). While the Order does not give specific guidance on toxic components of mixed waste, it requires that the hazardous component of DOE MLLW be managed to conform to the Resource Conservation and Recovery Act (RCRA) requirements of appropriate regional authorities. The list of toxic metals regulated under the RCRA (commonly referred to as RCRA metals) is given in Table I. The toxicity of waste is determined by the Toxicity Characteristic Leaching Procedure (TCLP), a test of the leachability of solids containing any one of a list of specific contaminants. A concentration of RCRA metal in the extract above its regulatory level (column 2 of Table I) renders the waste hazardous.

Table I

The Hazardous and Solid Waste Amendments (HSWA) to the RCRA prohibit land disposal of untreated hazardous wastes. This portion of the HSWA is referred to as the Land Disposal Restrictions (LDR). Treatment standards for wastes containing toxic metals have been established by EPA in the form of maximum permissible hazardous constituent concentrations in waste TCLP extracts. Corresponding concentrations are listed in column 3 (LDRs for characteristic wastes) and column 4 (universal treatment standards (UTS) for listed wastes) of Table I. EPA has recently proposed that the metal UTS replace LDR treatment standards for characteristic metal wastes in order to reduce confusion and make compliance easier (60 FR 43654, August 22, 1995). This proposed rule is not yet finalized.

TECHNICAL APPROACH

Model Assumptions

In accordance with performance assessment guidelines established by the DOE Performance Assessment Task Team (4): 1) a buffer zone of 100 m is used in assessing compliance with the performance objective for protection of groundwater resources, and 2) the requirement in 5820.2A to "protect groundwater resources consistent with federal, state, and local requirements" is taken to mean that the human exposure benchmarks are

Maximum Contamination Levels (MCLs) based on 40 CFR 141, the National Primary Drinking Water Regulations. The same document recommends a 10,000-year period of compliance for the performance objectives for groundwater protection.

The MCL values for RCRA metals that are used as human exposure benchmarks in the analysis are listed in column 5 of Table I. Arsenic (As), lead (Pb) and silver (Ag) do not currently have MCLs. The MCL values promulgated in 40 CFR 141.11, which were effective until December 7, 1992, were used for As and Pb. Silver is not considered in the analysis because it does not have an MCL, and the only potential adverse effect from exposure to Ag in drinking water is argyria (a discoloration of the skin), considered a cosmetic effect which does not impair health (56 FR 3526, January 30, 1991).

Immobilization/solidification techniques using cement-based grouts are considered the most common and universal option for the treatment and ultimate disposal of MLLW for variety of waste matrices (5,6). In particular, this stabilization technique is likely to be applied to the majority of MLLW streams containing toxic metals (i.e., inorganic solids and aqueous liquids (7)). Therefore, the analysis considers cement-based grout as a reference waste form for stabilization of the MLLW.

Because the uncertainty in hydrogeologic and geochemical parameters is frequently large even at well characterized sites, a simple and conservative groundwater pathway model (6), shown in Fig. 1, has been used in this analysis. The following major assumptions were used (6):

Source Term Assumptions. 1) The waste form was grout-stabilized. 2) Effects of contaminant solubility limits were neglected. 3) Performance of the disposal facility's engineered barriers is limited in time. After failure of the engineered barriers, the infiltration through the disposal facility is equal to the natural infiltration through local soils. 4) A continuous contaminant source was assumed for the transport calculations.

Environmental Transport Assumptions. 1) A one-dimensional analysis was performed. 2) Leachate dilution with groundwater was the only concentration attenuation mechanism accounted for. The contaminant-plume mixing depth in the saturated zone used in the "mixing cell" dilution model was estimated from either a) constraining hydrogeology (e.g., thin aquifers, sequences of units of high and low permeability) or b) vertical transverse solute dispersion. The last method was based on a one-dimensional flow, three-dimensional transport model (8). 3) Effects of longitudinal dispersion were neglected. (Once a continuous source model is assumed, this assumption is of minor importance.) 4) Fractured flow in the vadose zone was addressed by removing the fractured section from the stratigraphic column, implying complete and instantaneous transport through the fractured zone. 5) Fracture flow in the saturated zone was treated as an equivalent matrix porosity.

Calculations

The contaminant concentration attenuation in the groundwater pathway was calculated by assuming a mixing cell beneath the disposal facility in which contaminated leachate is mixed with clean groundwater (Fig. 1). The resulting concentration in the groundwater is calculated based on mass balance with the following formula:

Fig. 1

Eq. 1

where Cl is the leachate concentration (M/L3), Ql is the leachate volumetric flux (L3/T), Qgw is the groundwater volumetric flux (L3/T), a

is the length of the disposal facility parallel to the groundwater flow (L), b is the width of the disposal facility perpendicular to groundwater flow (L), ql is the infiltration rate through the disposal facility (L/T), dm is the mixing depth in the aquifer (L), and qgw is the groundwater Darcy velocity (L/T). The parameters a and b were assumed to be 50 m (6). Attenuation caused by environmental transport was represented by a dimensionless concentration reduction factor (CRFDW)

Eq. 2

The contaminant arrival time at the 100-m performance boundary was calculated as the sum of the retarded contaminant travel time in the vadose (tv) and saturated zones (ts):

Eq. 3

where lv is the depth from the disposal facility to the saturated zone (L), ls=100 m is the distance between the disposal facility and performance boundary, the qv and n are the volumetric moisture content in the vadose zone and the porosity in the saturated zone, respectively (dimensionless), and the retardation factors Rv and Rs (dimensionless) for the vadose and saturated zones are given by

Eq. 4

where Kd,v and Kd,s are the metal-soil distribution coefficients (L3/M) (discussed in the next section), and rb,v and rb,s are the soil bulk densities (M/L3) for the vadose and saturated zones, respectively. Similar to the attenuation for environmental transport, the toxic metal concentration attenuation provided by the waste form is represented by the source concentration reduction factor, CRFS. The CRFS was estimated based on the assumption that the toxic metal concentration in the waste form leachate is determined by desorption into infiltrating water (6):

Eq. 5

where Cw is the toxic metal concentration in the grouted waste form averaged over the entire volume of waste in the disposal facility (M/L3), qG is the volumetric water content of the grouted waste form (dimensionless), Kd,G is the distribution coefficient of the toxic metal in the grout (L3/M), rG is the bulk density of the grout waste form (M/L3), and fm is the mixing fraction, defined as the ratio of the volume of disposed waste in a unit volume of the facility (dimensionless). The following parameter values have been used in the calculations for equation (5) (6): qG =0.3, rG = 1.76 g/cm3, fm = 0.66.

Once the CRFDW and CRFS are calculated for each site and each toxic metal, the permissible waste form leachate concentrations (Cpl) and permissible concentrations in stabilized waste (Cpw) were calculated for the toxic metals as

Eq. 6

Eq. 7

Sorption Parameters for Toxic Metals

The RCRA toxic metals can be divided into three broad classes: 1) cationic elements (Ba, Cd, Pb, Hg), 2) anionic elements (As), and 3) redox-sensitive elements (Cr, Se) (9). In most cases, adsorption and precipitation cause very low mobility of cationic species in subsurface environments. For anionic species, adsorption and ion exchange will cause relatively little retardation in the subsurface, though oxyanions (e.g., AsO33-, CrO42-, SeO32-) are adsorbed specifically by mineralogical surfaces which carry a positive charge. Chemical speciation and mobility of the redox sensitive elements are strongly dependent on specific hydrochemical conditions.

Whenever site-specific RCRA metal sorption parameters were available, these were used in transport calculations. However, for a number of metals, site-specific sorption data were lacking. To provide the model with input parameters in the latter case, a generic toxic metal distribution coefficient (Kd) list was compiled (Table II). Literature review revealed applicable quantitative sorption data for some RCRA metals that are very limited for soils and especially for grout matrixes. In the latter case, values for sorption parameters have been assigned based on modeling or empirical relationships. The mercury (Hg) geosphere and grout Kd values are based on predictions of the MINTEQA2 geochemical model for the "medium pH of solution, medium in iron oxides and natural organic content matrix" conditions and "high pH of solution, low in iron oxides and natural organic content matrix" conditions respectively (13). As and Cr were assigned grout Kd values based on empirical pH-dependent "metal-sediment" partitioning relationships (15) assuming pH=12.5. Grout Kd values for the toxic metals are rounded to the nearest order of magnitude so as not to imply more precision in estimated values than is justified (6).

Table II

RESULTS AND DISCUSSION

Toxic Metal Concentrations in DOE MLLW

Very limited data are available on actual concentrations of toxic metals in DOE MLLW. Waste concentrations of RCRA toxic metals in several Oak Ridge Reservation MLLW waste streams are presented in Table III. Whether these data are representative of the entire DOE MLLW inventory is unknown, although similar processes should provide similar waste streams. The final column of Table III gives the weighted averages of all the waste streams.

Table III

Site-Specific Analyses

The summary (6) of the site-specific models and data for the six DOE sites considered in this analysis is presented in Table IV. Due to the extremely arid nature of the Nevada Test Site hydrogeological environment, a groundwater pathway is not believed to exist, implying total groundwater protection at this site (6). Therefore, no water pathway analysis was included in this work for the NTS. Calculated site-specific concentration reduction factors due to waste leachate mixing with groundwater (Table IV) show that leachate dilution provided by arid sites (INEL and LANL) is about two orders of magnitude greater than for the humid sites (ORR and SRS). The semi-arid Hanford Site has an intermediate dilution value. Estimated site-specific CRFDW (Table IV) and CRFS (Table II) were multiplied by the RCRA metal MCLs (Eqs. 6 and 7) to produce site-specific permissible leachate concentrations and stabilized waste concentrations for the RCRA metals (Table V). Values in Table V can be used as the basis for guidance to establish site-specific waste form performance criteria and waste acceptance criteria.

Table IV

The travel time calculations (Table V) show that, except for a few estimates (e.g., Ba and Se travel times for the Hanford Site), characteristic RCRA metal travel times for the arid and semi-arid sites exceed the DOE recommended LLW disposal compliance period of 10,000 years (4). For humid sites, the contaminant arrival times at the performance boundary will occur at least 500 years beyond the assumed performance of

engineered barriers, with most of the contaminant travel times falling within the higher range of thousands and tens of thousands years.

The last column of Table V shows estimated toxic metal concentrations in the example stabilized ORR waste. These estimates are based on data in Table II, assuming that the grout to MLLW mixing ratio is 1:1 and that the stabilized waste density is the same as the grout density. Assuming that RCRA metal concentrations in these example waste streams are typical for DOE sites, the data in Table V show that disposal of MLLW at the sites under analysis will satisfy or nearly satisfy groundwater protection requirements for most toxic metals. Exceptions are the anionic and redox-sensitive elements As, Cr and Se, and Hg. The concentrations of these metals are approximately an order of magnitude higher than the estimated permissible concentrations at some humid sites (e.g., ORR). Although this analysis is conservative, it indicates that care must be taken when disposing of As, Cr, Se, and Hg in humid environments. Waste streams containing a large amount of these elements may require enhanced stabilization (other than grout stabilization) prior to disposal, or even relocation for disposal to a semi-arid or arid environment. However, more detailed analyses that take credit for additional attenuation factors may provide higher permissible waste concentrations.

Table V

Implications for Regulation of Chemotoxic Waste Disposal

The rationale for the EPA TCLP-based TC, LDR, and UTS standards (Table I) is that corresponding concentration levels in the TCLP waste extract are considered to be an upper bound of contaminant concentrations in the disposed waste leachate under field conditions (55 FR 11862, March 22, 1990). The TC levels for toxic metals have been established as 100 times the MCLs which were in effect at the time of the rulemaking, with a generic 100-fold dilution/attenuation factor estimating attenuation of the toxic constituent leachate concentration in groundwater during subsurface transport from the disposal facility to the point of human exposure (45 FR 33084, May 19, 1980; 55 FR 1862, March 22, 1990). The LDR and UTS standards are also based on technology (i.e., based on performance of the Best Demonstrated Available Technology [BDAT] for hazardous waste treatment) (55 FR 22689, June 1, 1990; 59 FR 47982, September 19, 1994). However, the health risk considerations are implicit in the EPA rulemaking, because, with the exception of Se, the LDR standards for toxic metals are exactly the same as the TC levels.

Comparison of the site-specific permissible leachate concentrations estimated in this study (Table V) with TC, LDR and UTS standards (Table I) shows that, except for a few species (e.g., As), the UTS standards compare well with the estimates of permissible leachate concentrations for humid sites and will therefore ensure groundwater protection for most disposal environments, including humid ones; the LDR and TC standards will provide adequate protection at semi-arid and arid sites only. At the same time, the UTS waste treatment standards provide unnecessarily large margins of groundwater protection in arid disposal environments. Thus, the current EPA regulatory approach of setting a single waste treatment standard inherently does not account for a variety of hydrogeological disposal environments encountered nationwide. The result of such regulatory action is either potential underprotection of humid environments or unnecessarily high protection of arid sites. A performance-based regulatory approach, consisting of setting a performance objective to be fulfilled in every regulatory action, would

provide a much more consistent, flexible, and cost-effective way of protecting human health and the environment (2).

CONCLUSIONS

Arid sites appear to provide a greater degree of groundwater protection from land disposal of toxic metals because of higher attenuation and longer contaminant travel times. Arid sites also allow about 100 times greater leachate concentrations for stabilized waste compared to humid sites (e.g., Los Alamos National Laboratory vs. Oak Ridge Reservation). Most RCRA toxic metals are, in general, immobile in the subsurface environment, even at relatively humid sites. Exceptions to this conclusion are anionic species (e.g., arsenic and selenium), which tend to have lower Kd values. However, more specific analyses are needed to better determine the environmental fate of these and other metals.

From the limited available data on toxic metal concentrations in DOE MLLW and the estimated performance of grout-stabilized process waste, a margin of protection appears to be present in most cases when wastes containing RCRA toxic metals are disposed of at the evaluated DOE sites. Exceptions to this conclusion are As, Se, Cr and Hg when disposed at some humid sites. Waste streams containing large amounts of these elements may require enhanced stabilization (other than grout stabilization) prior to disposal, or relocation for disposal to semi-arid or arid sites.

The EPA's prescriptive approach of explicitly defining regulatory levels for toxic waste does not inherently account for the variety of disposal environments encountered nationwide, and may potentially result in either underprotection of groundwater resources at humid sites (e.g., TC levels) or an unnecessarily large margin of protection at arid sites (e.g., UTS levels).

ACKNOWLEDGMENTS

This work was sponsored in part by the United States National Academy of Sciences/National Research Council through the Collaboration in Basic Sciences and Engineering Program between the Eastern European countries and the United States.

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A STRATEGY FOR RCRA PERMIT COMPLIANCE

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ABSTRACT

The Resource Conservation and Recovery Act (RCRA) of 1976 established national requirements for safe management of hazardous wastes. All generators and transporters of hazardous waste and all operators of treatment, storage, and disposal (TSD) facilities, including federal facilities, are responsible for handling and disposing of that waste as prescribed by the act and its implementing regulations.

The objective of one program established by RCRA, under Subtitle C, is to ensure that hazardous waste is handled in a manner that protects human health and the environment. Under Subtitle C, any facility that treats, stores, or disposes of hazardous waste must obtain an operating permit. As part of the permit conditions, configuration and operating conditions cannot be changed without notification of or approval from the regulator. A change to a permitted facility requires a permit modification if it affects a specific permit condition.

Because of the large number and wide differences in the hazardous and mixed waste units managed at the U.S. Department of Energy's Idaho National Engineering Laboratory, it was necessary to develop and implement a new business practice to maintain compliance with its RCRA permit.

The new business practice, reviewing TSD facility changes for RCRA permit modifications, applies a six-step process to ensure permit compliance at INEL facilities. The review process includes permit applications under regulator review, as well as issued permits. Lessons learned from problems encountered during implementation have been used to refine the process, which can be applied to other permitted facilities.

The key to success in developing such a program is the involvement of all the affected organizations: document control, maintenance, engineering, training, and RCRA permitting. The program is easier to implement if the facility has an effective change control or configuration management program in place. Each facility should balance the risk of non-compliance with the cost of the permit review program to be established for the facility.

BACKGROUND

The INEL was established in 1949 as a center where nuclear power reactors and support facilities could be built, tested, and operated with maximum safety. The INEL site covers 890 square miles and is 22 miles west of Idaho Falls. Fifty-two reactors have been built at the INEL, and 12 are still operable. The primary mission of the INEL is to utilize its engineering and scientific capabilities, site, and facilities to support national defense and energy programs. Current major programs at the INEL include water reactor safety, materials and fuels testing, nuclear fuel storage, high-level waste calcining, breeder reactor operation and research, manufacturing of tank armor, and management of hazardous, low-level, transuranic, and radioactive mixed waste. A total of 53 hazardous or mixed waste units will be permitted. These units include container storage areas, tank treatment and storage systems, incinerators, and other facilities.

REQUIREMENTS FOR PERMIT MODIFICATION

At facilities operating under a RCRA permit, operating conditions and documents and equipment related to the permit cannot be changed without notification of or approval from the regulating agency. Changes at permitted facilities require permit modifications if conditions specified in the permit are affected.

The Environmental Protection Agency (EPA) developed regulations for permit modification to help owners and operators of TSD facilities comply with the regulations and to reduce the possibility of fines or civil or criminal prosecution for environmental compliance failures. Consistent identification of needed permit modifications is essential for permit compliance.

The process developed at the INEL to review facility changes for possible permit modification was designed to accomplish the following goals.

- maintaining configuration control of all items related to RCRA permits or permit applications
- processing RCRA permit or application modifications to meet regulatory time frames
- providing training and information, including identification of RCRA permit compliance items, to facilitate reviews

The review process included permit applications under review by the regulating agency as well as issued partial permits.

Owners or operators of facilities that treat, store, or dispose of hazardous waste are required to submit a comprehensive, two-part permit application that covers all aspects of the design, operation, and maintenance of the facility. Based on the application, the regulating

agency determines if the facility is in compliance with Subtitle C regulations and develops a facility-specific operating permit. The review by the regulating agency may take from one to three years.

During the term of a permit (usually 10 years), situations may arise that justify modifying, revoking and reissuing, or terminating the permit. The regulating agency may initiate revocation (and reissuance) or termination of a permit for noncompliance with the permit. The owner (or operator) or the regulating agency may initiate a permit modification to prevent noncompliance. Permit applications are modified during the review period to reflect any facility changes that occur before the permit is issued. The most common conditions requiring modifications to permits or permit applications are listed below.

Substantial alterations or additions are made to the facility.

New information about the facility becomes available.

New statutory or regulatory requirements affect existing permitted activities.

Regulations for modifying a permit, established in 40 CFR 270.40-43, provide a permit modification classification system and administrative procedures for permit modification, including interactions with the regulating agency and the public. In addition, the regulations authorize the regulating agency to grant temporary permits to allow facilities to respond promptly to changing conditions.

Within the state of Idaho, RCRA permits are administered and enforced by the Idaho Department of Health and Welfare (IDHW). The RCRA permit modification provisions are established in the Idaho Administrative Procedures Act (IDAPA) 16.01.05012, which adopts the EPA permit modification regulations.

RCRA permits, as defined in the federal regulations and IDAPA requirements, are issued by facility, and the IDHW has identified the INEL as a single facility, owned and operated by the Department of Energy Idaho Operations Office (DOE-ID). In practice, however, the permit requirements are applied to specific TSD units (also called facilities). TSD units may include containers, tank systems, surface impoundments, waste piles, land treatment units, landfills, incinerators, and miscellaneous units. At the INEL, a single permit is issued for the INEL as a facility, with partial permits issued for the individual TSD units. The partial permits are issued to DOE-ID, as the facility owner and operator, with the specific contractor for the TSD unit identified as co-operator.

RCRA PERMIT MODIFICATION CLASSES

To facilitate classifying RCRA permit modifications, the EPA identified foreseeable changes at a facility that would require a permit modification and designated a modification class for each. The identified changes and their designated classes are incorporated into the regulations at 40 CFR 270.42, Appendix I, Classification of Permit Modification.

If the proposed change is not listed, the reviewer may propose a modification class based on established class definitions and similarity of the proposed modification to previously classified modifications. The modification may be submitted as a Class 3 modification or a request for the IDHW to determine the modification class based on the proposed class. The regulations classify modifications into four types: class 1, class 1 requiring approval (class*1), class 2, and class 3, as shown in Table I.

Temporary modifications may be allowed for class 2 or 3 permit modifications for 180 days or less. The advantage to temporary modifications is that public interaction is not required.

Table I

TYPES OF FACILITY CHANGES

During the operation of TSD facilities at the INEL, waste management practices will be adapted, as needed, to keep pace with both technological and regulatory developments. Many of these adaptations will require changes to conditions specified or material incorporated by reference in a RCRA partial permit or permit application.

The following conditions require a RCRA permit or permit application modification.

- changing TSD unit design, operation, capacity, or waste types processed if RCRA permit or permit application specified conditions are affected
- transferring the TSD unit to a different owner or operator
- managing newly regulated wastes or waste management units

Changes associated with the TSD unit ownership or waste management regulations are unusual and can be anticipated by personnel managing RCRA permit compliance. Changes associated with the TSD unit design, operation, capacity, or waste type processed are the target of the RCRA permit modification review process. Although each facility has its own mechanisms for initiating changes, three types of change methods apply to most permitted facilities: 1) work orders for maintenance and modifications, 2) document changes, and 3) projects.

Work Orders

Work orders are used at the INEL to implement physical changes to TSD units or waste management processes. As part of the review and approval process, work orders are reviewed for needed RCRA permit or permit application modifications. Permit modifications originating from work orders will most likely be class 1 changes.

Document Changes

Document changes, used to implement changes to operating practices or designs for a TSD unit, may include generation of new documents or revisions to or cancellation of existing documents (procedures, manuals, lesson plans, process run plans, databases, forms, plant drawings). Reviews for RCRA permit or permit application modifications are part of the normal review and approval process for document changes at the INEL.

Project Changes

Projects are used to implement large-scale physical changes at a TSD unit, including construction of a new TSD unit. New projects or modifications to existing projects are reviewed for RCRA permit or permit application modifications as part of the development of the environmental checklist required for federal facilities by the National Environmental Policy Act. Project changes are the most likely to require class 2 or higher permit modification requiring approval from the regulating agency.

GENERAL APPROACH TO PERMIT COMPLIANCE

Any change at a TSD unit that affects specified conditions or referenced information in a RCRA permit or permit application requires a modification to the permit or application. Some changes may not affect compliance with applicable RCRA permits or permit applications. For example, an item of equipment whose manufacturer name and model number is specified in a RCRA permit or permit application can be replaced with an identical item (the same manufacturer name and model number) without affecting compliance with permit or permit application conditions.

However, if the same item of equipment is replaced with an item from a different manufacturer, a permit or permit application modification will be required. Permit modifications, which can be both time-consuming and costly, should be requested only when necessary or when they can be considered with another modification. The RCRA permit modification review process is used to confirm that modifications will be necessary.

To ease adjustment to this additional review process for TSD units at the INEL, a phased approach was used for implementing the RCRA permit compliance program. The basic goal of the phased approach was to maintain permit compliance while minimizing impacts and costs by using the existing infrastructure. Early compliance was made possible through a pilot phase to implement and evaluate trial-use procedures and training. Long-term compliance was made possible by issuing final procedures, distributing process and instrumentation diagrams showing hazardous waste equipment, and issuing RCRA document, instrument, and equipment lists on a controlled basis. Wherever possible, existing procedures and practices were adapted to implement the RCRA permit modification reviews.

Because a large number of changes are processed at the INEL, reviews for RCRA permit modifications need to be performed in accordance with the schedule for implementing the change and the overall priority of the change. After trained reviewers determine the need for RCRA permit or permit application modifications, permit applications can be updated in conjunction with implementation of the change, preventing any backtracking later. For modifications to issued permits, the permit modification process can be initiated as quickly as possible, preventing unnecessary delays in implementing the change.

BASIC STEPS FOR IDENTIFYING PERMIT MODIFICATIONS

The basic steps for determining whether a proposed change requires a permit modification are essentially the same for all types of changes.

1. Determine whether the change will impact any permit or permit application condition.
2. As applicable, determine the permit modification class.

The steps of the RCRA permit modification review are shown in Fig. 1.

Fig.1

IMPLEMENTING THE COMPLIANCE PROCESS

The initial steps to implement the program for RCRA permit compliance included 1) identifying permit conditions for compliance, 2) establishing procedures to implement the program, 3) developing a training program, and 4) developing the reviewer resources needed.

Identifying Permit Conditions for Compliance

Permit application writers reviewed the permit applications and the issued partial permits to identify the possible permit conditions for development of environmental requirements. The lists of conditions they developed included the application or partial permit, volume, section or module, revision, specific requirement, and implementing document for each identified condition.

As the final partial permits are issued, the lists will be revised to reflect changes.

Establishing Program Procedures

During the pilot phase, the trial-use procedures were based on current procedures for processing work orders, document changes, and projects. Supporting forms were developed from forms currently in use.

When the RCRA permit modification review program became effective, existing change control procedures were revised to include the permit

modification review, and supporting forms were revised to include a block for documenting review results.

A RCRA permit modification review flow diagram (Fig. 1) was developed to help the reviewing organizations decide if changes required permit modification. The review results are documented on the change proposal documents or a separate RCRA permit modification review form. Figure 2 shows the block used to record RCRA permit modification review results on INEL change forms.

Fig. 2

The only difference in the review and approval process for changes that do not impact permit conditions will be documentation to indicate that the change has been reviewed and found to have no impact.

For changes that do impact a permit condition, the reviewing organization will request that RCRA permitting personnel prepare the required permit modification notice or request. If the reviewing organization cannot determine the impact on permit conditions, RCRA permitting personnel may be asked to make the determination.

Training

Training was designed to be consolidated as much as possible to avoid duplication and to target the individuals who need to perform RCRA permit modification reviews. Two types of training were developed at the INEL: 1) performance training for reviewers who will determine if proposed changes impact RCRA permits or permit applications and 2) awareness training for individuals who need to know about RCRA permit modification reviews and the impacts on their work activities.

The 7-hour reviewer training includes a RCRA overview, study of the RCRA permit organization and the permit modification classes, a session to work through facility-specific examples, and written and performance tests. Personnel passing the class become qualified RCRA permit modification reviewers; personnel may exempt the class by challenging and passing both the written and performance tests. Reviewers will be retrained (or requalified by testing) every two years.

Awareness training was implemented by sending a letter of information sent to all TSD unit managers, document control, work control, and engineering personnel, project managers, and other personnel who could initiate or process changes that could be related to a RCRA permit or application. The letter addressed facility-specific change methods, provided awareness training in a question and answer format, and listed contacts for additional information or individualized training.

Reviewer Resources

A RCRA permit modification reviewer guide was developed and issued to each reviewer as an aid for determining if changes will impact permits or permit applications. The guide includes resources developed to assist reviewers in identifying permit-compliance items.

Copies of the issued partial permits and the permit applications prepared for TSD units were made available at convenient locations. In addition, electronic versions of the partial permits and permit applications were made available on an automated image management system and the Internet. Documents included in or referenced in a RCRA permit or permit application are listed by number on a document list. The list provides the specific volume and section in the permit or permit application where the documents are referenced or included. The document list is controlled and distributed to the reviewers.

Plant equipment and instruments identified in a RCRA permit or permit application may be located by number on equipment or instrument lists. Controlled copies of the equipment and instrument lists were issued to the reviewers in the reviewer guide.

Equipment and instrument lists are available through existing maintenance and engineering databases. The lists can be accessed in the equipment catalog under the main menu for each system. Equipment identified in a RCRA permit or permit application is indicated as "RCRA" in the environmental permit field.

Plant piping (waste lines) and large equipment (tank, vessels, etc.) identified in a RCRA permit or permit application are marked on applicable piping and instrumentation diagrams (P&IDs) with the following indications: see insert

Reviewers were instructed to assume that any equipment and instruments on the line, in addition to the equipment marked on the P&IDs (including pumps, valves, and jets), are also RCRA-identified equipment.

Regulatory interpretations and other information to clarify regulatory language or resolve problem areas in the review process were provided to reviewers as they were developed.

APPLICABILITY AT OTHER FACILITIES

This program could be adapted for use at other larger TSD facilities that will be operating under a RCRA permit. A formal program similar to the INEL program is necessary for larger TSD facilities. Small TSD facilities can use a single point of contact, familiar with the permit, to review all the change requests.

The key to success of developing such a program is the involvement of all the affected organizations: document control, maintenance, engineering, training, and RCRA permitting. The program is easier to implement if the facility has an effective change control or configuration management program in place. Each facility should balance the risk of non-compliance with the cost of the permit review program to be established for the facility.

BASIC ELEMENTS FOR AN EFFECTIVE PROGRAM

Evaluating the development and implementation of the RCRA permit compliance program at the INEL provides a list of basic steps and decisions that apply for similar programs.

1. Obtaining senior management approval. To obtain senior management approval for implementing the permit compliance program, the planned program must be demonstrated as workable, acceptable, and thorough.

2. Identifying permit requirements and compliance items at the facility. Specific requirements and references (RCRA permit or application, module or section) from the applicable partial permit or permit application should be identified. Structures, systems, equipment, equipment configurations, and documentation related to those permit requirements should be identified as compliance items, if changes to the items could affect the compliance status of the TSD unit. Corresponding implementing documents (procedures, drawings, databases, manuals) should be specified.

3. Identifying change methods in use at the facility. Using the permit compliance conditions identified from the permit or permit application, the methods used for making changes to those items can be identified. RCRA permit modification reviews should be incorporated into the change methods to ensure that all changes that could affect a permit or permit application are reviewed and tracked.

4. Establishing program procedures. Specific procedures for accomplishing RCRA permit modification reviews for facility changes should be established, with the following elements.

- Timely reviews: the RCRA permit compliance review should occur early enough in the process for any required notifications to be made.
- Documentation of reviews: the review for permit compliance should be documented on the change approval documents or attached forms or on electronic change documents.
- Implementing the program for different change methods: permit compliance reviews should be adapted for different change methods, including complex work orders and simple document revisions, which vary from facility to facility. Because of the complexity of the INEL, implementing the program and addressing site-wide change methods is presently in process.
- Implementation of class 1 modifications: because the regulating agency must be notified within 7 days after the changes are in effect, procedures for processing class 1 permit modifications must be specific.

5. Identifying and training reviewers. A pool of reviewers familiar with the RCRA permit or application should be established to evaluate proposed facility changes.

- Identifying the reviewers: personnel involved with management or change processes at RCRA-permitted facilities should be assigned and trained as RCRA permit modification reviewers. A listing of qualified reviewers should be available to personnel who normally make facility changes.
- Comprehensive training: training should be developed to accommodate personnel unfamiliar with RCRA and personnel with different levels of knowledge.

6. Providing reviewer resources. Resources to facilitate RCRA permit modification reviews of facility changes should be provided to the trained reviewers. Review resources include document and equipment lists and copies of the permit or permit application. Identifying RCRA equipment and instruments on P&ID's flags the RCRA-related items for reviewers and facility and engineering personnel.

7. Assessing program effectiveness. A built-in system for feedback and program assessment provides information about the review program and its effectiveness.

Session 27 -- LOW LEVEL AND MIXED WASTE VITRIFICATION
Co-chairs: Carol Jantzen, WSRC

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VITRIFICATION OF TRU-CONTAMINATED BURIED WASTE: RESULTS FROM RADIOACTIVE DEMONSTRATIONS AT TARANAKI

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ABSTRACT

The Maralinga Nuclear Test Range, located in South Australia, is a former nuclear weapons test site that was used by the British in the 1950's and early 1960's. Both nuclear detonations (major trials) as well as chemical detonations of warheads (minor trials) resulted in extensive contamination of the site. At Taranaki, Maralinga's most heavily contaminated area, a series of minor trials involving the explosive dispersal of plutonium and uranium resulting in extensive contamination of surface soil and generated massive quantities of contaminated debris. The heavily contaminated debris from the trials was subsequently buried in a series of shallow pits at Taranaki.

The Commonwealth Government's Department of Primary Industries and Energy (DPIE) is undertaking a program to rehabilitate the most heavily contaminated areas at the site. A major part of the program is directed to reduce the risk presented by the contaminated debris buried in the pits at Taranaki. DPIE has identified the in situ vitrification (ISV) technology as the preferred technology for treatment of the Taranaki Pits. As part of this program, Geosafe recently completed two multi-ton radioactive demonstrations of the ISV technology at the site. The demonstrations involved preparing test pits which included 37 wt% steel, and other debris including lead, baryte shielding bricks and organic-based materials. Actual plutonium-contaminated debris originating from the original weapons tests was used in one demonstration and each demonstration involved the vitrification of one kg of uranium oxide. Results indicate that all demonstration objectives were met and that >99.999% of the radioactive materials were retained in the melt. No detectable activity was found inside the off-gas containment hood or on the insides of the off-gas piping. Preliminary radiochemistry analyses and X-Ray Fluorescence analyses indicate that the radioactive materials are uniformly distributed through the vitreous product. Leach tests of the vitrified product using the Product Consistency Test procedure at 7 and 28 day leaching intervals indicate that the normalized leach rates are extremely low (<0.1 g/m2) for all oxide species.

This international application of the ISV technology on TRU-contaminated buried waste represents a major milestone in the deployment of the DOE-developed ISV technology. This paper will present a overview of the Maralinga Rehabilitation Program and discuss the two radioactive ISV demonstrations conducted at the site. In addition, plans for the remaining phases of work will be discussed.

THE MARALINGA SITE

Atomic weapons were developed and tested in Australia at Maralinga by the British Government from 1955 to 1963. Seven atomic explosions during 1956 and 1957 resulted in fission product fallout. Several hundred ancillary experiments were conducted, some of which involved explosive dispersion or burning of metallic plutonium, uranium and beryllium in the open environment. Weapons development ceased in 1963, following the Partial Test Ban Treaty. Several attempts at clean-up of the Maralinga site were made by Britain. The last was Operation Brumby in 1967, during which

contaminated areas of soil were plowed to mix and dilute the level of surface contamination, and debris pits containing plutonium were capped with concrete. The site then reverted to Australian control. Details of the operations at Maralinga were summarized by Symonds (1). Interest in rehabilitation of the site was revived in 1984 by the Australian Royal Commission into British Nuclear Tests in Australia. The recommendations of the Commissioner (2) included a further clean-up to permit unrestricted access of Aboriginal people to the former test sites.
Maralinga is situated in the State of South Australia, between the Nullarbor Plain and the Great Victorian Desert, 40 km north of Watson siding on the Trans Australia Indian Pacific Railway (Fig. 1). The area of the site is 3,210 km2. Maralinga has a semi-arid environment with an average of 200-mm annual rainfall. Average temperatures range from 33C in January to 18C in July, with summer temperatures frequently in excess of 40C. The weapons development tests were conducted on Tietkens Plain, an outcrop of limestone and dolomite, partly covered by sand and bordered by vegetated sand hills.
Fig. 1
THE MARALINGA REHABILITATION PROGRAM
The Technical Assessment Group (TAG) was established by the Australian Government in February, 1986, to conduct scientific and engineering field studies, laboratory research and pilot operations necessary to define a range of realistic and cost-effective rehabilitation options. The scientific studies commenced with an aerial radiological survey of the former test sites, and included field and laboratory work to assess concentrations of residual radioactive isotopes in native foodstuffs, soils and inhalable dusts at Maralinga.
Dosimetric modelling of potential radiological dose through the pathways of ingestion, inhalation and wound contamination during the activities of a semi-traditional Aboriginal lifestyle led the TAG to conclude that the current radiological hazard at Maralinga resulted from the dispersal by chemical explosive of about twenty two kilograms of metallic plutonium in twelve Vixen "B" one-shot minor trials at the Taranaki test site between August 1960 and April 1963. In these trials, each nuclear device was detonated by chemical explosive on an exposed "featherbed" structure. The "featherbed" consisted of massive steel plates and walls of lead and barytes bricks mounted on rolled steel joists. The detonation of the devices produced a measurable, but negligible, nuclear energy yield in most shots. Plutonium was dispersed as fine oxide dusts, as sub-millimeter particles, and as surface contamination on larger fragments of debris from the destruction of the "featherbed". Engineering studies by the TAG defined a series of engineering work packages using established technologies for treatment of contaminated land and debris pits.
REMEDIATION OPTIONS
The report by TAG (3) to the Australian Government contained nine rehabilitation options and about 29 sub-options. The options ranged from low cost/resource/risk (e.g., fencing and exclusion of contaminated areas) to high cost/resource/risk (e.g., collection and disposal of contaminated soil and the contents of debris pits). The scope of rehabilitation covered access by semi-traditional Aboriginal Communities, primarily the residents of Oak Valley, ranging from casual access to fully unrestricted habitation.
Data from safety trials conducted at the Nevada Test Site (4) indicated that about twenty percent of the plutonium detonated (i.e., about four

kilograms of plutonium) might have been deposited in the near field of the detonations. Twenty one numbered shallow debris pits in a fenced area at Taranaki have been reported to contain about 820 tonnes of debris and 1150 tonnes of soil contaminated with plutonium from the Vixen "B" trials. Pending further evidence, the pits have been assumed to contain between four and twenty kilograms of plutonium. This paper is focused on the stabilization of the contents of these debris pits.

STABILIZATION OF DEBRIS PITS

TAG (3) considered three options for stabilization of the plutonium debris pits: exhumation and reburial of the pit contents, stabilization by concrete grouting, and stabilization by in situ vitrification. Exhumation processes considered involved excavation of debris and contaminated soil into rectangular steel boxes, for disposal either in deep boreholes, or in a lined and capped sub-trench below the trench for disposal of collected contaminated surface soil. Exhumation would require intensive radiological protection of personnel against inhalation of plutonium dusts.

Grouting procedures assessed included a combination of in-pit grouting, grouting of adjacent rock and soil, with concrete capping, cut-off walls and tumuli over the grouted pits. A major problem in the grouting option was the uncertain degree of void filling and consequent doubtful degree of improvement in long-term safety.

In situ vitrification (ISV) is a U. S. Department of Energy developed process being commercially applied by Geosafe Corporationa. The process involves electric melting of contaminated soil and debris and/or other earthen materials for purposes of permanently destroying, removing, and/or immobilizing hazardous and radioactive contaminants. Melt temperatures typically reach 1400-2000C by passage of (typically) 3 to 4 MW of electrical power with a square array of four electrodes. Off-gases are collected for treatment in a steel containment hood that spans the area being processed. When electrical power is shut off, the molten mass solidifies into a vitrified monolith with unequalled physical, chemical, and weathering properties compared to alternative solidification/stabilization technologies. For the Maralinga application, the ISV process would melt the soil and debris contained in the pits. The plutonium oxide would be incorporated into a stable leach resistant vitreous/ceramic block, with steel debris melting to form an encapsulated steel ingot.

The ISV process appeared to have advantages of improved occupational, public, and environmental safety together with greatly improved containment of the radioactive materials in the vitrified product that would be much more durable compared with alternative stabilization methods. This conclusion was subject to the proviso that the presence of limestone and the contents of the pits did not adversely affect process efficiency and that the logistics for operation of the process at Maralinga could be resolved. The Australian Government decided to proceed on the basis of an option which involved collection and trench burial of the more highly contaminated surface soil, and determination of the applicability of ISV for stabilization of the contents of the debris pits.

DESCRIPTION OF THE MARALINGA ISV PROJECT

The ISV project was structured as a four phase project. Phase 1, conducted in 1993 and 1994, involved an initial study to determine if the ISV process was suitable for the application. The study included a site

visit to evaluate the site conditions and involved engineering-scale ISV tests and crucible melt studies using debris and uncontaminated soils from the site. Phase 1 results indicated that the ISV process could be applied to the soil and debris combinations at the site.

Phase 2, conducted in 1995, involved a series of ten on-site engineering-scale tests and three intermediate-scale demonstrations to obtain site-specific process data. Two of the intermediate-scale demonstrations used radioactive materials, including blast debris from the original weapons tests. These two radioactive demonstrations are the subject of this paper. The principal goal for the intermediate-scale radioactive ISV demonstrations was to collect sufficient data to determine if the ISV process could be expected to effectively treat the contaminated soil and debris in the Taranaki pits and to obtain data to confirm the behavior of plutonium in the process. Data from the tests and demonstrations were also gathered to support the design of a full-scale ISV process machine that will be tailored for the site-specific conditions and to develop a remedial design plan which will define the approach and logistics associated with the full-scale treatment at the site. Specific objectives were established for each demonstration so that the performance of the ISV process and the resulting vitrified product could be evaluated against the performance criteria established for the project.

Science and engineering advisors representing the Commonwealth helped determine ISV process performance criteria for the application and were present to observe activities during key stages of the demonstration project.

The demonstrations were configured in a manner that was thought to best represent the configuration of the actual pits as well as the actual types and amounts of debris buried in the pits. Standard scaling relationships established for the ISV process were used in conjunction with historical data that describes the pits and the pit contents to develop scale mock-ups of a typical Taranaki pit.

An intermediate-scale (85 kW) system capable of producing melts up to 4,500 kg (5 tons) was constructed for the project. Figure 2 is a photograph of the ISV equipment as positioned for the second radioactive demonstration involving plutonium. This size of system provides cost effective data that can be directly scaled to the full-size application. The off-gas treatment system was designed specifically to handle the higher off-gas generation rates and higher off-gas temperatures expected to result from processing buried wastes. In addition to the steel and radioactive materials, the pits contained significant amounts of gas generating materials such as sulfates, carbonates, and organics.

Fig. 2

RADIOACTIVE ISV DEMONSTRATIONS

The two radioactive demonstrations involved the treatment of soil, 37 wt% steel debris, and other debris including bitumen-stabilized soil, lead, plastic, electrical cable and barytes bricks. The barytes bricks were originally used as radiation shielding material and are composed of barium sulfate. Figure 3 is a photograph of one of the pits being filled with debris and soil. One kilogram of uranium oxide was buried in each pit to serve as a surrogate for plutonium. For each demonstration melt, the uranium oxide was contained in a plastic bag and located in the center of the pit to serve as a highly localized area of contamination. The second radioactive demonstration included a steel plate, originating from the weapons tests, that was contaminated with approximately 0.5

grams of plutonium oxide (predominantly 239Pu with about 3% being 241Pu). About 90% of the 241Pu originally on the plate had decayed to 241Am.
Fig. 3
Each demonstration melt was conducted at opposite ends of a trench. In order to best represent the geochemistry of the limestone-based soil surrounding the Taranaki pits, the tests were conducted in the Taranaki area adjacent to two of the larger waste burial pits.
The two demonstrations were conducted in September and October of 1995. The first demonstration occurred over an 84 hour time period while the second demonstration occurred over a 96 hour time period. During the operations, process-related data, such as electrical power and off-gas related data, was collected to support the design process for a full-scale ISV machine that will be tailored specifically for the site.
Following the two demonstrations, the resulting vitrified monoliths were excavated for examination, weighing, and sampling. The mass of the first demonstration monolith was determined to be 3,766 kg (4.15 tons). The mass of the second demonstration monolith was determined to be 4,292 kg (4.73 tons). Figure 4 is a photograph of the second demonstration monolith being weighed.
Fig. 4
RESULTS AND OBSERVATIONS
Both demonstrations were completed successfully. Physical characterization of the vitrified blocks and preliminary radiochemical analyses have been completed. Additional analyses, including a variety of leach tests, are currently underway. Based on the available data, the following observations and conclusions can be made concerning the demonstrations:

The ISV process was demonstrated to be capable of melting the soil and debris combinations in the pit including the 37 wt% steel. In addition, the non-steel debris in the pit (barytes bricks, cable, lead, bitumen stabilized soil, and plastic) did not pose any processing difficulties.

The voids and gas generating materials in the pits (carbonates, sulfates, and organics) did not pose any processing difficulties with respect to off-gas containment. The off-gas treatment system's high off-gas flow rate was fully sufficient to accommodate the high steady state off-gas generation rates and transient off-gas surges that resulted from processing the gas generating materials and voids.

The volume reduction for the soil and debris treated was 47% for the first demonstration melt and 55% for the second demonstration melt.

Based on isokinetic off-gas sampling, the amount of uranium oxide retained in the first demonstration melt was 99.99987% and the amount retained in the second demonstration melt was 99.99968%. Using the same isokinetic off-gas sampling methods, the amount of plutonium retained in the second demonstration melt was determined to be in excess of 99.99999%.

Following the demonstrations, health physics-related surveys of the equipment established that the insides of the off-gas containment hood, off-gas piping, and primary HEPA filters were free of detectable contamination above background levels (less than 0.25 Bq alpha and beta combined per 100 cm2 surface area). Consequently, decontamination of the equipment was not required.

The plutonium, uranium, and americium in the vitreous phase are not smearable. Significant intrusive sampling activities resulted in the creation and handling of many small fragments of vitrified product,

including dusts, but did not result in the transfer of any detectable contamination to tools or personnel.

Based on X-Ray Fluorescence analyses and alpha spectrometry analyses of samples, the plutonium and uranium oxides were found to be uniformly distributed throughout the vitreous phase. Table I provides a summary of the data for several samples from the plutonium demonstration.

The metal phase at the base of each melt was determined to be free of plutonium and uranium based on qualitative analyses. Quantitative analyses of the metal phase have not yet been completed. (Earlier Phase 2 tests involving cerium as a plutonium surrogate established that the cerium did not partition to the metal phase.)

Leach tests of six samples of vitrified product, conducted in triplicate, using the Product Consistency Test Procedure at intervals of 7 and 28 days indicate that the normalized leach rates for all oxides in the vitrified product are less than 0.1 g/m2. The leach tests included standard PCT tests as well as modified leach tests using leachants with pH values of 5, 7 and 10.

Table I

CONCLUSIONS

The data and observations resulting from the radioactive ISV demonstrations conducted at the Maralinga site support the following primary conclusions concerning the likely performance of the ISV process on the Taranaki pits:

The ISV process, at full-scale, can be expected to effectively treat the soil and debris combinations in the Taranaki pits.

The data indicates that an ISV process machine designed specifically for this application will be capable of handling the higher off-gas temperatures and transient off-gas flows associated with the treatment of the buried wastes.

The vast majority (>99.999%) of the plutonium and uranium will be retained in the melt and will be uniformly distributed throughout the vitreous phase.

The vitrified product will be a uniform, dense, hard product of high strength with exceptional leach resistance.

Plutonium will not be distributed to any significant extent to other phases in the melt.

The ISV process can be safely applied to the materials present at the Taranaki site.

PLANS FOR SUBSEQUENT PHASES OF WORK

The two radioactive demonstrations provided an opportunity to obtain site-specific process performance data to evaluate the ISV process for this application. The data will be used to develop a remedial design plan for the full-scale application to determine the most efficient, safe and economical approach to treat the Taranaki pits with the ISV technology. In addition, the data is being used to design a full-scale ISV process machine that is being tailored to accommodate the specific characteristics and treatment requirements of the site. Phase 3 will involve the construction of the full-scale ISV machine. Phase 3 is expected to commence in 1996. Phase 4 involves the actual treatment of the Taranaki pits. Phase 4 is expected to commence in 1997.

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VITRIFICATION OF INTERMEDIATE, LOW-LEVEL RADIOACTIVE AND TOXIC WASTES WITH A COLD CRUCIBLE

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ABSTRACT

In the framework of a cooperation agreement between SIA "Radon" (Russia) and SGN (France), the cold crucible technology developed by Radon has been tested with two different simulated wastes. These surrogates are representative of VVER (nuclear reactor of water-water type used at Nuclear Power Plants) and Radon waste respectively. The operating conditions, the performance of the cold crucible, the off-gas treatment, and properties of the produced glass have been determined.

Moreover, joint vitrification of low-level liquid wastes with solid wastes and crushed lead-containing cathode ray tubes (CRT) glass were conducted. Glasses produced from sodium nitrate or sodium borate based wastes and lead-containing waste can be melted at 1200-1300oC and these are high chemically durable. Their viscosity range is between 2 and 7 Pa s. These glasses, with resistivity 0.02-0.05 Wm, are able to be produced by induction melting in a "cold crucible". When a "yellow phase" forms, it may be dispersed in molten glass by high active agitation followed by a rapid temperature decrease to solidify the "yellow phase" (sulfates) in the glassy matrix. Vitrification tests in the experimental plant based on cold crucible were performed and the operating conditions and the glass properties were determined.

Experimental results have shown that lead glass is a suitable additive for radwaste vitrification. Joint vitrification of radwaste with lead-containing waste yields glass or glass-composite material with low leach rate both alkali cations and lead.

The next milestone of the joint development program to reach industrial scale-up are also described.

INTRODUCTION

Successful application of the cold crucible technology at SIA "Radon" for intermediate-level waste vitrification (1) stimulated a search of new applications, including the use of the cold crucible for lead-containing toxic waste immobilization (2). In the frame of a cooperation agreement between SIA "Radon" (Russia) and SGN (France), the cold crucible technology developed by Radon has been tested with two different simulated wastes. These surrogates are representative VVER and RBMK (channel-type uranium-graphite reactor) wastes.

The tests were conducted in an experimental plant based on coreless induction melter - cold crucible with industrial-scale dimensions.

The main aims of these tests are characterization of this technology with simulated intermediate-level Radon and SGN specified wastes, determination of basic process parameters, heat and mass balance, product, off-gas and secondary waste characterization and determination of the ability of cold crucible to maintain an underpressure at operation.

Glasses prepared were analyzed for chemical composition, viscosity, resistivity, chemical durability, and homogeneity. Aerosol concentration and chemical composition, gas specific activity, components volatility and gas concentration (HCl, SOx, COx, NOx) have been determined from off-gas analysis. The estimation of sleeve filter workability (operation efficiency and regenerability) was carried out as well.

Experiments on the joint vitrification of low-level liquid and solid wastes and crushed lead-containing CRT glass were also conducted. The same cold crucible based plant was used. Glass properties, lead volatilization and leaching were studied.

EXPERIMENTAL

Plant Description

The basic unit of the plant design (Fig. 1) is an induction melter - a cold crucible (IMCC). The melter was manufactured from water cooled stainless steel pipes. It has a water-cooled lid. An agitator may be installed in the lid if necessary.

Fig. 1

The crucible was installed in a protective access box, equipped with exhaust ventilation and filter. The cold crucible dimensions during test No. 1 and 2 were 590 mm in length, 300 mm in width, 655 mm in height and 10.18 dm2 in melt surface area. The crucible was surrounded by a two-turn inductor energized from a high-frequency (HF) generator. The generator' s power and operating frequency were 160 kW and 1.76 MHz respectively. The plant is equipped with water flow rate indicators. Generator's lamp was cooled with distilled water.

Tests No. 3-5 with lead glass were performed with small cylindrical cold crucible (100 mm in diameter) energized from HF generator operated at 1.76 MHz with 60 kW of power.

Batch preparation was carried out manually in containers. The batch prepared was charged in a mixer followed by feeding into the melter. Glass melt was poured into metallic container (measures 400x400x150 mm). Off-gas was purified on coarse (sleeve) filter and cooled in a heat-exchanger.

Experimental Procedure

Five vitrification tests were performed. During the first test a simulated SGN waste was vitrified and during the second test a simulated "Radon" waste was treated with borosilicate additives. The salts were crushed previously to be undersized of 10 mm and dry blended to glass forming additives. SGN and Radon waste compositions are presented in Table I.

Table I

The glass-forming additive compositions including loam clay and CRT glass are given in Table II.

Table II

Dry batches compositions for the resulting glasses are shown in Table III. Before the first and the second tests water was admixed to batch to produce a paste with moisture of about 20 wt.%. The prepared paste was fed into the mixer in 20-22 kg batches. The batch capacity was evaluated by change of level in mixer taking into account its density.

Table III

Initiating of melting in the cold crucible was produced by means of heating a short-circuited loop (titanium ring) in an electromagnetic field followed by the titanium ring oxidation and incorporation of titania in the glass.

After the melt formation the batch was fed into the cold crucible in batches of 0.5-1 kg. A need for charge was determined visually by measure of melt surface opening in the crucible. An average melt capacity was evaluated by measuring the weight of the containers filled with glass. The current, voltage of feeding supply line, generator's electric parameters, flow rate and temperature of cooling water, underpressure in the cold crucible and filter resistance were checked every 20 minutes during the experiment. The glass samples were taken at the cold crucible discharge as well as from the containers after cooling. The condensate after heat-exchanger was drained and studied. During the first test the vitrification of SGN waste vitrification. The experiment was carried out for 16 h. Sampling was started after the operation conditions hit and was carried out in accord with feed cycle.

During the second test the batch contained simulated "Radon" waste. The experiment was conducted for 7 h. Two hours of this time a batch containing the radionuclide 137Cs with total activity of 5.5 MBq was vitrified. Off-gas sampling was carried out at filter outlet to exclude the effect of water soluble constituents of aerosols.

The next three experiments were performed to determine a feasibility and conditions of lead-containing toxic waste vitrification. At first, broken CRT glass without additives was remelted. Then, joint vitrification of Radon waste and lead glass was performed. Finally, joint vitrification of SGN waste, soil and lead glass was carried out.

During all of the experiments the underpressure of 15-20 mm H2O was maintained, the off-gas flow rate was checked and filter inlet and outlet temperatures were recorded. Simultaneously the gas flow rate and the filter resistance were recorded. A filter regeneration was carried out by transitory (0.2 s) pulse air feed in every sleeve by turns in two cases: when filter resistance reached 1.5 kPa and at instant of glass pouring. Glasses, off-gas deposits and condensates produced were analyzed for constituents by atomic absorption spectroscopy (S-115 spectrophotometer, Russia) and emission spectral analysis techniques (unit based on ISP-30 spectrograph). A glass resistivity and viscosity were measured with Institute of Glass (Russia) unit modified in Radon. Leach rates of sodium

and potassium ions and 137Cs were measured by IAEA technique (MCC-1 test) (3).
Specific activity of glasses containing 137Cs was measured by counter SI-8B supplied with device PSO-2/4 in comparison with standard source.
Incorporation of Simulated SGN and Radon Wastes in Borosilicate Glass
Before the first test the melter was filled with glass that remained from the previous experiments. Before and after the operation the glass level was set the same this level, corresponded to the minimum possible level of glass left after the glass pouring. Average process variables are shown in Table IV.
Table IV
The generator power was elevated as far as glass level in melter was elevated. Glass properties are given in Tables V.
Table V
In the middle of the first test some problems occurred with the glass sampling due to a change in glass composition that resulted in the increase of glass resistivity (4). "Yellow phase" formation due to high sulfate content was also observed. An attempt to disperse the "yellow phase" by means of agitation before pouring was not successful because of low melt viscosity.
No problems were encountered with Radon waste vitrification. Glass compositions are given in Table V. These are similar to common Radon waste glass (5).
At 1200-1300oC both melts had a suitable viscosity and fluidity, which all the melt pouring to be implemented for 2-3 minutes.
The effect of melt agitation on melter capacity was estimated during Radon waste vitrification. Melt agitation was carried out by blade stirrer installed at an angle to melt surface. A 10 kg batch was fed in the mixer and the time for every batch treatment was determined. Two batches were doped with 137Cs. According to preliminary data, agitation increased glass productivity by approximately 15-20%.
The glasses prepared had a low specific activity. Therefore a significant error occurs at 137Cs volatilization and leach rate measurements. Volatilization and chemical durability of glasses have been evaluated more precisely from volatilization and leaching of sodium and potassium cations. It is seen from data shown in Table IV that volatilization during melting and leach rate of these cations from glasses are similar to other known borosilicate glasses. Approximate measurements showed cesium leach rate has the same order of magnitude as sodium and potassium.
Solubility of SGN and Radon waste glasses in HF indicating vitreous phase fraction is 90% and higher (Table V).
Incorporation of Mixed Waste in Glass
Melting of broken CRT glass yielded a homogeneous melt containing minor inclusions of metallic contacts. Their low content did not result in an accumulation of molten metal phase at the crucible bottom. Process variables are represented in Table IV.
Glass compositions of the joint vitrification were determined from a laboratory study (2). Two of the mixtures (Table III) were fully melted at 1200oC. Melt resistivity at this temperature ranged between 0.02 and 0.05 Wm. This range is suitable for electric melting in the cold crucible. Sodium, potassium, lead and boron losses were measured (Table IV). Glasses (Table V) contained about 30% of Radon and SGN waste respectively. No problems were encountered during melting. Lead-

containing glasses were able to dissolve an increased quantity of sulfate and chloride ions (6) and "yellow phase" formation was not observed. Homogeneous glasses with good chemical durability were produced. Sodium, potassium and lead leaching was insignificant (Table V).

Off-gas Analysis

Cesium and other components losses from the melter were determined from off-gas specific activity and dust content taking into account the off-gas flow rate.

After the first run with simulated SGN waste, 3.1 kg of dust was collected in sleeve filter as well as 200 g of dust was deposited on off-gas pipe walls. Chemical composition of off-gas pipe and sleeve filter deposits are given in Table VI.

Table VI

Relative fraction of insoluble compounds (calcium, magnesium, aluminum and iron silicates and sulfates) from the off-gas pipe was higher compared to the product discharged from the collector and filter sleeves.

A sleeve filter was charged with filter cloth from carbon fiber with a filter surface area 2.8 m2. An average off-gas flow rate during the test time was 50 m3/h (recalculated to normal condition). Inlet temperature was maintained at the level 130-160oC and filter outlet temperature was 90-120oC. Gas filtration flow rate at filter operation was 0.8-1.0 cm/s. Filter resistance ranged between 0.3 and 1.5-2 kPa. Reverse blow-out decreased the resistance to 0.5-1.2 kPa. After sleeve filter parameters at SGN and Radon waste vitrification aerosol concentration was decreased from 930 to 22 and from 1500-2000 to 15-30 mg/m3 respectively. Off-gas specific activity at Radon waste treatment was decreased from 7.25 to 0.16 Bq/m3. Trapping efficiency of aerosols and 137Cs was 97.52.0 and 96.20.7% respectively.

During simulated liquid waste treatment the gas constituents concentrations were measured. Average HCl and SO2 concentrations were 11 and 6 mg/m3 respectively during batch feeding and 14 and 4 mg/m3 respectively during glass pouring. Nitrogen oxides concentration ranged between 27 and 38 (averaged -31) g/m3. Carbon monoxide in the off-gas had not been revealed. Oxygen concentration was found to range within conventional values (19.5-21 vol.%). During simulated waste vitrification without agitation the off-gas dust content ranged between 0.3 at the end of batch portion melting and 4.8 g/l immediately after the next batch portion feeding.

A test run conducted to determine material loss from the melter during batch feeding of 10 kg portions has shown that melt agitation increases the solid aerosols losses by factor of 2-3: from 0.27% without agitation to 0.87% with agitation. A dust content in off-gas in the latter case increased to 9.7 g/m3. An increase of solid entrainment at melt agitation resulted in greater dust deposition in off-gas tube (480 g per 1300 g of free-flowing bulk material from collector and filter sleeves. Overall dust entrainment has achieved of 1.9% from produced bulk glass.

At the end of the experiment two runs of batch feeding spiked with 137Cs were carried out. Overall batch activity of 4.45 MBq was fed into melter. Average off-gas specific activity at sleeve filter inlet was 7.3 Bq/l (maximal value - 14.3 Bq/l). Radioactive cesium loss from melter (with respect to off-gas specific activity) was 12.6% from loaded activity. These data should be considered as preliminary due to very short time of the test.

Condensate compositions obtained after the heat-exchanger contained mainly nitric acid. Moreover minor chloride and sulfate ions were detected (up to 2.5 and 0.1 g/l respectively). Average specific activity of the condensate at simulated "Radon" waste operation was 520 Bq/l.

CONCLUSION

Simulated SGN and Radon waste vitrification tests in a "cold crucible" have demonstrated the good workability, reliability and high productivity, and ability of the cold crucible to maintain an underpressure during the operation. Cold crucible capacity can be increased by additional melt agitation. The quality of glasses produced is also high. Lead-containing broken CRT glass can be remelted with formation of chemically durable glass. Common vitrification of lead glass and intermediate-level waste also yields homogeneous glass. Leach rates of sodium, potassium cations and cesium radionuclides were low and they were found to be on the level of good borosilicate glasses.

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MULTIPLE APPLICATIONS OF COLD CRUCIBLE MELTING

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ABSTRACT

Induction-heated cold crucible melting is increasingly seen as a promising technique by the international scientific and technical community. It is capable of achieving high temperatures and power densities with virtually no melter corrosion, making it an ideal solution for many applications ranging from simple melting of metals, molten salts or glass, to complex processes involving chemical reactions in molten baths, or even to the incineration of combustible waste or ion exchange resins.

This technique has been developed by the Commissariat l'Energie Atomique (CEA) for many years, and is now reaching maturity in many sectors of activity.

Metals. A prototype melter has been built in France to ensure the containment of irradiated fuel cladding scraps. The unit produces ingots 20 cm in diameter and 1 m high weighing 200 kg each, with a volume reduction factor of nearly 6.

Molten salts or glass. The CEA operates several prototype units primarily intended to vitrify fission product solutions. They routinely achieve glass production rates of 50 kg per hour, and may be coupled with a calciner to evaporate and calcine the feed solution. Similar units can be used to obtain metals from their oxides by reduction in a molten bath; metal alloys have also been produced.

Waste treatment. Cold crucible melters are used today for simultaneous incineration and containment of combustible wastes including plastics or resins; the latter are burned directly on the surface of the glass melt in which the residue is directly confined, and which may be cast into a container for direct disposal.

As a result of these developments, induction-heated cold crucible melting has reached a level of maturity that makes it a feasible alternative for practically all high-temperature waste treatment processes.

INTRODUCTION

The ever more pressing need to develop a satisfactory solution to the problem of radioactive waste, for lack of a means of eliminating the radioactivity entirely, has led to efforts to confine and contain the waste in increasingly small material volumes. This represents an implementation of the "2C" principle - containment and concentration - as opposed to the "2D" principle - dilution and dispersion.

Three decades of work in this area have shown that this objective requires the use of high-temperature processes, which destroy any organic matter and produce radioactive waste in metallic or generally in oxide form, stabilized within a suitable matrix (1). In France, these principles are implemented by the Commissariat l'Energie Atomique (CEA) and its subsidiaries, COGEMA and SGN; the exemplary operating record of the fission product solution vitrification facilities at Marcoule (since 1978) and La Hague (since 1989) clearly demonstrate the benefits of this technique (2).

Nevertheless, the CEA is continuing to develop increasingly effective processes applicable to an increasingly wide range of waste materials. This is the impetus behind the induction-heated cold crucible melting technique (3) described here.

PRINCIPLE

The technique is based on the use of a water-cooled structure that is transparent to the electric field produced by an induction coil surrounding it; this allows currents to be generated inside the material contained in the structure. The upper diagram in Fig. 1 shows a closed,

water-cooled cylindrical metal structure forming a Faraday cage: the electric field inside is nil, as the current flows only around the periphery. The second diagram in Fig. 1 shows a sectorized metal structure that allows currents to flow in each sector - notably on the inner face, creating a nonzero electric field that induces currents in the material and thus heats it by Joule effect. The process material is thus molten at the core, and solidified on contact with the cooled melter wall.

Fig. 1

Schematically, the current penetration depth p (cf. Fig. 1b) depends primarily on the resistivity r of the material and on the current frequency F:

Eq. 1

where k is a constant depending on the units used. The penetration depth should be roughly equal to the crucible radius for maximum efficiency.

FEATURES AND ADVANTAGES OF COLD CRUCIBLE MELTING

The salient feature of the process lies in the fact that all the equipment components may be cooled. The molten material at the center of the crucible thus solidifies near the melter walls, which never exceed temperatures of about 200C. This has a number of implications:

The crucible is protected against any risk of corrosion, ensuring a very long operating lifetime and allowing it to be used to process a wide range of materials containing even highly corrosive elements: glass with high P_2O_5 and even SO_3^{2-} concentrations may be produced without difficulty. A glass melter has been used at Marcoule for more than a decade, during which it has produced a broad spectrum of glass compositions and highly corrosive molten salts, and yet it still appears virtually new today.

Extremely high - practically unlimited - core temperatures may be reached in the process material. Crucibles of this type have been used to melt uranium and zirconium oxides at temperatures exceeding 2500C, as well as zirconium metal at 1800C and even hafnium at over 2200C.

A cold-crucible melter may be easily dismantled. Neither glass nor metal adheres to the cooled wall, which is therefore never subject to strong contamination.

The melter is highly compact, considering its ability to generate extremely high power densities in the glass - much higher than in an electrode furnace, where the power density is limited by electrode wear. Cold crucible melters are thus particularly well suited for obtaining high throughput in a small volume, and extend the range of potential compositions for solidifying fission product solutions or any other type of waste for which this type of treatment is applicable.

COLD CRUCIBLE MELTING OF METALS

Hull Melting

Metal melting applications mainly involve metal waste consisting of irradiated fuel cladding "hulls" (4) produced by reprocessing spent fuel from different types of reactors. After the fuel subassemblies are cut up and the fuel is dissolved in a nitric acid solution, the waste consists primarily of short cladding tube scraps made of zircaloy in pressurized water reactors, or of stainless steel in fast reactors. The hulls are both alpha and high-level beta-gamma wastes, as they may contain transuranic nuclides in residual undissolved fuel as well as high-level fission or activation products.

The melting process is illustrated schematically in Fig. 2. The water-cooled copper crucible is surrounded by an inductor. A water-cooled mobile bed is initially positioned so that its top face is level with the bottom of the inductor. The hulls are then placed in the crucible and melting occurs in the presence of oxide or fluoride flux that fills the gap formed by magnetic striction between the molten metal and the cold crucible. Hulls are fed continuously to the melter with a suitable percentage of slag. When the molten metal bath reaches a predetermined level, the bed begins to move downward continuously to maintain a constant level in the crucible. The metal solidifies as it exits the inductor, allowing the extraction of an ingot with the diameter of the crucible and with any desired length, surrounded by a thin layer of solidified slag. Most of the slag detaches naturally from the ingot during cooling, and is suitable for further treatment. The flux is important in digesting the oxides formed during melting, as well as any residual (e.g. uranium and plutonium) oxides still contaminating the hulls after fuel dissolution.

Fig. 2

A full-scale prototype unit implementing this process has been built by the CEA. The facility includes the following major components:

the melter assembly comprising a 20 cm ID cold crucible surrounded by an inductor supplied with 10 kHz current from a 300 kW motor generator; a conical manifold above the crucible provides for process inlet and outlet flows (hulls, flux, argon, off-gas); the complete assembly is cooled with demineralized water;

the hull feed system, including a feed hopper and a rotating disk distributor;

the flux feed system installed in a glove box and connected to the cold crucible via a series of shutoff valves;

the off-gas treatment system, comprising a particle separator, a scrubbing column and a HEPA filter stage;

the ingot drawing machine.

Melting is performed automatically using a centralized control system to produce ingots 200 mm in diameter and 1000 mm long. Tens of metric tons of nonradioactive zircaloy and stainless steel hulls have been melted in this prototype unit.

Development work on this project is temporarily suspended, as COGEMA is now considering a compaction process to handle such wastes in its reprocessing plant at LaHague.

Other Applications

A cold crucible melter is not limited to merely remelting metals. It may also be used to implement chemical reactions, for example to produce metals - including uranium. Uranium metal is generally produced by metallothermic reactions between UF_4 or U_3O_8 and reactant metals such as calcium, magnesium or aluminum. A cold crucible reaction process would present a number of advantages: high-frequency induction melting of raw materials creates suitable reaction conditions; magnetodynamic phenomena created by the electric fields may favor coalescence of the metal and improve the uranium ingot quality; the cold structure is capable of removing the heat generated by the exothermic reaction, and thus shorting the fabrication cycle; the process is well suited to continuous production. The feasibility of the method has been demonstrated by calciothermic tests with UF_4 and U_3O_8.

Cold crucible induction calciothermy may also be applied to other metals. Tests with neodymium fluoride to produce Nd-Fe alloys have yielded promising results.

COLD CRUCIBLE MELTING OF MOLTEN SALTS AND GLASS

Molten salt processes will only be mentioned here as a potential application for which work is only is only beginning, i.e. the interactive chemistry involving molten salts and metals, with possible applications in reprocessing. The advantages of the cold crucible are obvious in this context, where the extent of the corrosion problems encountered has hamperered the design a suitable contact reactor; today this is no longer an obstacle and a new avenue of research has been opened.

The following discussion covers glass melting, beginning with the solidification of fission product solutions. Direct induction is possible in molten glass (5), which is slightly conductive with an electrical resistivity ranging from 1 to 50 Wm. The required induction frequency is inversely proportional to the crucible radius, between 200 and 300 kHz for a melter a few tens of centimeters diameter.

Figure 3 shows a cold-crucible melter of the type used today at Marcoule, ranging from 30 to 55 cm in diameter and from 30 to 70 cm high. Several units are in use, and have logged a cumulative total of several thousand hours in operation, supplied either directly with glass or with a calcine and frit mixture from a calciner similar to the ones used in the R7 and T7 vitrification facilities at LaHague.

Fig. 3

Melting may be initiated using a microwave system by incorporating a few hundred grams of a suitable glass composition, or by placing an annular aluminum, titanium or zirconium starter element that burns in air when induction heated, melting the surrounding glass and allowing induction currents to develop in the melt.

The capacity of this type of melter depends on both the surface area and the operating temperature. The actual value is roughly proportional to the surface area and to the fourth power of the temperature: melting occurs by radiation from the molten glass to the surface. Table I indicates the rates extrapolated from a throughput rate of 30 kgh-1 for a melter 55 cm in diameter operating at 1200C, as demonstrated with the R7 glass used at LaHague. In fact, these values are underestimated, as they do not allow for the effects of stirring the melt (which could easily be obtained either by sparging or mechanically using a cooled stirring device), which would increase the capacity by at least 50% - assuming the generator is powerful enough to maintain the glass in the molten state under these conditions.

Table I

A new melter 1 meter in diameter has been built at Marcoule and is now being tested with a 160 kW transistor generator at a frequency of 100-200 kHz. This unit opens the way to a new generation of large melters that have been thoroughly modeled by the CEA, and which will probably constitute the most suitable means for vitrifying the very large backlog of high-level fission product solutions now in interim storage throughout the world. This potential was not lost on the international experts called together by WHC at Hanford in mid-1994 to select a melting process for vitrification of the high-level waste at the site: the cold crucible melter tied for first place with the liquid-fed ceramic melter.

Similarly, COGEMA has undertaken a program to nuclearize the cold

crucible melter for testing in one of the vitrification facilities at LaHague.

Low- and medium-level radioactive waste produced in variable quantities by power reactors - VVERs in the East and PWRs in the West - could also be vitrified in cold crucible melters. The feasibility of this application has been demonstrated in France using a two-step process of calcination in a rotating kiln followed by melting in a 50 cm diameter cold crucible supplied with both calcine and a natural aluminosilicate mineral, clinoptinolite. This operation was conducted to substantiate the potential for vitrifying liquid waste with high boron and soda content from the Czech power plant at Temelin; several cubic meters of simulated solution were vitrified in a few hundred kilograms of glass (6).

The possibility of feeding the process solution directly to the molten glass in a cold crucible has been investigated and tests have been conducted. This procedure significantly diminishes the melter capacity, especially with dilute feed solutions. Russian scientists routinely use this method in cold crucible melters by first concentrating the solution and its additives (7). The advantage of this approach is to eliminate the prior calcining step, and is most appreciable in the case of solutions that are difficult to calcine, e.g. those containing large amounts of sodium, as with low- and medium-level waste solutions. A drawback, however, is the increased entrainment of particle material, which complicates the design of the off-gas treatment system.

Incinerator ashes are another waste form that can easily be melted in a cold crucible. Whether produced by a or bg incinerators, they contain mainly silica (SiO_2) and other oxides (CaO, MgO, etc.). The refractory nature of their components makes them generally difficult to melt as is; simply adding basalt or a small amount of sodium carbonate flux largely facilitates the process. This is a simple operation that today routinely yields several tens of kgh^{-1} of a glass or vitrocrystalline material with potentially very good containment properties, and allowing volume reduction by at least a factor of five.

NEW DEVELOPMENTS AND APPLICATIONS FOR COLD-CRUCIBLE MELTING

New Developments and Applications for Cold-Crucible Melting

In addition to increasing the capacity - an experimental unit 1 m in diameter has already been demonstrated with a capacity of 100 kgh^{-1} - the CEA is now orienting its research program toward an increasingly wide range of wastes, and notably incinerable waste. The objective is to achieve incineration (i.e. thermal decomposition of the organic structure) and vitrification of the inorganic residue simultaneously and directly in the molten glass. This process is of particular interest for plastic and cellulosic wastes - as well as ion exchange resins, which are currently conditioned in cement of polymer matrices representing a significant waste volume for disposal. A number of difficulties must be overcome before incineration-vitrification becomes a viable process:

The combustion off-gases are generally chlorinated in the case of plastic wastes such as polyvinyl chloride, and generally sulfated in the case of resins, many of which are sulfonated. The CEA has acquired considerable experience in incinerating highly chlorinated plastic wastes, and has developed an incineration process for a-bearing waste (8,9) now implemented in industrial facilities at two French nuclear centers and which could be of interest to the world nuclear community. Virtually all the chlorine from the feed material enters the off-gas stream and is eventually converted to hydrochloric acid from which all

the radioactivity must be eliminated. Similarly, the sulfur contained in sulfonated resins will most likely be recovered in sulfuric acid or sulfate form.

A second difficulty involves the actual formulation of the glass or vitrocrystalline material used to confine the inorganic residue and thus the radioactivity. It is virtually impossible to include chlorine or sulfur in significant amounts in vitreous or vitrocrystalline materials with good containment properties. The advantage of the cold crucible technique in this area is that it obviates all the corrosion problems attributable to chlorine and especially sulfur, which at very high temperatures, and often as a molten salt, is incompatible with the use of conventional refractory or metal materials.

Laboratory tests are in progress at the CEA's Marcoule center, and a small-scale pilot facility is now under construction with a 30 cm diameter cold crucible and a suitable off-gas treatment system.

Another potential application of cold crucible technology has also been investigated: the incineration of graphite. Under a CEA research and development program to recover plutonium from the graphite at a French defense site, the feasibility of this approach was demonstrated using a cold crucible 30 cm in diameter together with a post-combustion chamber. The problem here, of course, is not to burn the graphite, but to control its combustion. The nonradioactive tests completed to date have been conclusive, and the process will be implemented to burn plutonium-contaminated graphite in French defense centers.

CONCLUSION

Vitrification today is gradually supplanting other waste conditioning and containment processes. It is clearly of economic interest in that it provides for a very significant volume reduction - unlike the other available processes, which increase the volume of the final waste form. It is also of ecologic interest, by creating a high quality containment matrix with excellent long-term behavior.

Cold crucible melting will continue to advance as simple, relatively inexpensive techniques that generate no secondary waste are demonstrated and proven industrially. High-level waste has been vitrified by COGEMA in France for more than 15 years; low-level waste will one day be vitrified - and followed by other liquid and solid waste forms.

The cold crucible melter is unrivaled for its flexible operation, for the extremely high temperatures it allows, for its high specific capacity and compact size, for its operating lifetime and virtual absence of secondary waste, and for its potential to "digest" practically any form of liquid or solid waste including both metals and combustible materials.

It is now necessary to develop units with higher capacities and to conclusively demonstrate the possibility not only of melting but also of incinerating liquids as well as solids. Considerable work remains to be done, but the future of cold crucible melting is extremely promising.

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27-5

TECHNICAL AND ECONOMICAL ASSESSMENT FOR VITRIFICATION OF LOW-LEVEL RADIOACTIVE WASTE FROM NUCLEAR POWER PLANTS IN KOREA

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ABSTRACT
In order to obtain yearly generation volume and composition data for low- and medium-Level radioactive waste, we investigated the characteristics and generation trends for each waste which has been produced at nuclear power plants in Korea. The data were utilized for technical and economical assessment. For technical assessment, four promising melters, which were the cold crucible heated by direct induction (CCM), the cold crucible melter heated by vertical electrodes (CCVE), the quantum catalytic extraction process (QCEP), and the plasma torch melter (PT), were selected to determine the best available melter in the view point from Korean situation. Four melters were evaluated according to evaluation factors such as waste feeding, melter life, replacement materials during melter life, throughput, entrainment of volatile Cs and toxic material, and recycle potential of spent scrubbing solution to melter itself. Economical assessment was carried out for four treatment strategies with the melters selected in the technical assessment. For each strategy, the capital and the operation cost were estimated and the

disposal volume was calculated with reasonably estimated volume reduction factors with respect to the waste type and the treatment concept. Then the disposal costs were calculated according to the several unit disposal costs. The treatment and disposal cost were plotted for several unit disposal costs from 2,500 to 20,000 US$/m3. The results showed that the strategies adopting the CCM and/or the PT were more economical than the current treatment system for overall unit disposal cost range if the waste treatment system is newly constructed according to the strategies, and they were more economical only for the unit disposal cost of 6,250 US$/m3 and more in case that the current system is modified with the CCM and/or the PT.

INTRODUCTION

In Korea, new treatment technology development of Low-Level radioactive waste (LLW) is necessary due to high concern for the protection of environment, the difficulty in disposal site selection, and the disposal cost increase. The vitrification technology is considered as the most promising one of new LLW treatment technologies and it is a world-wide trend to apply it for the treatment of LLW. Korea Electric Power Research Institute (KEPRI) has already launched a feasibility study on LLW vitrification to determine whether or not vitrification technology is adopted to treat LLW from domestic nuclear power plant and to suggest a reasonable development strategy in the case that its adoption turns out to be reasonable. Therefore, the objectives of this study are focused on the technical and the economical assessment of vitrification for LLW from nuclear power plant in Korea.

For the study, waste generation trends are first analyzed to determine the yearly generation volume and characteristics for each waste stream. Secondly, we will compare and analyze the characteristics of glass melters which are being developed and studied in the world, including melter type, operation temperature, expected life time, waste feed, throughput, pre-treatment method, off-gas system, etc.. In order to select melters being utilized for economical assessment, detailed technical assessment is going to be conducted for four promising melters. Then the economics for four LLW treatment strategies with the melters selected is to be evaluated in detail, considering comparison of treatment and disposal cost for each strategy.

WASTE CHARACTERISTICS AND GENERATION VOLUME

LLW generated at nuclear power plant could be categorized as combustible and non-combustible dry active waste (DAW), spent resin, spent liquid filter, and evaporator bottom products (borated liquid waste concentrate). In order to obtain yearly generation volume and composition data for low- and medium-Level radioactive waste, we investigated the characteristics and generation trend for each waste which has been produced at nuclear power plants in Korea (1-4). Table I shows the waste characteristics and the maximum volume generated at a site having two 1,000 MWe PWR. Total generation volume turned out to be 346.5 m3 (216.97 TON) and was composed of combustible DAW of 200 m3 (40 TON), non-combustible DAW of 50 m3 (75 TON), evaporator bottom products of 80 m3 (88 TON), spent resin of 14 m3 (10.22 TON), and spent liquid filter of 2.5 m3 (3.75 TON). Radiological data in Table I were quoted from EPRI report (5) in which main nuclides were Fe-55, Co-60, Ni-63, and Cs-137 and specific activity for each waste stream was increased as order of spent resin - spent filter - DAW - evaporator concentrate.

Table I

In Korea, combustible and non-combustible DAW have been drummed after sorting according to four classification categories. In other words, combustible waste has been classified as vinyl seats & shoes cover, protection clothes & shoes, paper & wood, and sponge, etc., and in the case of non-combustible DAW as concrete & asbestos, air filters, glass & metallic, and sand, etc.. Table II and III show, respectively, composition and ingredients for each DAW. As shown in Table I to III, LLW was composed of very complex ingredients which would generate various types of chemicals if DAW was pyrolysed or incinerated (6).

Table II

Table III

TECHNICAL ASSESSMENTS FOR FOUR PROMISING MELTERS

For the technical assessment, we chose four types of melters such as the cold crucible heated by direct induction (CCM), the cold crucible melter heated by vertical electrodes (CCVE), the quantum catalytic extraction process (QCEP), and the plasma torch (PT). Table IV shows the characteristics for each melter such as melter life, throughput, operation temperature, and waste feeding & volume reduction factor, etc. which were drawn from materials provided by companies (7-9). As shown in Table V, important items were selected as evaluation factors, which have scoring range of 1 to 5, and weighting multiplier with respect to importance for cost benefit of operation and maintenance. The assessment items include the throughput, the maintainability, the off-gas entrainment for volatile material like Cs nuclide, the melter life, and the waste feeds. Evaluation criterion for throughput, 150 Kg/h was calculated based on the assumption that a vitrification facility could deal with all waste in Table I and was operated for 2,080 hours per year.

Table IV

Table V

The technical assessment resulted in Table VI in which the CCM was the most promising system from the viewpoint of Korean situation. It is also desirable that both the CCM and the PT are applied to LLW treatment for two reasons such that 1) The molten glass of CCM has good retaining capability for nuclides and toxic materials having low boiling point (10), and the spent scrubbing solution can be recycled to the CCM. However, there is a limit on the content of metallics for input into the CCM; 2) The PT can treat all kinds of waste without any pre-treatment, but the PT has disadvantages that lots of radioactive Cs might be volatilized to increase the shielding cost for off-gas treatment system and the generation volume of spent scrubbing solution which can not be recycled to the PT. Therefore the PT might generate a lot of secondary wastes.

Table VI

ECONOMICAL ASSESSMENT FOR FOUR TREATMENT STRATEGIES

The economical assessment was conducted for four treatment strategies as follows:

Strategy I: the current treatment concept adopted in Korea. Currently, the DAW is compacted by super compactor. The spent resin is solidified with cement by the in-line mixing system and spent liquid filter is drummed into cement lined container. The liquid radwaste has been concentrated with the evaporator and then solidified with cement. In the near future, evaporator bottoms will be dried and solidified with paraffin if the dryer installation is completed.

Strategy II:

1) the metallic waste and the air filter of non-combustible DAW: super compacting,
2) the liquid filter: drumming into cement lined container,
3) the evaporator bottoms: vitrification (CCM: 200 Kg/h throughput) after drying.
4) the non-combustible DAW excluding metal and the air filter, the combustible DAW, and the spent resin: direct vitrification (CCM) without incinerator,

Strategy III:
1) the non-combustible DAW and the liquid filter: treatment with plasma torch melter (PT: PACT-2 throughput(see Table VIII)),
2) the spent scrubbing solution and the evaporator bottoms: vitrification (CCM: 200 Kg/h throughput) after drying,
3) the others: direct vitrification (CCM) without incinerator.

Strategy IV:
1) the spent scrubbing solution: vitrification (CCM: 10 Kg/h throughput) after drying,
2) the others: treatment with plasma torch melter (PT: PACT-6 throughput).

Table VII shows the reasonable estimated volume reduction factors (VRF) with respect to the waste type and the treatment concept, final volume, and treatment cost for each strategy. Treatment costs were calculated with data of Table VIII in which almost all data were provided by companies and a few data were taken from EPRI (11) and EPA (12) report. This data was recalculated considering escalation rate.

Table VII

Table VIII

For strategies II to IV, both minimum and maximum VRF were applied to combustible DAW, spent resin, and evaporator concentrate which were likely to have very wide range of VRF. Because all VRF data for the plasma torch could not be found. So we determined somewhat larger values than those of vitrification. If spent resin is treated by plasma torch melter without considering radiological aspect, maximum VRF for spent resin is likely to be around 150. However, it was determined as 30 because 1) it is desirable that nuclides contents in glass/slag waste forms do not exceed class C; 2) SEG (U.S.A) suggested that maximum VRF for spent resins was about 30 if they were treated by QCEP which had capability to obtain similar VRF as a plasma torch melter. We also assumed that 1) generation volume of spent scrubbing solution from plasma torch melter (PT) and cold crucible (CCM) were about 1.0 and 0.3 m3/TON of waste, respectively; 2) only air filter of noncombustibles contributed to generation of spent scrubbing solution; 3) all sand and glass of noncombustibles were used as glass/slag former. This resulted in abnormal high VRF for noncombustibles.

For treatment cost calculation, all equipment costs utilized for a strategy were summed and divided by the amortization period (15 years). Then operation costs were added to them. Total treatment costs for strategy II to IV have lower and upper values. Upper values includes the capital and operation cost per year in case that the waste treatment system is newly constructed in accordance with the strategies, and lower value means in case that the current system is modified with plasma torch and/or vitrification melter. For strategy IV, TCY= 730.7 and 343.3 represent treatment cost with evaporator and vitrification melter, respectively.

As compared with strategy I, Strategy II to IV reduce the disposal volume by 70 - 92 %. Disposal costs were calculated with respect to the several unit disposal costs from 2,500 to 20,000 US$/m3. Figure 1 and 2 show that treatment and disposal cost for each strategy with respect to construction type (new or modification) of waste treatment system when minimum VRF is applied. From the figures, it can be shown that the strategies adopting the CCM and/or the PT are more economical than the current treatment system for overall unit disposal cost range if the waste treatment system is newly constructed according to the strategies, and they are more economical only for the unit disposal cost of 6,250 US$/m3, which is lower than the disposal cost of Barnwell site (8,300 $/m3), and more in case that the current system is modified with the CCM and/or the PT.

Fig. 1

Fig. 2

CONCLUSIONS

In Korea, the maximum volume generated at a site having two 1,000 MWe PWR turned out to be 346.5 m3 (216.97 TON) and was composed of combustible DAW of 200 m3 (40 TON), non-combustible DAW of 50 m3 (75 TON), evaporator bottom products of 80 m3 (88 TON), spent resin of 14 m3 (10.22 TON), and spent liquid filter of 2.5 m3 (3.75 TON).In order that these all wastes are economically encapsulated into glass/slag waste forms, it is desirable that both the direct induction heated cold crucible and the plasma torch hearth are adopted. From the viewpoint of technical and economical aspects, it might be concluded that strategy (III) is the best concept in which noncombustibles and spent liquid filter are treated by the plasma torch hearth and the others by the cold crucible.

The vitrification facility to be constructed by strategy (III), which has enough throughput of 250 kg/hr (CCM: 200 kg/hr, PT: 50 kg/hr) to deal with all wastes from a site having four 1,000 MWe PWRs, will have the average volume reduction factor of 12.63 - 24.87. The number of glass/slag waste forms generated from 12 units (1,000 MWe PWR) for 30 years will be 12,060 - 23,580 drums (200 liter size).

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Session 28 -- PROGRESS IN HL TANK WASTE REMEDIATION
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28-1

ASSURING SAFE INTERIM STORAGE OF HANFORD HIGH-LEVEL TANK WASTES

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ABSTRACT

This paper describes the redirection of the Tank Waste Storage Program. The redirection will allow placement of all tanks in a well-defined safety envelop and will prepare the tanks for treatment and long-term disposal of tank waste.

This paper will present the new directions and associated accomplishments. The redirections are the result of Tank Waste Remediation System (TWRS) fiscal constraints, changes in U.S. Department of Energy management philosophies, and changing regulatory and oversight requirements - quantum changes from past business practices.

INTRODUCTION

The federal government established the Hanford Site in South-Eastern Washington near the City of Richland in 1943 to produce plutonium for national defense purposes. The Hanford Site occupies approximately 1,450 square kilometers (560 square miles) of land North of the City of Richland. The production mission ended in 1988, transforming the Hanford Site mission to waste management, environmental restoration, and waste disposal. Thus the primary site mission has shifted from production to the management and disposal of radioactive, hazardous, and mixed waste that exist at the Hanford Site.

This paper describes the focus and challenges facing the Tank Waste Remediation System (TWRS) Program related to the dual and parallel missions of interim safe storage and disposal of the tank associated waste. These wastes are presently stored in 2.08E+05 liters (55,000) to 4.16E+06 liters (1,100,000) gallon low-carbon steel tanks. There are 149 single- and 28 double-shell radioactive underground storage tanks, as well as approximately 40 inactive miscellaneous underground storage tanks. In addition, the TWRS mission includes the storage and disposal of the inventory of 1,929 cesium and strontium capsules created as part of waste management efforts.

Tank waste was a by-product of producing plutonium and other defense related materials. From 1944 through 1990, four (4) different major chemical processing facilities at the Hanford Site processed irradiated (spent) fuel from defense reactors to separate and recover plutonium for weapons production. As new and improved processes were developed over the last 50 years, the processing efficiency improved and the waste compositions sent to the tanks for storage changed both chemically and radiologically. The earliest separation processes (e.g.,bismuth phosphate coprecipitation) carried out in TPlant (1944-1956) and BPlant (1945-1952) recovered only plutonium. All remaining dissolved fuel elements, including enriched uranium, were sent to the tanks as alkaline waste. Later processes, such as the Reduction Oxidation Plant Process (REDOX) and Plutonium Uranium Extraction (PUREX) flowsheets were developed to also recover uranium which was then recycled back into making reactor fuel. The process of purification of both plutonium (ZPlant) and uranium (BPlant) also lead to the creation of waste streams which, after neutralization to a pH10, were added to the tanks.

Most processes associated with plutonium recovery from spent fuel involved dissolving the material in nitric acid. After extensive acid side chemical separations to recover plutonium, uranium, and often neptunium, the waste streams were made alkaline by addition of sodium hydroxide and/or calcium carbonate prior to their transfer to the low-carbon steel waste tanks. Making waste alkaline produced large quantities of metal oxyhydroxides, which along with solids from the bismuth phosphate process formed the sludge found in the bottom of the tanks. The waste composition in tanks was complicated further by the recovery of uranium by sluicing during 1952-1958. The waste was made alkaline to prevent corrosion of the low-carbon steel tanks, thereby introducing large volumes of sodium nitrate and other sodium salts into the waste tanks. Sodium nickel ferrocyanide was added to 20 of the tanks during the 1950s in order to precipitate solids and create additional space in the tanks.

To increase useful storage capacity, volume reduction methods (e.g., in-tank and external evaporation), and recovery of heat producing cesium and strontium in BPlant (1968-1985) were carried out. The concentration of the originally soluble sodium salt-rich waste led to the production of the saltcake, which is often found overlying the sludge waste in the tanks. Most of the hazardous chemicals and radionuclides are found in the sludge. Only radio-cesium, -iodine and -technetium are significantly soluble in alkaline salt solutions.

The single-shell tanks were taken out of active service in 1980, and no new waste has been added to these tanks since then. Sixty-seven (67) of these tanks are assumed or confirmed leakers. Removal of drainable liquid by saltwell pumping (interim stabilization), waste sampling in support of characterization, installation of new monitoring equipment, and/or any mitigation or remediation deemed necessary to assure interim safe storage of the waste are the only significant intrusive activities into these tanks. Drainable liquid removed from the single-shell tanks, as well as dilute waste resulting from decontamination of production facilities are added to the double-shell tanks after due consideration of waste compatibility concerns. There are approximately 2.12E+08 liters (55 million gallons) of waste in the TWRS single- and double-shell tank system.

THE TWRS MISSION

The TWRS Program is the largest environmental clean-up program in the United States.

The purpose of the TWRS interim safe storage mission is to place all tanks in a safety envelope and prepare for long-term disposal of tank waste.

Mission Related Activities Include:

1. Continuation of the current program of tank monitoring and maintenance including any enhancements needed to assure interim safe waste storage (see the discussion of controlled, clean, and stable below);
2. Resolution of safety issues related to interim safe storage and/or disposal of the waste in the tanks;
3. Removal and transfer of pumpable liquids from single-shell tanks by saltwell pumping (interim stabilization);
4. Performing waste and tank characterization to the extent needed to either resolve tank safety issues or to support safe retrieval, transfer, processing (pretreatment) and disposal of the waste;
5. Continuation of receipt and storage of newly generated waste in double-shell tanks; and
6. Concentration of waste to the maximum extent safely practical in the 242-A Evaporator.

These activities form the basis for assuring that the tank contents and the farms themselves are maintained in a controlled, clean, and stable mode until the waste is retrieved and processed for disposal.

In addition, major resources are being expended on:

1. Development and application of systems engineering to assure integration of the overall TWRS mission activities;
2. Preparation for phased waste retrieval, treatment and waste vitrification efforts that form DOEs first major step in demonstrating waste disposal utilizing private sector resources (privatization); and
3. Achieving an integrated and responsive authorization basis to assure continued safe operation of the tank farms as the mission changes from storage to disposal.
4. Gaining knowledge of tank waste.

The remainder of this paper will highlight recent achievements in the areas of resolution of tank waste safety issues, progress toward putting the tank farms into a controlled - clean - stable mode, and transition to an operating mode in which privatization is a key operating factor.

TANK WASTE SAFETY ISSUES

All U.S. Department of Energy (DOE) facilities that store hazardous or radioactive materials have documented safety analyses, which establish a range of operating parameters (e.g., temperature, pressure, concentration) within which routine operations are conducted. These safety analyses also evaluate the effects of potential accidents, abnormal events, and natural disasters. The DOE has a formal program which requires identifying any known or suspected conditions that have not been analyzed or fall outside of the observed safety range as an Unreviewed Safety Question (USQ). Following identification of a USQ, a review takes place that may result in a change to the safety documentation or a change in operations. Following the review process, the USQ may be closed from an administrative standpoint when conditions surrounding the safety issue have been reviewed and their effects bounded. However, the safety issue may still exist and may require operational constraints, ongoing monitoring or mitigation. [In that fashion, safety issues, and USQs are related but not identical]

Concern over waste tanks having the potential for releasing high-level radioactive wastes to the environment resulted in the passing of Public Law 101-510, section 3137, "Safety Measures for Waste Tanks at Hanford Nuclear Reservation," also known as the Wyden Amendment. In response, DOE has developed a set of criteria to identify tanks with potential safety concerns as "watch list" tanks. There are currently 54 "watch list" tanks, with 10 tanks that are listed in more than one (1) of four (4) different categories based on the specific safety concerns described below.

Over 50 years of fuel reprocessing at Hanford has resulted in the accumulation of nearly 2.12E+08 liters (55 million gallons) of waste in the single- and double-shell tanks. Prior to the 1990s, it was generally believed that the stored wastes were chemically unreactive under both the present storage conditions and plausible accident scenarios, as well as being chemically stable. This paradigm was proven wrong when detailed evaluation of tank contents and behavior discovered that:

1. Twenty-five (25) of the stored single- and double-shell tanks waste tanks were generating, storing, and releasing hydrogen in quantities that might lead to flammable gas concentrations above the safety margin of 25% of their lower flammability safety limit (The Flammable Gas Issue). Recent measurements have shown that changes in atmospheric pressure may cause changes in liquid levels in some of these and other waste tanks. This may indicate a greater amount of gas stored in the waste than previously believed, and 25 additional tanks may be added to the flammable gas watch list until this issue is resolved. Screening of all waste tanks for potential flammable gas accumulation has been initiated.
2. Twenty (20) single-shell tanks contain organic materials in the presence of excess sodium nitrate and sodium nitrite oxidizing agents that could lead to a potential propagating reaction, and ensuing release of radioactive and hazardous materials to the environment, if the waste was dried and heated to threshold temperatures above 200C. (The Organic Safety Issue)
3. Eighteen (18) single-shell tanks contained sodium nickel ferrocyanide which could similarly pose a threat of a propagating reaction, if dried and heated to above 250C. Ferrocyanide based deflagration represents the bounding "worse case" accident scenario previously identified in the Hanford Environmental Impact Statement. (The Ferrocyanide Safety Issue)
4. A single-shell tank (241-C-106) contains sufficient heat producing radio-strontium that it requires addition of cooling water to prevent tank failure from structural damage, if its temperature is allowed to fall outside of safe operating criteria. (The High Heat Issue)

The Waste Tank Safety Program was chartered in 1990 to address the four (4) safety issues discussed above. If risks were high, relative to interim safe storage of the wastes until they could be retrieved and permanently disposed, then the safety program assured that either adequate controls were in place to prevent the condition of risk from occurring, or if that were not possible, mitigation or remediation of the condition was initiated to actively remove the cause at the tank farms. Until such data were collected and interpreted to assure continued safety, the tanks associated with safety concerns were placed under stringent operating controls.

Extensive work and increased knowledge over the last few years has led to the demonstration and documentation of a significantly lower-risk condition with the tank waste. This will allow us to close some of the

safety issues discussed above. For example, laboratory studies with simulants and analyses of actual waste samples have bounded the energetics of the fuel-rich materials added to the tanks. In addition, storage of the wastes over the last 30 to 40 years resulted in degradation of the organics and ferrocyanide significantly reducing the potential for explosive reactions.

Key Understandings Leading to Safety Issue Resolution Include the Following:

Demonstration that radiolytically or chemically induced waste aging processes have destroyed or significantly lowered the energy content of a vast majority of organic materials added to the tanks. Therefore, conditions exist which can no longer have the potential for a propagating reaction in even a dry tank. Furthermore, the exploration of waste species energetics, waste species solubility, and waste tank chemistry demonstrate that the organic rich tanks contain a sufficient amount of moisture to preclude a risk from propagation under even the bounding "worse case" accident scenario. A combination of experimental work and waste characterization is continuing to provide data to substantiate these findings of waste safety.

Results of extensive tank monitoring and surveillance show that most of the tanks on the Flammable Gas Safety Watch List pose little potential for exceeding 25% of the lower flammability limits for generated gases, or if such gas accumulation potential exists, it can be mitigated by installing a low-flow ventilation system on the tank.

PROGRESS TOWARD PLACING THE SINGLE-SHELL TANK FARMS INTO A CONTROLLED - CLEAN - STABLE MODE

An essential element of the strategy for meeting the TWRS mission is achieving a Controlled, Clean, and Stable condition in the Tank Farms. This strategy is essential in achieving an interim, safe, low-cost status until retrieval and disposal operations commence in the tanks.

The definition of controlled, clean, and stable is as follows:

1. Controlled

a) All necessary (as determined by safety analysis) active and passive safety systems are in place.

b) Resolution of any safety issues related to interim safe storage and/or disposal of the waste stored in watch list tanks.

c) All controls necessary to provide assurance of meeting risk acceptance criteria for current operations associated with these USQs are in place. Continuous, remote, on-line monitoring of key parameters, such as waste volume and temperature is in place to adequately control the Hanford Site high-level waste tanks. This includes continuation of the current program of remote tank monitoring and maintenance, including any enhancements needed to assure interim safe waste storage.

2. Clean

a) Surface contamination areas are cleaned and reduced to radiological control areas or even less controls.

b) Unused contaminated equipment is removed from the tank farm.

c) Reusable equipment is stored, if not in use.
3. Stable
a) Removal and transfer of pumpable liquids from single-shell tanks by saltwell pumping (interim stabilization);
b) All penetrations where liquids could intrude into the tanks are sealed.

Controlled, Clean, and Stable Strategy

The controlled, clean, and stable strategy has the following four (4) elements, which are discussed below: 1) reduce the mortgage; 2) provide safe storage; 3) reduce worker exposure to hazards; and 4) maintain compliance with regulatory requirements.

1. Reduce the Tank Farm "mortgage." - The "mortgage" is the current operational costs to monitor the tanks and their waste. This task is currently a high labor-intensive effort that includes a large number of tank farms and tank entries. The procedures for single-shell tank farm entries will be modified to:
a) Require tank farm entry only on a non-routine basis.
b) Provide remote monitoring of all essential parameters.
c) Allow access for waste sampling and characterization, if required.
2. Provide safe storage prior to retrieval. - This task includes updates to safety analyses as well as specification of necessary engineering design features, and operational and administrative controls. It also includes any equipment modifications and development of procedures and training for the equipment.
3. Reduce worker exposure to hazards. - Every time a worker enters a tank farm, or operates equipment in or near a tank, there is some level of potential exposure to radiological and chemical hazards. The reduction of the areas specified as either radiological controlled areas (RCAs) or surface contamination areas (SCAs) reduces worker exposure to these hazards.
4. Maintain compliance with regulatory requirements. - The Hanford Site Tank Farms' are regulated as a Treatment Storage and Disposal Facility with associated permits and closure agreements in compliance with regulatory requirements. All interim storage activities must be performed in a manner to assure current and continued future compliance with regulatory requirements.

Planned Upgrades in Support of Controlled - Clean - Stable

Focused tank farm upgrades are planned to improve the reliability of safety-related systems, minimize on-site health and safety hazards, improve the regulatory compliance of tank farm support systems, and put the tank farms into a controlled, stable work environment until disposal is completed. The following upgrades are planned:

Instrumentation such as automatic tank data gathering, management control systems, and closed circuit television monitoring will be upgraded or added to minimize personnel exposure and to provide more accurate data for tank status assessment.

Tank ventilation systems will be upgraded to replace outdated ventilation systems.

Electrical systems will be upgraded to meet capacity needs for both routine monitoring and to support retrieval, as well as to comply with current electrical codes; and

Piping systems will be upgraded to enable transfer of liquid or waste slurries from the decontamination and decommissioning of other selected Hanford Site facilities to the tank waste system.

CHARACTERIZATION OF TANK WASTES

Another essential element of the strategy to achieve the TWRS mission is the characterization of tanks wastes. "Characterization" is understanding the Hanford Tank Waste chemical, physical, and radiological properties to the extent necessary to ensure safe storage, interim operation, and ultimate disposition of the waste. Due to the many processes that have been used at the Hanford Site and the varied waste resulting from them, coupled with ongoing reactions in the waste storage tanks, there is a great deal of uncertainty about the exact waste inventory in many of the tanks. Knowledge of the waste in the tanks is essential to define the extent of existing safety issues, to resolve the safety issues, and to support retrieval, treatment, and disposal system designs.

There are currently five (5) sampling methods used to gather information on tank wastes.

1. Grab sampling of supernatant liquids for laboratory analysis. (Use of a bottle to "grab" liquid at the tank waste surface)
2. Vapor sampling for both on-line and laboratory analysis.
3. Core sampling of solid wastes using core sampling systems designed to drill or push into the waste to retrieve segments that are about 2.5 cm (1-inch) in meter by 50 cm (19 inches) long. These segments are then transported to on-Site laboratories for extrusion and analysis.
4. Auger sampling of the top 40 cm of waste in the tanks.
5. In-situ measurement of the void volumes in the waste and the viscosity of the waste.

To address the challenge of safely characterizing the waste tanks and to bring focus to the program, the TWRS Tank Characterization Project was formed in February 1995. This led to the following accomplishments during the year:

Two (2) new rotary core sampling trucks were delivered and accepted in July 1995. Field testing was completed in September 1995, and operational production commenced in October 1995.

Key improvements were made to the drilling equipment and the drill bits to make them more compatible with the type of waste being drilled and to improve sample recovery. This resulted in sample recoveries increasing from an average of 20% to over 90%. Sampling equipment improvements included:

- developed equipment for sampling different types of waste (dry or wet salt cake, sludge, liquid) which are highly radioactive and/or toxic as well as potentially flammable.
- developed shielding equipment for protecting personnel from potentially high radiation exposure and contamination.
- developed a complete core sampling system including the sampler, sample truck, nitrogen purge supply, exhauster, x-ray imager, and a cask truck for transportation of the waste samples.
- secured radiation hardened video cameras for in-tank color photography.

New x-ray imaging system was added to sampling truck to determine the amount of sample recovery in the field immediately following sample removal from the tank.

The Tank Waste Characterization Basis document was completed for the safety program in June 1995, and upgraded to include the disposal program in August 1995. This document:

- established a prioritization basis for sampling which integrates known safety and disposal programmatic needs.

- defined the key waste tanks to be sampled based on grouping tanks into similar categories and selecting tanks to answer specific safety questions.

Four (4) core sampling crews were trained and certified.

Data quality objectives were issued for the five (5) primary sampling needs: safety program; retrieval, pretreatment, and disposal program; waste compatibility; historical model evaluation; and privatization waste characterization.

Analytical laboratories were upgraded, and the goal throughput of five (5) analytical equivalent units was achieved in October 1995 for the first time.

These changes resulted in a ten-fold increase in the number of samples taken during CY1995 compared to CY1994. This was particularly important for the full-length core samples where 49 core samples were obtained compared to five (5) core samples during CY1994. Sample loads through the analytical laboratory more than doubled. Finally, by the end of CY1995, 116 of the 177 underground storage tanks had been sampled using one (1) of the five (5) sampling methods listed above.

Improvements in characterization capabilities planned for this year include:

1. Deployment of a core sampling system capable of retaining gas in the core samples and then analyzing those samples for the gases that are trapped in the waste.
2. Deployment of a cone penetrometer system for in-situ measurements of rheological properties of the waste and moisture content (using a neutron moisture probe). The cone penetrometer includes a Raman spectroscopy system capable of in-place speciation.
3. Deployment of Surface Moisture Measurement System capable of measuring the moisture content of surface wastes within the tanks up to 6 feet off-center below the 4-inch tank risers into the tank vapor space.
4. Deployment of a Light Duty Utility Arm capable of robotic operations over 10 feet off- center in the vapor space below 12-inch risers into the tanks. This system will provide significant flexibility for operations within the tanks including: inspection, sampling, gripping, and cutting operations.
5. Redesign of some of the core sampling equipment to allow rotary sampling (necessary to retrieve samples from very hard wastes) from tanks that could potentially contain explosive mixtures of gases within the waste.

SUPPORT OF DOE PRIVATIZATION EFFORTS

In September 1995, DOE announced its intent to "privatize" the disposal part of the tank waste remediation program. The idea was to turn clean-up of Hanford's tank waste over to a private company that would do the design work and pay construction costs without Federal appropriations. The company would then be paid for the glass waste logs it produced. The privatization would be done in two (2) phases: 1) design and construction of waste treatment, and immobilization facilities for a small fraction of the waste (6-13%), followed by; 2) design and construction of waste retrieval, treatment, and immobilization facilities for the bulk of the waste. Characteristics of the two (2) phases are summarized in Table I (Page 11). During Phase I, waste retrieval would be performed by the existing Site contractor.

Table I

A draft Request for Proposal (RFP) was issued for comment in November 1995, and a final RFP is expected to be released in March 1995 for the first phase. In August 1996, up to three (3) companies will be selected for more in-depth design work. Each company will design a prototype vitrification plant to immobilize the low-level radioactive wastes. Bidders will also have the option of adding a prototype plant to glassify high-level radioactive waste. In February 1998, DOE will pick the best two (2) of the three (3) bidders proposals to build the low-level waste plants with hot operations to begin by December 2002. It is possible that a third plant to vitrify high-level wastes will be authorized for the same time period.

DOE's goal is for the two (2) prototype plants to process 13,200 (~3%) of waste in the first 2l/2 years and another 24,200 tons (~5-6%) in the second 2l/2 years. As the first phase nears completion, DOE will put out an RFP to build two (2) larger low-level waste vitrification plants, plus a full-scale high-level waste vitrification plant. The second phase bidding process will be open to any company interested, not to just the successful Phase I bidders. Construction of the larger second-phase low-level waste plants is scheduled to begin in 2008. The second-phase high-level waste plant is to begin operating in 2010, and the low-level waste plants are to start in 2011. All of Hanford's liquid radioactive wastes are to be immobilized by 2028.

SUMMARY

To date, 8 tanks have been "characterized"; that is, present requirements for the tank waste information for those tanks have been met. Several tank farms have been designated controlled, clean, and stable. 1995 was a year in which giant steps were taken on the path of real progress in closure of safety issues and ultimate disposal of tank waste.

28-3

IN-TANK PRETREATMENT OF HIGH-LEVEL TANK WASTES: THE SIPS SYSTEM*

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ABSTRACT

A new approach, termed SIPS (Small In-Tank Processing System), that enables the in-tank processing and separation of high-level tank wastes into high-level waste (HLW) and low- level waste (LLW) streams that are suitable for vitrification, is described. Presently proposed pretreatment systems, such as enhanced sludge washing (ESW) and TRUEX, require that the high-level tank wastes be retrieved and pumped to a large, centralized processing facility, where the various waste components are separated into a relatively small, radioactively concentrated stream (HLW), and a relatively large, predominantly non-radioactive stream (LLW). In SIPS, a small process module, typically on the order of 1 meter in diameter and 4 meters in length, is inserted into a tank. During a period of approximately six months, it processes the solid/liquid materials in the tank, separating them into liquid HLW and liquid LLW output streams that are pumped away in two small diameter (typically 3 cm o.d.) pipes. The SIPS module would be serviced by six small diameter (~3 cm o.d.) pipes - the two output pipes for the HLW and LLW streams

mentioned above, a water input pipe, a nitric acid (~3 M) input pipe, and input/output pipes to hydraulically load/unload ion exchange beads. An illustrative SIPS processing cycle is described. During the first half of the cycle, solid/liquid slurry from the tank is admitted to an internal compartment in the module (Compartment "B"), where the material is water washed. The liquid component flows to another internal compartment (Compartment "D") where cesium is extracted by an inorganic ion exchange bead material (e.g., silico-titanates). The cesium free liquid then flows out as part of the LLW output stream. The solid particle components remaining after water washing are trapped using a magnetic particle seeding/trapping process. After several hours of washing/trapping, the solid/liquid feed from the tank is switched to Compartment "C", while the remaining trapped solids in Compartment "B" are dissolved by nitric acid. The resultant acidified liquid HLW stream is then contacted in a fourth compartment ("E") with a selective extractant (e.g., a chelating polymer with ultra filtration) that removes a portion of the non-radioactive component in HLW streams so that they can be combined with the LLW stream. This reduces the volume of the HLW to be vitrified. Compartments "B" and "C" alternate functions every several hours. When "B" is water washing, "C" is acid dissolving; they then switch functions, and "C" washes while "B" acid dissolves. The ion exchange material in Compartment "D", together with the extracted cesium, is hydraulically unloaded every several days, and a fresh change of material hydraulically loaded. When the ~6 month processing interval is completed, the SIPS module would be moved to a fresh tank. Using an array of only 10 modules, the total inventory of 179 Hanford tanks could be processed in a few years. The construction and operation parameters for the SIPS module, based on Hanford process data, are described.

The SIPS approval appears attractive. It enables a lower cost and more redundant and maintainable processing system. It is very flexible, and can readily adapt to continuing technology improvements and widely disparate feed streams. It can be developed quickly, and is amenable to privatization.

DESCRIPTION OF THE SIPS CONCEPT

SIPS (SIPS In-Tank Pretreatment System), utilizes a small, low cost module that is lowered into a tank to pretreat the waste and separate it into LLW and HLW streams. Virtually all solids are dissolved inside the tank, so that the waste streams leave the tanks as solutions, not slurries.

Figure 1A shows an overall view of the SIPS module inside a waste tank. The module is lowered through a small opening (either preexisting or added) in the tank cover into the tank, and positioned either above or under the liquid surface. Small diameter flexible lines are connected to the intake and discharge ports in the module. Two small diameter lines lead into the tank carrying waste and nitric acid, while two small lines carry out the LLW and dissolved HLW streams. These connect to external transfer lines that lead to final processing facilities.

Fig. 1

Two designs of the SIPS module have been investigated, a "baseline" design, and an "advanced design." The baseline design essentially serves as the head end of the TRUEX process. The supernate and water soluble solids in the tank flow out as the LLW waste stream to an external central facility that extracts 99+% of the Cs (possibly also Tc and complexed Sr). The remaining solids are then dissolved in 3M nitric acid

and flow out as the HLW stream, to be further processed in an external facility.

In the advanced design, the SIPS module removes Cs (Tc and Sr also, if required) from the LLW stream using an inorganic ion exchange material so that the LLW stream requires no additional processing before vitrification. A separate processing zone in the advanced module extracts a portion of the non-radioactive components in the HLW stream, adding them to the LLW stream. No additional processing of the HLW is then necessary.

Figure 2 shows a "black-box" view of the baseline SIPS concept. Four small diameter (~3 cm) liquid flow pipes are attached to the support column that holds the SIPS module. Two of the pipes transport process feed liquids into the SIPS module from external lines outside the tank, i.e., H2O with colloidal ferromagnetic particles, and nitric acid solution (~3 M). The other two pipes transport process liquids away from the SIPS module to external lines outside the tank - i.e., liquid supernate to the cesium extraction and LLW vitrification facility, and nitric acid with dissolved HLW solids to the TRUEX processing and HLW vitrification facility.

Fig. 2

As illustrated in Fig. 3, colloidal ferromagnetic particles suspended in stream 1 are first combined with a slurry of tank solids and liquids (stream 2) in the mixing chamber (Region A). The colloidal ferromagnetic particles absorb on and bind to the solid particles. The slurry mixture then flows into a region where the particles that are not soluble in water are magnetically trapped. Based on the large body of experiments and process experience with magnetic separation, the undissolved particles should be quantitatively removed from the slurry that flow through Region B, leaving only a clear supernate that contains the water soluble materials (including cesium). A portion of this supernate flows back into the tank (Stream 5), with the remainder flowing out of the tank to the LLW processing facility (Stream 4). While the nonsoluble solids (i.e., the HLW solids) trapping process is being carried out in Region B, the previously trapped solids are being dissolved in Region C by nitric acid (Stream 6) fed from outside the tank. The dissolved solution (Stream 7) then flows out from the tank to an external processing facility, and ultimately to a vitrification unit where the concentrated HLW is converted to glass. After a suitable time interval, the flows through Regions B and C are interchanged, so that the trapped HLW solids in Region B are dissolved, and fresh HLW solids are trapped in Region C. The interchange of flows relies only on a simple on-off flow control valve. A total of 11 valves is required. These could be hydraulically or electrically actuated. If desired, all valves could be located outside of the tank for easy accessibility.

Fig. 3

The flow rates are small, and the process equipment volume in the SIPS module and its associated piping are quite reasonable. Table I gives the average values for the process flow rates, based on the Hanford TWRS flow sheet (1) and the stated overall efficiency for simple sludge washing. The integrated total volume of the LLW stream for all Hanford tanks is 251,490 kilogallons. For an inventory of 177 tanks, this corresponds to an average LLW flow volume of 1420 kilogal per tank. Over a 6-month processing interval this would require an H2O feed rate of 5.4 gal per minute into the tank, and an equal LLW removal rate of 5.4 gal/minute out

of the tank. Based on a feed pipe inside diameter of 2.5 cm for these streams, the corresponding flow velocities would be 67 cm/sec (~2 ft/sec). The pressure drop at this flow velocity is very small, on the order of 10-3 atmosphere (0.015 psi) per meter of length. Even smaller pipes could be used, if desired. The TWRS flow sheet shows the total mass of the vitrified HLW waste as 2.28x104 metric tons. Based on the stated loading of 45% by weight of waste oxides in the vitrified HLW output, 1.03x107 kilogram of waste oxides is carried in the 36,040 kilogal of sludge sent to the HLW plant. This corresponds to an average solids loading of 75 grams per liter in the liquid/solid sludge flow stream.
Table I
The flow rates for streams 6 and 7 account for the fact that: a) the mass of solids left after washing by stream 1 will be greater because simple sludge washing - i.e., washing with H2O - is used rather than enhanced sludge washing, which involved the use of 3M NaOH to reduce the volume of HLW solids; and b) there will be additional colloidal ferromagnetic particles added to trap out the solids. In addition, the equivalent solids loading in the solution after dissolution by nitric acid may not equal 75 grams/liter. With regard to point a), Westinghouse states that enhanced sludge washing removes an additional 40% of the solids that remain after simple sludge washing. With regard to point b), 20% additional solids in the form of colloidal ferromagnetic particles are included. With regard to the last point, the equivalent solids (i.e., dissolved) loading in the dissolved solution (stream 7) to be 50 grams per liter, or only 66% of the loading in the TWRS flow sheet. Even with conservative assumptions, the flow rate of dissolved HLW solution from a SIPS module in the waste tank is quite small, and easily handled in a 2.5 cm diameter line. With more concentrated solutions even smaller lines could be used.

MAGNETIC TRAPPING OF SLUDGE PARTICLES

The magnetic trapping process is illustrated in Fig. 4. Water containing colloidal ferromagnetic particles, e.g., iron oxide, is mixed with tank sludge. The colloidal ferromagnetic particles adsorb on the solid particles in the sludge. The combination is attracted to, and held on surfaces where the magnetic field concentrates. This can be provided by contact between ferromagnetic bodies with different geometric shapes (e.g., teeth or cones contacting a plate, particle beds, packed filters, etc.) or by periodically varying the dimensions of the ferromagnetic body (e.e., a helical screw surface). A toroidal magnetic geometry minimizes the magnetizing requirements. Liquid flows radially inwards along the module. Radial flow geometry allows a lower liquid velocity than axial flow, resulting in less hydrodynamic drag on trapped particles. Ferromagnetic particles suspended in the liquid are attracted to zones of higher magnetic field, and stick to surfaces where the magnetic field lines converge. One possible trapping arrangement uses toroidal iron plates that are flat on one side and bumpy on the other. Magnetic flux is greatest where the iron projections contact the bottom of the adjacent plate. Ferromagnetic particles will tend to concentrate at these contact points so as to decrease the magnetic reluctance of the toroidal circuit. Flat-topped cones enable good magnetic contact between adjacent plates, and result in flow passages that are well interconnected with a good open area for flow. Other types of geometries can be considered, including iron fibers or balls between flat plates.
Fig. 4

The ferromagnetic material in the SIPS module must not react (i.e., dissolve or rust) in the water or acid wash or with the trapped solids. Uncoated iron or steel probably is not satisfactory; however, it can be treated with a protective coating. Non-reactive ferromagnetic material, such as nickel, can also be used. Magnetic excitation is necessary to create the magnetic flux in the toroidal circuit. This can be provided by a current winding, or by excitation from high coercive force permanent magnets, such as neodymium-iron. Only about 2 percent of the magnetic circuit has to be neodymium-iron; the remainder is ordinary iron/steel or nickel. If a current winding excitation is used, the current can be de-energized whenever it is desired to flush away any particles that remain after the acid dissolution step. If permanent magnet excitation is used, its excitation can be temporarily nulled out by using a pulsed current winding. The magnetic field in the toroidal trapping region would then go to zero when the pulsed deexcitation winding is turned on.

The excitation power to energize the field in both Regions B and C is low, i.e., only about 13 kilowatts. This is very reasonable. The conductor winding weighs approximately 800 kilograms; this weight could be reduced by a factor of four with an aluminum winding. If permanent magnet excitation is used, the time-average magnetic excitation power would be much less - below a kilowatt - because the deexcitation winding would be on for a very short time.

PROCESS PERFORMANCE OF THE BASELINE SIPS MODULE

Estimates of SIPS performance are based on TWRS experiments (2) and design studies (1). SIPS can operate in a batch wash mode. However, this is not as efficient as the continuous wash mode. In the continuous mode, sludge particles are pre-mixed with water containing colloidal ferromagnetic particles (Region A) and then continuously flow into Region B or C (depending on which phase of the process cycle the module is operating in), where they are magnetically trapped and washed. The soluble solids dissolve, leaving behind the insoluble solids which are dissolved later by HNO_3, during the acid dissolution phase of the cycle. The volume of the trapped insoluble solids builds up with time as the water wash phase proceeds. (In addition, there is always some volume of trapped soluble solids that has not yet dissolved.) After four hours, the total volume of trapped insoluble solids is about 11 gallons. However, approximately 200 gallons of soluble salts have washed away, so that the insoluble fraction remaining after washing is only 5.2%.

A long wash period, e.g., four hours, is adopted. The insoluble solids occupy only about 11% of the flow passage area. Adding the soluble solids that have not yet dissolved, the total fraction of the flow passage area occupied by solids is about 25%, assuming a period of 20 minutes to dissolve soluble solids. With a typical flow passage dimension of ~1 millimeter, this corresponds to an average solids thickness of ~250 microns. This should not significantly change flow hydraulics in the trapping region. The average flow velocity through the passages in the trapping region is very small, about 0.05 centimeters per second. The wash period of four hours should dissolve all soluble solids. There probably would be a wash-only stage at the end during which fresh solids were not introduced. This wash-only stage would be 20-30 minutes out of a 4-hour cycle.

The corresponding 4-hour-long acid dissolution phase for the residual solid appears adequate, based on experiments by Lumetta (2). Undissolved solids would be flushed out into the tank, to reenter later for

additional treatment during the 6-month-long processing campaign. At the end of the campaign, there should be a very small amount of residual undissolved solids left in the waste tank. If 5% of the HLW solids that remained after simple washing (which itself removed 95% of the solids) did not dissolve in the acid dissolution phase, there would be about 500 gallons of undissolved solids in the tank. These could be treated for an additional month or so with special solutions to dissolve most of the remaining material. However, even if no further dissolution occurred, the residual solid volume would still be well below the allowable level of <1% specified in the TPA. It appears likely that special processing will remove virtually all of the 500 gallons of residual solids.

ADVANCED SIPS MODULES

The baseline SIPS module generates LLW and HLW liquid waste streams that go to an external facility for further processing. Cesium is extracted from the LLW stream by ion exchange, while most of the non-radioactive components in the HLW stream are extracted by the TRUEX process.

The advanced SIPS module would eliminate external processing. To achieve this, the advanced SIPS module must remove 1) 99% of the cesium in the LLW stream before it leaves the module, and 2) 75% of the non-radioactive components from the HLW stream before it leaves the module [removal fraction may depend on the element being removed] while retaining 99% of the radioactive species present in the HLW stream. The process technologies for these two functions must be compatible with the relatively small volume of the SIPS module. Moreover, they also must be simple to operate, reliable, and of acceptable cost.

Candidate technologies include both existing processes, as well as promising new processes under development. The 1995 Efficient Separations and Processing Cross Cutting Program (ESP) Annual Technical Exchange Meeting summarizes some of these promising new processes (3).

The cesium removal rate in an advanced SIPS module will be about 0.25 kg per week, assuming a 6-month period to process the tank. Ion-exchange is a proven technology and compatible with a small SIPS module if the IEX material is periodically replaced, i.e., every week or so. It appears simple to hydraulically load and unload IEX beads from the module; alternatively, the spent IEX beads could be periodically dissolved and added to the HLW stream. For simplicity, a non-eluting, inorganic type of IEX material is desirable. Promising new ion-exchange materials have been developed for cesium extraction from tank waste solutions. One of the most attractive are the crystalline silico-titantates (CST). These show very high selectivity for cesium (4) at the conditions expected for the LLW stream: highly alkaline (pH ~15), high sodium concentration (~5 M Na), and low cesium concentration (~2 ppm). Distribution coefficients for cesium extraction of ~2000 mL/g have been measured for CST in simulated alkaline Hanford waste (5). Based on these experiments, the absorptive capacity of CST in an ion exchange column with a 2 ppm cesium feed is estimated as~4x10-3 g cesium per g of absorber. This corresponds to a CST inventory in the SIPS module of ~80 Kg (30% extra above saturation to allow for a breakthrough margin) if the ion-exchange material were to be replaced on a weekly basis, [~35 kg if replaced on a semiweekly basis]. The corresponding volume in the SIPS module is well below 100 liters. The replacement schedule appears acceptable. The volume of the HLW solids to be vitrified increase by ~5% if the spent IEX material is added to the HLW stream without further processing. This minor increase would not significantly impact vitrification cost, nor should it affect repository

cost and availability. If Sr is present in significant concentration in the LLW stream from the SIPS module, an inorganic IEX material could extract it. Distribution coefficients in excess of 10,000 have been demonstrated for Sr removal from alkaline LLW type waste streams using sodium titanate. The amount of IEX material to remove Sr removal would be considerably smaller than for cesium extraction.

There are a number of candidate technologies to remove non-radioactive components from the HLW stream. Solvent extraction processes are probably not satisfactory for an advanced SIPS. A water soluble chelating polymer/ultra-filtration process appears attractive. Retentivity values of 99.9% (much greater than required) of plutonium and americium have been demonstrated under acidic conditions (6). Chelating polymers for retaining cesium and strontium require development. Another candidate is fractional precipitation; here, water would be extracted from the HLW stream using reverse osmosis, resulting in a fractional precipitation of the dissolved solids. Since the non-radioactive components are present at very low concentrations, i.e., a few parts per million, non-radioactive components will preferentially precipitate, which would then be transferred to the LLW stream. However, since co-precipitation will undoubtedly occur, it probably will be necessary to fractionally precipitate several times to adequately decontaminate the separated non-radioactive material.

SUMMARY

The SIPS concept appears attractive for pretreating high level wastes, since it would: 1) process waste in-situ in the tanks, 2) be cheaper and more reliable than a larger centralized facility, 3) be quickly demonstrable at full scale, 4) have less technical risk, 5) avoid having to transfer unstable slurries for long distances, and 6) be simple to decommission and dispose of. Further investigation of the SIPS concept appears desirable, including experimental testing and development of subscale demonstration units.

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UNDERSTANDING WASTE PHENOMENOLOGY TO REDUCE THE AMOUNT OF SAMPLING AND ANALYSIS REQUIRED TO RESOLVE HANFORD WASTE TANK SAFETY ISSUES

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ABSTRACT

Safety issues associated with Hanford Site waste tanks arose because of inadequate safety analyses and high levels of uncertainty over the

release of radioactivity resulting from condensed phase exothermic chemical reactions (organic solvent fires, organic complexant-nitrate reactions, and ferrocyanide-nitrate reactions). The approach to resolving the Organic Complexant, Organic Solvent, and Ferrocyanide safety issues has changed considerably since 1990. The approach formerly utilized core sampling and extensive analysis of the samples with the expectation the data would provide insight into the hazard. This resulted in high costs and the generation of a large amount of data that was of limited value in resolving the safety issues. The new approach relies on an understanding of the hazard phenomenology to focus sampling and analysis on those analytes that are key to ensuring safe storage of the waste.

INTRODUCTION

Safety issues arose for the 177 (149 single-shell and 28 double-shell) Hanford Site underground storage tanks because of uncertainty over the release of radioactivity resulting from condensed phase exothermic chemical reactions. There are three distinct safety issues that involve condensed phase exothermic chemical reactions: organic solvent fires, organic complexant-nitrate reactions, and ferrocyanide nitrate reactions. A comprehensive safety program was begun in 1990 to examine and resolve these safety issues. Only limited understanding of the hazards existed; consequently, sampling and analysis activities were directed towards sampling the condensed phase and performing complete suites of analyses. The large number of analyses conducted on each waste sample resulted in few tanks being sampled and a large quantity of data that had only limited use in resolving the safety issues.

Much progress has been made on understanding the waste phenomenology and the safety hazards. This understanding has focused sampling effort and narrowed the suite of analyses to those contributing to resolution of the safety issues. The following sections review the background and current understanding of the waste phenomenology, as well as characterization needs based on the current understanding of the safety issues.

ORGANIC SOLVENT SAFETY ISSUE

Background

Various separation processes involving organic solvents have been used at the Hanford Site. These organic solvents were sent to the underground storage tanks, and subsequent waste transfer operations might have distributed organic solvent among several of the 177 high-level waste tanks (1). Given a sufficient ignition source, there are two potential hazards associated with organic solvent, an organic solvent pool fire, and ignition of organic solvent that is entrained in waste solids (a wick fire). A solvent fire would generate and heat headspace gases, pressurizing the tank. This pressurization might be significant enough to collapse the tank dome, releasing radioactive material to the environs.

Hazard Phenomenology

The organic solvent used at Hanford contains semivolatiles compounds (e.g., dodecane, tridecane, and tributyl phosphate); therefore, there is a direct relationship between liquid organic solvent in a tank and the organic solvent vapors found in the headspace. Tanks containing organic solvent can be identified by vapor sampling the tank headspaces. The mass transfer of semi-volatile species in an organic liquid to the headspace vapor is determined by several parameters, including the mass transfer coefficient, gas-liquid contact area, ventilation flow rate, and solvent volatility.

If organic solvent is present, organic solvent vapors should be detectable in a tank headspace, even if the tank is actively ventilated. Most single-shell tanks are only passively ventilated (i.e., the ventilation flow rate is quite small); thus the organic solvent vapors should be present at roughly equilibrium concentrations. Indeed, this was the case for tank 241-C-103, which contains a floating organic solvent pool. Analyses of the liquid organic solvent pool and the vapors in the headspace suggest that semivolatile concentrations are present at close-to-equilibrium concentrations in the tank headspace.

Currently, one tank (241-C-103) is known to contain an organic solvent pool. Additional tanks that might contain an organic solvent pool will be identified through vapor sampling of the tank headspaces. Criteria for organic solvent headspace concentrations have been developed using theoretical analyses and organic solvent sample data from tank 241-C-103. All 149 single-shell tanks will be vapor sampled and screened against the criteria to identify potential organic solvent tanks.

The 28 double-shell tanks do not require headspace sampling for organic solvent. Double-shell tank design will accommodate a substantially larger pressure transient than single-shell tanks. An organic pool or wick fire could not build enough pressure to collapse the double-shell tank dome. Therefore, this safety issue only applies to single-shell tanks.

Organic solvent pool fires are difficult to ignite (2). Sparks, impacts, shocks, and friction sources lack sufficient energy to ignite pool fires, and credible ignition sources have been narrowed to robust and/or sustained energy sources. Experiments with a waste simulant (dodecane) required exposure to a propane torch for 10 - 15 seconds (>1000 Watts) to ignite a 5cm puddle.

Even if a pool fire could be ignited, consequences from such a fire would be very low (within risk acceptance guidelines) if an adequate vent path area exists. A pool fire would heat headspace gases pressurizing the tank. The fire would burn until the oxygen was depleted. The pressurization from a postulated pool fire would increase with the fire spread rate. Calculations indicate that a puddle would have to be larger than 2m2 to create enough pressure to collapse the tank dome. If adequate vent path area was available, the tank dome would not collapse and any radioactive release would be minor. Of the 43 tanks recently vapor sampled, only one has a floating organic solvent pool (241-C-103) and adequate venting is available in this tank (2).

Entrained organic solvent is also difficult to ignite. Hot steel spheres (greater than 270 Joules) and an electronic match (about 138 Joules) failed to ignite entrained organic solvent (dodecane) during ignition experiments. Sparks, impacts, shocks, and friction sources could not ignite entrained organic solvent either.

The consequences from an entrained organic solvent fire are less than an organic pool fire. Open literature and preliminary calculations show that the spread rate for an entrained solvent fire is about an order of magnitude lower then that for a pool fire and would not result in tank over-pressurization. Therefore, the safety issue is bounded by the organic solvent pool fire hazard.

Resolution of the organic solvent safety issue requires two steps a)identification of tanks containing significant quantities of organic solvent (i.e., greater than a 2m2 puddle), and b)ensuring adequate vent path in those tanks that contain significant organic solvent.

Characterization Needs

By understanding hazard phenomenology, the characterization needs can be reduced to vapor sampling of single-shell tanks. Core sampling of the tanks is not required to resolve this safety issue. Only the 149 single-shell tanks need to be vapor sampled for the presence of organic solvent and this will be completed by the end of FY 1998.

ORGANIC COMPLEXANT SAFETY ISSUE

Background

Organic complexants were sent to the high-level waste tanks during the defense mission at the Hanford Site. These compounds and their decomposition products have the potential to react exothermically when combined with an oxidizer. The waste tanks also contain high quantities of sodium nitrate, a strong oxidizer. The organic complexant hazard is represented by two distinct types of reactions, spontaneous chemical runaway (self heating) reactions, and propagating chemical reactions typified by a passing reaction front. These reactions could produce gases that pressurize the tank and cause failure of the tank dome. The aerosols produced would carry radioactive materials to the environs, resulting in unacceptable onsite and offsite doses.

Hazard Phenomenology

Spontaneous chemical runaway reactions are not possible under current storage conditions. This conclusion was reached by evaluating the energy balance for the tanks. For a spontaneous chemical runaway reaction to occur, the chemical heating rate must exceed the tank cooling rate (3). This can be evaluated by comparing the tank cooling response time (i.e., the time required to reach a new equilibrium temperature following an instantaneous change in the heating rate) with the waste storage time. Calculations show that the tank cooling response times range from a few hours to 3.1 years. Some waste has been stored for more than 40 years, and there has been no transfer of waste into the single-shell tanks for about 15 years. Several cooling response times have passed over the last 15 years of storage; consequently, it can be concluded that bulk runaway reactions are not a hazard under current storage conditions. In addition, no credible mechanisms have been identified that would increase tank temperatures and cause a chemical runaway reaction.

Propagating reactions require an ignition source and sufficient fuel and oxidizer. Tube propagation tests with waste surrogates and theoretical analyses have shown that ignition sources greater than 1 Joule (10,000 times more than that required to ignite flammable gas) are required to initiate organic complexant reactions in dry waste. Sparks and static electric shocks lack sufficient energy to initiate organic complexant propagating reactions.

It may be possible to show that no credible initiators exist that cannot be controlled. Laboratory experiments on simulants indicate about 5 weight percent (wt%) free water will prevent initiation of propagating reactions by 100 J ignition sources. Tests on actual waste samples will be conducted to determine a minimum moisture concentration under in-tank waste storage conditions. Confirmation that waste free water concentrations exceed approximately 5 wt%, would reduce the amount of sampling and analysis required to resolve the safety issue.

Safe storage criteria have been established through theoretical analysis and tests on waste surrogates. The minimum fuel concentration required to support a propagating reaction has been determined using a contact-temperature ignition model (4). A necessary (but not sufficient) condition for a propagating reaction is that the fuel concentration be

greater than 1200 J/g (4.5 wt%total organic carbon), on an energy equivalent basis (4).

For fuel concentrations between 1200 and 2100J/g, the waste moisture (free water) content required to prevent a propagating reaction varies linearly from 0 to 20 wt%. Above 20 wt%, the fuel-moisture linear relationship no longer holds because the mixture becomes liquid continuous and a stoichiometric fuel-oxidizer mixture will not support a propagating reaction (4). All of the waste in the double-shell tanks have aqueous supernatants and the waste would be too wet to support a propagating reaction.

Experiments on waste simulants indicate that fuel concentrations in the tanks have been decreased by saltwell pumping and waste aging (i.e., decomposition of the high energy waste into low energy products). Experiments show that the reactive organic complexant salts (e.g., sodium acetate, EDTA, HEDTA) remain soluble in the tank solutions and are removed by saltwell pumping (5). Most of the single-shell tanks have been saltwell pumped (113 out of the 149 total) and the liquid sent to the double-shell tanks. Hence, much of the fuel values in the single-shell tanks might have already been removed.

Experiments also indicate that organic compounds age to less energetic products (e.g., oxalate) (6). Laboratory experiments have shown that these organic salts will not support propagating reactions (4). Therefore, any remaining fuel in the single-shell tanks might not be reactive enough to be a hazard.

It might be possible to show that the fuel concentrations in the single-shell tanks are too low to support propagating reactions. The solubility model indicates that energetic organic species are present principally as solutes in tank liquids. The aging model indicates that organic complexants age over time to less energetic species, providing additional confidence regarding the stability of stored waste. Organic speciation analyses of actual waste will be conducted on selected tanks to confirm that the organic complexants remained soluble, and to confirm that any remaining fuel has aged.

It might be possible to show that tank waste contents meet the safe storage criteria. Information from the solubility and aging models and sample data from selected tanks will be used to assess the potential fuel and moisture concentrations in the tanks. In situ moisture monitoring capability is being developed in an attempt to provide representative measurements of waste moisture concentration in the single-shell tanks.

Characterization Needs

From an understanding of hazard phenomenology, the sampling needs can be focused and the number of analyses performed can be reduced. Only selected single-shell tanks require sampling to resolve the safety issue. Analyses have been reduced to include only measurement of energetics, total organic carbon, moisture, and some organic speciation to confirm organic solubility and organic aging. If tests on actual waste samples indicate that the waste will retain 5 wt% free water, the sampling and analysis needs will be reduced further.

FERROCYANIDE SAFETY ISSUE

Background

During the 1950s, additional tank storage space was required to support the defense mission. To obtain this additional storage volume within a short time period, and to minimize the need for constructing additional storage tanks, Hanford Site scientists developed a process to scavenge

cesium-137 from tank waste liquids. In implementing this process, approximately 140 metric tons of ferrocyanide were added to waste that was later routed to eighteen Hanford Site single-shell tanks. Ferrocyanide, in the presence of oxidizing material such as sodium nitrate, can propagate by heating it to high temperatures or by providing an electrical spark of sufficient energy. Because the scavenging process precipitated ferrocyanide from solutions containing nitrate, an intimate mixture of ferrocyanides and nitrates may have existed in some regions of the eighteen ferrocyanide tanks.

Hazard Phenomenology

Ferrocyanide ages (i.e., decomposes to lower energy products) when exposed to tank waste conditions. Three parameters strongly affect the rate of aging, temperature, exposure to high pH, and radiation dose (7). The current fuel concentration is a function of the starting concentration and the amount of aging that has occurred. Historical data (8) show that all the ferrocyanide tanks have been exposed to enough caustic to promote aging (i.e., had pH values higher than 10). However, ferrocyanide sludge depths in the tanks range from 0.1 to 2.6 meters (9) and there is some question whether the caustic solutions would penetrate more than one meter into ferrocyanide sludge (10). Therefore, ferrocyanide at greater depths might not have been exposed to high pH solutions and might not have aged.

Tanks with high ferrocyanide concentrations and sludge depths, and low temperature and radiation dose histories, have been selected for core sampling to bound aging. If the ferrocyanide has aged in these tanks, then as much or more aging should have occurred in the remaining ferrocyanide tanks.

Nine ferrocyanide tanks have been selected for sampling and analysis to bound ferrocyanide aging. Sampling and analysis have been completed for six of these nine tanks and the data is reviewed in Table I (11). Results indicate that ferrocyanide has aged to concentrations more than a factor of ten lower than the original concentrations. The remaining ferrocyanide concentrations are substantially below the 1200J/g minimum required to support a propagating reaction (4).

Table I

Characterization Needs

Understanding the hazard phenomenology has reduced the number of ferrocyanide tanks that need to be sampled from all eighteen to the nine that bound aging. The analyses required to resolve the safety issue have been narrowed to energetics and ferrocyanide [Na2NiFe(CN)6]. Six of the nine tanks that bound ferrocyanide aging have been sampled and analyzed, and only three tanks remain to be sampled. If aging is confirmed in these final three tanks as expected, then no additional sampling is required to resolve the ferrocyanide safety issue.

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NEW INSTRUMENTS FOR CHARACTERIZATION OF HIGH-LEVEL WASTE STORAGE TANKS AT THE HANFORD SITE

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ABSTRACT

Four new instruments (a waste void fraction meter, a waste viscometer, a surface- moisture measurement system, and a cone penetrometer) have been developed at the Hanford Site for in-situ characterization of selected physical and chemical properties of high-level waste in underground storage tanks at Hanford. The basis for development of these instruments, the design features, and measurement principles are described along with test results to-date. Planned measurements with the cone penetrometer and surface-moisture measurement system, which have not been deployed in the field, are described.

INTRODUCTION

Extensive efforts are underway to characterize high-level waste stored in 177 underground tanks at the Hanford Site in southeastern Washington

State. In addition to the collection and analysis of samples of vapor, supernate, sludge, and saltcake, several new instruments have been developed for in-situ measurements of waste properties. In some cases, these in-situ measurements may prove to be more representative of waste characteristics than samples that are withdrawn from the tanks. In other cases, the in-situ measurements are less expensive, because full-core samples and their attendant hot cell analysis can cost several hundred thousand dollars each.

Four new instruments have been developed at Hanford for in-situ characterization of selected physical and chemical properties of high-level waste in underground storage tanks at the Hanford Site. These instruments consist of a cone penetrometer, a surface-moisture measurement system, a waste void fraction meter, and a waste viscometer. Measurements performed to date are described for the void fraction meter and waste viscometer. Measurements that are planned later this year with the surface moisture monitor and the cone penetrometer, which have not yet been deployed, are described.

WASTE VOID FRACTION METER

A waste void fraction helps determine the quantity and vertical profile of potentially flammable gases, such as hydrogen, ammonia, and nitrous oxide, that are stored within the thick waste sludges. The design and application of this device was described by Stewart et al. (1995) (1). The quantity of stored gas in the tanks continues to be a high-priority safety issue at the Hanford Site. Efforts that are underway to assess and mitigate this issue have been described by Lentsch et al. (1994, 1995) (2,3).

The principle of the void fraction meter is to insert the device into the waste and capture a sample of waste in a steel chamber. After remote closure and sealing of the chamber, the sample is pressurized to approximately 500 lbf/in2 (gauge) with nitrogen. This compresses any gas voids in the waste sample. The change in volume of the sample is computed from precision measurement of the change in pressure of nitrogen in the supply cylinder. The fraction of the sample originally occupied by gas is then computed. The device is equipped with temperature sensors on the sample chamber, tubing, and supply cabinet for correction of computed volumes.

The void fraction meter is mounted on a 0.76-m-long pivot arm that is attached to a 21.34-m-long, 3-in.-diameter stainless steel pipe weighing approximately 589 kgs. The chamber inlet edges are sharpened to minimize waste disturbance and assure representative collection of the samples. After measurement, the chamber is opened pneumatically to exhaust the sample.

The instrument transducers and electronic processing modules are mounted in an enclosure on the end of the pipe on the outside of the tank. A computer control and data acquisition system is provided.

The entire assembly is lowered by crane into the tank. Measurements are typically performed from the top down, at 0.61-cm intervals, and repeated at 120-degree rotations of the arm. Following removal and decontamination, the device is deployed at a second location on the opposite side of the tank.

Four double-shell, high-level waste tanks have been measured to date (4). The results for three of these tanks are shown in Fig. 1. Significant variation is seen in void fraction, both vertically and laterally. This is not unexpected because the large volume of waste in the tanks is

unmixed. (The tanks are up to 22.86 m in diameter, and the waste is up to 10.67 m deep). The exception is Tank 101-SY, which is periodically mixed with a large mixer pump. Waste in this tank shows less variation.
Fig. 1
The thick, particulate-laden sludge at the bottom of the tanks contains void fractions up to approximately 13 percent, whereas the dense overlying supernate has less than 1 percent void. Average sludge void fractions range from 5 to 7 percent, corresponding to tank gas inventories of 2,000 to 6,000 scf.

WASTE VISCOMETER

A waste viscometer (ball rheometer) was developed to determine the density, yield strength, and viscosity of waste vertical profiles (1). This information is needed for safety analyses of potential flammable gas release accidents. It is also needed to design equipment such as mixer pumps for retrieval of waste from the tanks for waste treatment and disposal. Laboratory analyses of extruded waste cores have not provided accurate measurements of these parameters because of sample disturbance. The waste viscometer consists of a 5.90-kg, 8.89-cm-diameter tungsten ball that is suspended by cable in the waste. The tension in the cable is measured by a precision load cell mounted on a riser penetration above the tank. The cable tension is measured with the ball at rest, during initial movement, and during raising and lowering with the cable spool motor at various controlled velocities. The density, yield strength, and viscosity, respectively, can then be computed from these static loads, starting forces, and dynamic drag forces.
Four tanks have been measured to date with this instrument. The results for one of the tanks are displayed in Fig. 2. The density results show a heavy, convectively mixed, supernatant layer in this tank, with a 250-cm sludge layer. Densities are not accurate in the sludge, which has a yield strength of 50 to 350 Pa in this tank. The apparent viscosity of the supernate in this tank was initially found to be approximately 18 cP, whereas the sludge varies up to 30 million cP. After repeated passes of the ball, the sludge viscosity decreased by an order of magnitude.
Fig. 2

SURFACE MOISTURE MEASUREMENT SYSTEM

An instrument to measure the moisture content of the upper waste layer has been developed and is scheduled to be deployed in a waste tank later this year. The moisture content of the waste is an important safety parameter that strongly determines the combustibility of waste materials that contain appreciable quantities of organic chemicals and nitrates.Key components of the instrument are the deployment mast, Data Acquisition Van (DAV), a neutron probe, and a decontamination spray manifold. The deployment mast consists of a 15.24-m-long stainless steel pipe with a 1.83-m cable-operated drop arm and cable-suspended probe. An upper enclosure supports the mast on the tank top and contains manual winches, cable reels, and electronic components. The DAV supplies power to the probe electronics and processes the neutron information for moisture analysis. The probe consists of an 45.72-cm-long, 8.89-cm-diameter stainless steel certified pressure housing that contains the neutron signal conditioning electronics, three Boron-10-lined thermal neutron proportional detectors, and a Californium-252 fast neutron source. The deployment mast installs in a 10.16-cm tank riser penetration through a spray ring manifold. Spray nozzles, mounted in the manifold, allow high-pressure water to decontaminate the mast and probe upon removal.

A 1.83-m-radius circle, on the surface of the waste, can be measured. The 1.83-m arm allows measurement out from under the riser, where condensation and wash water may have altered the water content of the waste surface.

A video camera is installed in an adjacent tank riser to help position the neutron probe on the waste surface. Measurements will be made at multiple points inside the 1.83-m radius by rotating and repositioning the device.

The moisture concentration is computed from ratios of the count rates from the three neutron detectors, which detect thermal neutrons that are moderated by water in the upper approximate 20.32 cm of waste. Signal processing, of the neutron detector outputs, is used to screen out gamma ray contributions.

This system has been designed to measure down to 3 weight percent water with an accuracy of approximately 1 weight percent. Laboratory testing is underway. Test plans are to validate measurements with this device by correlating the results of surface moisture measurements with measurements of water in vertical waste profiles (by core sampling and laboratory analysis, or by vertical measurements in liquid observation wells that are installed in many of the waste tanks).

Following validation, the surface moisture device will be used for initial safety screening and perhaps to conduct routine future surveillance of tank moisture levels.

CONE PENETROMETER

A specialized cone penetrometer is completing development. This device is used in geological investigation for soil stratigraphy, by pushing a bevelled sensor package into the soil under a heavy load. The instrument being developed for waste tank characterization consists of a custom sensor package with compressive and shear strength load cells, temperature sensors, a pore pressure sensor, a neutron moisture probe, and a Raman spectrometer. The sensor package is shown in Fig. 3.

Hydraulic actuators will place a load of up to 32 metric tons onto the heavy-wall, 4.83-cm-diameter, steel penetrometer push rod. A heavy-wall, 15.24-cm steel guide tube extends through a tank riser down to the waste surface to prevent bucking of the push rod. Testing shows that this device should be capable of pushing the bevelled tip of the sensor package into fairly hard wastes with compressive strengths up to several thousand pounds per square inch. The tip of the sensor contains a magnetrometer to detect the steel tank bottom liner, preventing potential penetration of the tank. An inclinometer is also provided to detect push rod bending prior to buckling.

Fig. 3

A moisture probe using two Boron-10-lined thermal neutron proportional detectors and a Californium-252 fast-neutron source is provided. This probe is inserted into the penetrometer rod by cable after the rod has been inserted. Gamma pulses are screened out by signal processing. Neutron pulse rates are ratioed from the two detectors that are placed end-to-end in the probe at 30.48-cm center-to-center spacing. In this manner, a fairly linear response is obtained between the near-to-far detector ratio and waste water concentration from 5 to 100 percent, with minimal effect from neutron poisons.

The Raman probe directs a beam of near infrared light into the waste through a diamond window, and the reflected light is directed by fibre optics to a tank top spectrometer. This application is highly

developmental. It is hoped that qualitative data will be obtained on the vertical distribution of chemical species such as organics, ferrocyanides, and nitrates that are of concern for tank safety, in conjunction with waste temperature and moisture.

The tip and sleeve load cells on the sensor package will provide information on waste stratigraphy. The tanks are known to contain layers of crust, supernate, sludge, and hard saltcake. This data will be helpful in waste core sampling. It will also be invaluable in determining waste material properties for the design of waste retrieval methods (mixing, lancing, etc.).

This cone penetrometer is scheduled for deployment in the first tank in late 1996.

CONCLUSIONS

A waste void fraction meter and a waste viscometer have been successfully developed and deployed in four waste tanks. The results of these measurements have been useful in assessing the flammable gas safety issue, and in determining physical properties of high-level waste in underground storage tanks at the Hanford Site.

A waste surface moisture monitor and a specialized cone penetrometer have been developed for deployment later in 1996. They will measure waste moisture and physical properties on the surface and at depth. These measurements will be used to assess safety aspects of potential organic-nitrate reactions, to improve core sampling operations, and to design waste retrieval equipment and processes.

These instruments have and will continue to be used as cost-effective tools for tank waste characterization, thereby supplementing the current intensive core sampling and laboratory analysis efforts.

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ORGANIC REACTIVITY ANALYSIS IN HANFORD SINGLE-SHELL TANKS - EXPERIMENTAL AND MODELING BASIS FOR AN EXPANDED SAFETY CRITERION

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ABSTRACT

Large quantities of organic complexants were used in support of the defense mission at the Hanford Site, and are currently stored as part of 55 million gallons of radioactive waste. The major organic complexants that are believed to have been stored in the tanks are glycolate, citrate hydroxyethylethylenediaminetriacetate (HEDTA), and ethylenediameinetetraacetate (EDTA). Estimated quantities are listed here in kilograms and metric tons:

Glycolic acid, 8.8x105 kg (880 metric tons)
Citric acid, 8.5x105 kg (850 metric tons)
Hydroxyethylethylenediaminetriacetic acid, 8.5x105 kg (830 metric tons)
Ethylenediaminetetraacetic acid, 2.2x105 kg (220 metric tons).

In addition to these complexants, lesser amounts of complexants such as nitrilotriacetic (NTA), di(2-ethylhexyl) phosphoric, and oxalic acids were used, but amounts of these complexants that actually were stored in the tanks are not known.

Despite demonstrated safe storage in terms of chemical stability of the Hanford high level waste for many decades, including decreasing waste temperatures and continuing aging of chemicals to less energetic states, concerns continue relative to assurance of long-term safe storage. Review of potential chemical safety hazards has been of particular recent interest in response to serious incidents within the Nuclear Weapons Complexes in the former Soviet Union (the 1957 Kyshtym and the 1993 Tomsk-7 incidents).

Based upon an evaluation of the extensive new information and understanding that have developed over the last few years, it is concluded that the Hanford waste is stored safely and that concerns related to potential chemical safety hazards are not warranted. Spontaneous bulk runaway reactions of the Kyshtym incident type and other potential condensed-phase propagating reactions can be ruled out by assuring appropriate tank operating controls are in place and by limiting tank intrusive activities. This paper summarizes the technical basis for this position.

CONDENSED-PHASE CHEMICAL REACTIONS

The presence of organic complexants along with oxidizers like NaNO2 and NaNO3 present the potential for condensed-phase chemical reactions, including spontaneous runaway reactions of the Arrhenius type and propagating reactionsa initiated by a local ignition source.

SPONTANEOUS RUNAWAY REACTION

Spontaneous bulk runaway reaction without ignition sources can occur when the chemical heat generation rate produced by an Arrhenius type reaction mechanism exceeds the rate of heat dissipation by conduction in some volume of waste. The requirements for stability are therefore often expressed in terms of two characteristic time constants

Eq. 1

where ta is the characteristic time of adiabatic runaway

Eq. 2

Here T (K) is the temperature, (K/s) is the rate of temperature rise corresponding to temperature T due to chemical heating, Ea is the

activation energy (J/mol), R (8.314 J/mol-K) is the gas constant, H(m) is the height and a(m2/s) is the thermal diffusivity.
The characteristic cooling time tc for an infinite slab geometry is approximatelyb
Eq. 3
While values for tc can be reliably estimated for the waste tanks, values for ta are not known and would be difficult to obtain from waste sampling and analytical measurements.
Fortunately, this concern can be addressed by recognizing that the waste has remained safely stored over a period of time (the 149 single shell tanks have not received wastes since about 1977). That is well in excess of the characteristic cooling times, tc, which range from tens of hours to about 3 years (1). This observation provides proof that Inequality (1) is indeed satisfied, and can be made without a knowledge of ta or the actual chemical composition, water content and waste uniformity, and will not be altered by waste aging or long-term moisture loss. Both shrinkage associated with moisture loss and increasing gaseous porosity both have the effect of decreasing the characteristic cooling times (1).
CONDENSED-PHASE PROPAGATING REACTIONS
In addition to satisfying a certain minimum fuel concentration similar to the lower flammability limit (LFL) for gases, a significant (>10 J) ignition source must be available to assure a propagating reaction. Measured combustion characteristics for several non-degraded organic complexants are summarized in Table I below (2).
Especially noteworthy in Table I is the extremely low burn velocities, implying that if propagating reaction is possible, the chemical energy release rate and corresponding gas evolution rates would be small, limiting the possibility of tank structural tank damage and the release of significant quantities of radioactive material. Furthermore, since the Tig values are much higher than the current waste temperatures, onset of combustion also requires the presence of an adequate ignition source. Potential ignition sources are discussed below including sparks, flames, hot objects and lightning for example, refer to Table I (Sparks) and Table II (Flames). In the discussion to follow it is assumed that combustible conditions exist, i.e., the fuel concentration exceeds its LCL and that moisture concentrations required to suppress combustion are not available unless otherwise specified. The likelihood of satisfying the necessary fuel concentration for a propagating reaction is addressed later.
Table I
Table II
SPARKS
An estimate of the minimum ignition energy (MIE) to initiate combustion can be obtained from
Eq. 4
where r(kg/m3) is the density of the combustible mixture, c (J/kg-K) is the specific heat of the combustible mixture, Tig (K) is the ignition temperature, To (K) is the ambient temperature, and rcrit (m) is the critical radius which is determined from the following heat balance (heat release equals heat loss)
Eq. 5
where Tf (K) is the flame temperature and k (w/m-K) is the thermal conductivity, and results in
Eq. 6

where a(m2/s) is the thermal diffusivity.

First, we will demonstrate that the above ignition theory can be used to predict the MIE of a combustible stoichiometric gas mixture. Considering the following property values for a typical hydrocarbon gas mixture such as methane-air evaluated at Tig = 810 K; r 0.45 kg/m3, c = 1000 K, a = 9 10-5 m2/s, Ub = 0.45 m/s and To = 300 K, we estimate Qmin= 0.21 mJ, which is in excellent agreement with the measured spark energy value of 0.29 mJ (4). The suggested strong sensitivity to the burn velocity (Q ~ Ub-3) is also consistent with experimental observation.

Verification of the strong burn velocity dependence is significant to the application of condensed-phase combustion involving organic complexants such as included in Table I where the burn velocities are noted to be quite low. Application of the above ignition theory using the following property values evaluated at Tig = 520 K; r = 1500 kg/m3, c=1500 J/kg-K, a = 2 10-7 m2/s, Ub = 5 10-4 m/s and To = 320 K, lead to Qmin 3.3 J, i.e., about four orders of magnitude larger than for flammable hydro-carbon mixtures. This energy is well in excess of the maximum spark energies expected from discharge of various types of electrically conducting objects that could be present in the Hanford waste tanks. Typical maximum theoretical spark energies associated with various objects are illustrated in Table II (5).

The above assessment of the MIE value for condensed-phase combustion in case of short duration sparks is considered quite conservative. In contrast to a combustible gaseous mixture, the energy from the spark must be delivered to the condensed phase material. While the power density of the spark is extremely high, the short duration (-second time scale) prevents a sufficient thermal boundary layer build-up required for sustained combustion prior to the dissipation of the spark energy. Considering the previously established ignition theory, the critical ignition thickness for plane geometries is

Eq. 7

The corresponding ignition time, tig, can be obtained by equating the thermal boundary layer thickness

Eq. 8

with the critical ignition front thickness resulting in

Eq. 9

Setting a = 2 10-7 m2/s and Ub = 5 10-4 m/s, the critical ignition time is of the order of 1s. The significance of this value is that potential ignition sources, such as sparks, which have extremely high power densities but are also extremely short-lived (s range), are not capable of initiating condensed-phase propagating reactions in materials of interest. The thermal boundary layer resulting from such ignition sources are much too small to satisfy the requirement for sustained combustion.

FLAMES

In view of the relative ease of igniting a combustible gas mixture, it is of interest to evaluate its potential for subsequently initiating a condensed-phase propagating reaction. The maximum condensed-phase material temperature, Tmax, following a postulated burn can be estimated from Epstein (6).

Eq. 10

where s is the Boltzman constant, e is the emissivity, rg,o (kg/m3) is the initial gas density, Cv,g (J/kg-K) is the gas specific heat, Tg,o (K) is the initial flame temperature, as (m2/s) is the thermal diffusivity of the solid phase, V (m3) is the volume of the burn, A (m2) is the

corresponding surface area of the burn or the surrounding condensed-phase material and ks (w/m-K) is the conductivity of the condensed-phase material. Tabulated values of (Tmax - To) and Tg,o are provided in Table III based on e = 1, Cv,g = 1,000 J/kg-K, V/A = 1, as = 2 10-7 m2/s, Ks = 0.5 w/m-K and rg,o = 1.1 kg/m3.

Table III

Considering that transient burning (if at all credible) would be limited to temperatures of about 1000 K, this type of ignition source would not appear to be a threat to initiating condensed-phase propagating reactions.

The above observation is consistent with the considerable length of time the condensed-phase material must be exposed to a constant burning flame or fire prior to experiencing sustained combustion. For this case, the time to reach the ignition temperature can be estimated from

Eq. 11

where q (w/m2) is the flame heat flux and is usually taken to be of the order of 105 w/m2 (corresponding to the flame temperature of about 1100 K). As such we calculate the time to ignition to be of the order of 10 seconds. It is of interest to note that a burning wooden match develops about 100 W. When subjecting a stoichiometric mixture of Na3Citrate 2H2O and NaNO3 initially at an ambient temperature of 30C to a burning wooden match, a propagating reaction was noticed after about 10 seconds (7).

HOT OBJECTS

The necessary minimum temperature of a hot object to initiate a propagating reaction if brought in direct contact with condensed-phase combustible material can be estimated from the Contact Temperature Ignition (CTI) criterion (2)

Eq. 12

where TH (K) is the temperature of the hot object and s is given by

Eq. 13

where subscript o refers to the combustible condensed-phase material. In the case of a hot steel object, the value of s is about 8, requiring TH to be only about 10% higher than Tig, i.e., about 250C for combustible materials of interest (see Table I). Furthermore, recalling the critical ignition time of the order of 1 second, a minimum steel thickness (considering plane geometry)

Eq. 14

also needs to be satisfied. Setting a = 5 10-6 m2/s, we estimate the value of xmin of the order of 2 mm. It is therefore not surprising that a small hot steel ball (diameter of about 1.6 mm, and TH 1300C and energy content of 10 J) is capable of producing sustained combustion when brought into contact with combustible mixtures, such as those summarized in Table I (7). Such ignition sources, however, can be ruled out by limiting tank intrusive activities.

For completeness, we note that for such ignition sources to be effective requires essentially the absence of free water. Small quantities of water will prevent the contact temperature from reaching the ignition temperature (2).

LIGHTNING

The frequency of lightning striking a Hanford SST is estimated to be about 5 10-4 per year (8). However, as discussed below, ignition of condensed-phase waste materials, including organic solvents, from such a scenario is not considered a credible event. It is important to recognize that the Hanford tanks are buried and the extensive amount of rebar in

the SST concrete domes provide significant overall shielding properties, so that there is a high tendency for lightning produced current fields to be excluded from the interior of the tanks. However, there may be electrically discontinuous paths short that could result in arcing. These paths include arcing between equipment extending through risers, and the risers, arcing at bolted flanges, arcing between the riser or equipment and the rebar in the concrete dome. Lightning will, if the rebar is not well connected and connected to the tank walls in the SSTsc (which it is apparently not) arc to rebar and from rebar to the tank walls. As such, it would be difficult to rule out generation of sparks that would be capable of igniting the presence of a flammable gas mixture in the tank head space. However, considering the limiting durations such flammable gas mixtures could exist in connection with gas release events, these are considered negligible risk scenarios (8).

The above arcing phenomena are much more likely than arcing to the waste surface inside the tanks, as the various tank structures represent much better targets for the lightning current(s). If there is a gap in the conductors carrying a lightning current, the current may arc across the gap. Quoting Uman (9), "The arc energy appears as heated gas, in the tens of thousands of C range, and as heated and melted electrode material, much as in the case of purposeful arc welding. Typical lightning transfers 25 coulombs of charge, and thus an arc due to lightning between metal electrodes could liberate 250 joules of energy at the arc spot, in a volume certainly less than a cubic centimeter, perhaps as small a cubic millimeter." However, further quoting Uman (9). "It is not likely that the total lightning current would flow across one interior gap in a SST or DST because of the many parallel paths available to the lightning current. On the other hand, sparks in the millijoule range are generally thought to be capable of igniting flammable gas." It follows that significant steel melting in connection with such arcing phenomena that subsequently could fall down onto the waste surface would appear very remote. Approximately 250 joules alone would be required to produce a molten steel droplet of about 4 mm diameter, which is less than the capillary size of about 6 mm, i.e., the melted material would quickly refreeze in place.

Furthermore, we have evaluated the likelihood of arcing directly to the waste relative to the above metal arcing dissipation mechanisms for the lightning current energy using the similarity between the arcing caused by a lightning current and purposeful arc welding (9).

To conservatively test the effect or relationship between waste material electrical conductivity and arcing potential, small samples of simulated wastes were placed in a highly electrical conductive environment, i.e., in a small steel crucible (a pipe end cap of about 1.5 inch diameter and 1.5 inch height) clamped to a well-grounded welding table (see Fig. 1). The tungsten electrode was positioned above the waste surface, in direct contact with, as well as below the waste surface. Various types of wastes including organic solvent consisting of dodecane (1a), a stoichiometric mixture of NaCitrate 2H2O-NaNO3 soaked with dodecane (1b), dry stoichiometric mixture of NaCitrate 2H2O-NaNO3 (1c), and stoichiometric NaCitrate 2H2O-NaNO3 mixtures with increasing free water inventories of 5 and 10 wt% H2O (1d and 1e) were placed in the steel crucibles and subjected to arc welding currents of approximately 200 amperes for about one-second intervals.

Fig. 1

No sign of arcing were observed with waste mixtures a, b, and c, including variations in the electrode position relative to the waste material surface. With increasing water inventory, limiting arcing and "crackling" were observed, but in no cases was sustained ignition noticed. While the increasing presence of moisture or water clearly increases the electrical conductivity allowing current to flow, its presence also prevents ignition from occurring by dissipating the arc energy by latent heat of vaporization, keeping the waste temperature well below the ignition temperature. The above observations suggest that if lightning should strike an SST (a very unlikely event), the lightning current and its energy would be largely dissipated before entering the interior of the tank. Multiple paths, including arcing between equipment extending through risers, and the risers, arcing at bolted flanges, arcing between the riser or equipment and the rebar in the concrete dome, arcing from rebar to rebar and from rebar to the tank walls, are much better targets for lightning current energy dissipation than arcing deep into the interior of the tank, such as to the waste surface or below the waste surface. Furthermore, arcing to the waste, if at all likely, would appear to require the presence of moisture to improve its electrical properties, but this moisture would also prevent ignition. A lightning strike is, therefore, not considered a significant risk scenario in terms of presenting an ignition potential for organic solvents or solid wastes.

SUMMARY

Potential chemical safety hazards associated with condensed-phase propagating organic complexant-nitrite/nitrate reactions have been assessed for the Hanford Tank Farm leading to the following observations:

Condensed-phase spontaneous (no ignition source required) bulk runaway reactions are not possible, and

Credible ignition sources, including lightning, that could initiate a condensed-phase propagating reaction can be ruled out by assuring appropriate tank operating controls are in place and by limiting tank intrusive activities.

These observations can be made without requiring a detailed knowledge of the wastes including fuel content, water content, and heterogeneity, and will not be altered by waste aging or long term moisture loss.

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INVESTIGATION INTO THE CHEMICAL, THERMAL, AND RADIOLOGICAL CHANGES OF ORGANIC CHEMICALS ADDED TO THE UNDERGROUND STORAGE TANKS AT HANFORD

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ABSTRACT

The underground storage tanks at the Hanford site contain wastes generated from many years of plutonium production and recovery processes, and mixed wastes from radiological degradation processes. The chemical changes of the organic materials used in the extraction processes have a direct bearing on several specific safety issues, including hazards associated with fuel-nitrate combustion accidents. This paper details the second year's findings of a study charged with determining how thermal and radiological processes may change the composition of organic compounds disposed to the tanks. Our approach relies on literature precedent, experiments with simulated waste, and studies of model reactions. Our efforts have focused on the global reaction kinetics of a simulated waste exposed to g radiation.

Our experiments have subjected an "organic tanks" non-radioactive simulant to a range of temperatures and gamma radiation fluxes to determine the effects of radiation on aging of organic tank waste compounds. The simulant contained equi-molar amounts of dodecane, ethylenediaminetetraacetate (EDTA), tributyl phosphate (TBP), dibutyl phosphate (DBP), methyl isobutyl ketone (hexone), stearate, and citrate in an inorganic matrix containing hydroxide, nitrate, nitrite, aluminum hydroxide, and a variety of alkali, alkaline earth, and transition metal cations. We found that gas production is predominantly radiolytically induced. The main gases found are hydrogen, nitrous oxide, and nitrogen. Concurrent with gas generation we observe the disappearance of starting organic compounds and the appearance of condensed-phase products, dodecanones, heptadecane, isobutyrate, succinate, oxalate, formate, and glycolate. The apparent rate of "aging" is TBP>>hexone EDTA>stearate> citratedodecane. In the absence of radiolysis, TBP readily decomposes to DBP and butanol in the basic medium (1-3). However, radiation clearly accelerated consumption of the other compounds.

Our mechanistic studies, combined with literature precedent, suggest that oximes and possibly organic nitro compounds are key intermediates in the radiolytically-induced redox reactions of organic compounds with nitrate/nitrite. Hydrolysis, dehydration and autoxidation of these intermediates lead to ultimate products in which C-C and C-H bonds have been converted to C-O and C=O bonds. The finding of dodecanones in our simulant aging experiments, and homologous series of nitriles, aldehydes and ketones in vapor spaces above tank wastes containing NPH, are consistent with this scheme.

These results indicate that radiation promotes redox reactions between organic compounds (reducing agents) and nitrates/nitrites (oxidizing agents) in the wastes leading progressively to compounds with greater numbers of C-O bonds and fewer C-H and C-C bonds,resulting in an overall lower energy content for the organic inventory. Nitrate and nitrite are reduced to nitrogen compounds of lower oxidation state, primarily N2, N2O and NH3. Thus, to the extent that the tanks have been adequately ventilated and wastes continually dosed by radioactive elements, the total energy content of the tank wastes should have declined. However, aging processes appear to be converting high energy, nonpropagating organic compounds such as NPH, TBP and DBP to lower energy, propagating carboxylate salts. Thus, the level of current risk depends on how rapidly carboxylic salts of moderate energy content degrade to low energy oxalate and formate.

INTRODUCTION

Uranium and plutonium production at the Hanford site produced large quantities of radioactive by-products and contaminated process chemicals that are presently stored in underground tanks awaiting treatment and disposal. Having been made strongly alkaline and then subjected to successive water evaporation campaigns to increase storage capacity, the wastes now exist in the physical forms of salt cakes, metal oxide sludges, and saturated aqueous brine solutions. Those waste storage tanks containing organic process chemicals mixed with nitrate/nitrite salt wastes are thought to be at risk for fuel-nitrate combustion accidents. To assess the hazard and provide information needed to establish safety criteria, test programs must use simulants, at least until actual tank samples and hot cell test methods become available. Historical records can be used to identify the organics originally purchased and potentially present in the wastes, thus allowing experimental determination of the reactivity of mixtures of these materials with nitrates and nitrites as a first estimate of the hazard associated with the wastes. However, only postulated degradation products or analyses of individual tanks can be used to evaluate the current hazard associated with the organic wastes. To obtain more reliable knowledge of the degradation products, experimental studies are needed.

Objective

The purpose of the Waste Aging Task is to elucidate how chemical and radiological processes will have aged or degraded organic compounds stored in the tanks. This information supports efforts to evaluate the hazard as well as develop potential control and mitigation strategies.

Background

Each of the 177 waste tanks on the Hanford Site has a unique and largely unknown composition of organic, inorganic, and radioactive elements. A number of studies have been conducted to assess the inventory of chemicals added to the tanks as a result of chemical processes, such as

uranium recovery, reduction and oxidation (REDOX), plutonium uranium reduction and extraction (PUREX), and waste fractionation and encapsulation, that were performed at the Hanford Site (4-10). Some of these studies were summarized in PNL-8339, Assessment of Concentration Mechanisms for Organic Wastes in Underground Storage Tanks at Hanford (11), which assessed the concentration mechanisms for organic wastes that were believed to have been added to the tanks.

Many of the single-shell waste tanks have been sampled and total organic carbon (TOC) contents measured. However, knowledge of the TOC in a particular storage tank is insufficient to bound the safety risk without knowing the kinds of organic compounds that are in the waste.Because nitrate reaction energies vary widely for organic compounds, some information about the identity and oxidation state of the organic carbon is needed as well. Organic-containing wastes have been stored in Hanford Site underground storage tanks for tens of years. During that time the wastes have been exposed to radiation, temperatures of 20 to 140C (68 to 284F), and a reactive chemical environment having high concentrations of active chemicals, including hydroxide, nitrate, nitrite, aluminate, and transition metal oxides including noble metals, radioactive elements (uranium, plutonium, cesium, strontium, etc.) and many other materials that could act as catalysts and affect aging pathways.

The major organic compounds that were added to the tanks are divided into two classes: extractants/solvents and complexants for di-, tri-, and tetravalent cations. The major organic complexants that are believed to have been stored in the tanks are glycolate, citrate, hydroxyethylethylenediaminetriacetate (HEDTA), and ethylenediaminetetraacetate (EDTA). Allen (4) estimated the approximate quantities, which are listed here in kilograms and metric tons:

Glycolic acid, 8.8x105kg (880 metric tons)
Citric acid, 8.5x105kg (850 metric tons)
Hydroxyethylethylenediaminetriacetic acid, 8.3x105kg (830 metric tons)
Ethylenediaminetetraacetic acid, 2.2x105kg (220 metric tons).

In addition to these complexants, lesser amounts of complexants such as nitrilotriacetic (NTA), di(2-ethylhexyl)phosphoric, and oxalic acids were used, but amounts of these complexants that actually were stored in the tanks are not known.

Process solvents of concern that were used in Hanford Site chemical processes and stored in the tanks are tributyl phosphate (TBP), normal paraffin hydrocarbons (NPHs), and methyl isobutyl ketone. A recent examination by Sederburg and Reddick (12) of the PUREX plant material balances from 1955 to 1991 indicates that the quantities of TBP and NPH that went to tank storage are 7.22x105 and 1.31x106kg (720 and 1,310 metric tons). Other processes also used organic solvents and organic phosphate extractants, but less is known about the quantities that were added to the tanks. Di(2-ethylhexyl)phosphate diluted with hydrocarbon solvent was used in the waste fractionation and encapsulation process. The reflux solvent extraction process used TBP/carbon tetrachloride and dibutyl butyl phosphonate/carbon tetrachloride for extraction solvents. Considerable quantities of methyl isobutyl ketone (hexone) were used in the REDOX process as both extractant and solvent (11). For example, 6.5x104kg (65 metric tons) of hexone were retrieved from one storage tank, treated, and disposed. The evidence suggests that the quantities of organic solvents added to the tanks rival the quantities of complexants. However, the fraction of organic solvents that have escaped the wastes

via evaporation, or, in the case of phosphate esters, have been saponified in the alkaline wastes (1) is not known. Sederburg and Reddick (12) have pointed out that during early PUREX operations, the organic wash waste was combined with high-level wastes that generated enough thermal heat to cause the tank wastes to self-boil.

For complexants such as EDTA, HEDTA, glycolic acid, and citric acid, extensive studies of degradation mechanisms (aging) have been performed by Argonne National Laboratory (ANL) (13-16), Georgia Institute of Technology (Georgia Tech) (17) Pacific Northwest National Laboratory (PNNL) (18) and Westinghouse Hanford Company (WHC) (19,20). The purpose of these studies has been to determine the mechanisms by which organic wastes decompose and generate gases. Much of the work has focused on the role that chelators play in generating H2, N2, and N2O in tanks such as Tank 241-SY-101. The Georgia Tech group has focused primarily on the thermal non-radiolytic pathways that degrade the complexants, mainly HEDTA and glycolate, and produce H2 (21,22). The ANL and PNNL groups have explored direct and indirect radiation-induced pathways, as well as thermally activated pathways.

The simulants used in the above work did not include organic solvents, such as hexone, TBP, and NPH, that were widely used in the processing plants, mainly because the flammable gas-producing tanks of greatest concern did not receive significant quantities of these compounds. Analyses of core samples from Tank 241-SY-101 have shown that a significant portion of the organic carbon is contained in chelators, chelator fragments, and low molecular weight carboxylic acid (i.e., formic, acetic, glycolic, oxalic, succinic, and citric acids). The NPH components amounted to only 2 to 3% of the organic carbon (23). In contrast to Tank 241-SY-101, Tank 241-C-103 contains 15 to 21 kL of organic liquid composed primarily of TBP and NPH floating on the aqueous wastes (24,25). Considering the large quantities and high energy content of the organic solvents disposed to the tanks, how they have aged and become distributed in the tanks is very relevant to the organic tanks safety issue.

As a precursor to the experimental studies described in this report, a literature review was performed to assemble a bibliography (26) of literature relevant to understanding the chemical transformations that have occurred in the tanks (27). The bibliography focused on the hydrolytic, radiolytic, and free radical-mediated chemistry of normal paraffinic hydrocarbons (NPH), TBP, hexone, and other organic constituents. Aging mechanisms of ferri/ferrocyanides and the EDTA-type complexants were not addressed in the bibliography as they have been examined recently by Lilga (28), Meisel (13-16) and Ashby (22). Additional literatures review were done in FY94 (29) to address specific questions concerning aging pathways and products, i.e., alkaline hydrolysis of nitrate and phosphate esters, effects of oxygen and radical initiators on degradation of nitroalkanes, production of organic nitro compounds via reaction of nitrite with organic radicals, and radiation-induced decomposition pathways of phosphate esters.

Approach

Hazards posed by uncontrolled exothermic oxidation of organic compounds by nitrate and nitrite relate directly to the energy content and oxidation kinetics of the various organic compounds present in the waste. Until sampling and analysis of the tank wastes become routine procedures, a viable approach to assessing the current organic content of the tanks

is to simulate the chemical conditions of the tanks and elucidate mechanistic pathways that are key to knowing if the hazards have increased, decreased or remained constant with time. Accordingly, this project is using simulants in its studies of radiation-induced chemical aging effects. The work is proceeding in three sequential steps: 1) simulant selection and preparation, 2) scoping studies, and 3) long term aging studies. To facilitate studies of radiolytic-induced chemical aging, external radiation from a g source rather than use of radioactive chemicals are used. Radiation doses and/or temperatures are selected to produce aging reactions over times ranging from a few days to several months. After being irradiated, the disappearance and appearance of detectable organic products in the simulant and evolved gases are determined.

Although simulant aging studies will provide global pathways for aging, they will not allow confident elucidation of how these reactions proceed on the molecular scale because of the complexity of the simulants. Therefore, in concert with and in support of the simulant aging studies, less complex reaction systems that are more readily probed and understood are being examined. In addition, this part of the aging studies is addressing specific safety/treatment questions. As these questions arise, the literature is consulted for pertinent information, and supplemental experiments are designed and performed as needed.

EXPERIMENTAL SECTION

All of the chemicals used in this study were purchased from Aldrich Chemical Company, Eastman Kodak, or Baker Chemicals. The purity of the chemicals was reagent grade or better, as defined by the American Chemical Society. The chemicals used in this work were not further purified. A high-shear mixer (PoyltronTM PT6000, Brinkman Instruments Inc., Westbury, NY), was used to mix the simulant. The mixer speed was ~6000 rpm. Water used for preparing the simulant was deionized by a Milli-QTM Deionization System (Millipore Corporation, San Francisco, CA).

Simulant Preparation

The simulant used in this portion of the study is designated SY1-SIM-94C. Its composition is listed in Table I. The inorganic components of SY1-SIM-94C are largely based on analytical data for Tank 241-SY-101 waste (30). Sodium nitrate, nitrite, and hydroxide comprise the largest share of the mass of the inorganic and non-radioactive species (30). Also, significant amounts of aluminum are present, presumably, as sodium aluminate, not aluminum oxide, due to high levels of sodium hydroxide. Compared to SY1-SIM-93, five inorganic elements, Ce, Zr, Pd, Rh, and Ru, were added and levels of Cl and F were reduced. The levels of the noble metals Pd, Ru, Rh (as nitrates) were chosen based a report by Reynolds (30). Cerium (III) was added so as to include an element from the lanthanide series. The concentration was set at one third the level of Ca. Zirconium was added and the halides reduced based on information Scheele (31) obtained about the chemical compositions of Sr and Cs removal process waste streams.

The organic components were limited to seven compounds, each added on an equi-molar basis, to facilitate analyses of reactants and products in the condensed phase. Stearate, a long chain carboxylate anion, and dibutyl phosphate were included in the simulant to reflect partial oxidation of NPH and hydrolysis of TBP under plutonium extraction (PUREX) process conditions. Glycolate and HEDTA, present in SY1-SIM-93C, were excluded to

simplify analyses and because their radiolytic breakdown had been elucidated by Meisel (15,16).

The simulant was prepared by first dissolving the total amount of sodium hydroxide in 700 mL of water. Sodium aluminate was added slowly, with stirring. The remainder of the inorganic constituents were weighed out and added successively to the mixture with periodic mixing to homogenize the ingredients. The organic constituents, the first seven compounds in Table I, were weighed into a separate container, and then combined with the inorganic mixture. Next, the remainder of the water was used to wash the organic residue into the mixture. The entire simulant mixture was mixed for an additional 10 min. The resultant light-green solution had the appearance and consistency of a milk shake. The simulant was stored in a tightly capped polyethylene jar at ~4C.

Table I
Irradiation of an Organic Tank Waste Simulant

Until sampling and analysis of the tanks become a routine procedure, hazard assessment must rely on information obtained from studies with simulants. Other studies conducted at Hanford have made extensive use of simulants to mimic the pertinent chemistry of the flammable gas-producing wastes. Their simulants did not include organic solvents, such as hexone, TBP, and NPH, which were widely used in the processing plants. This paper will examine both complexant and organic solvent aging.

The basis of this paper was based on an experimental plan that called for an organic tanks simulant, performance of scoping experiments, and then a series of long-term aging experiments. The simulant composition was determined by consensus of PNNL and WHC investigators involved with tank safety programs and having knowledge of the chemical process streams that were fed to the Organic Tanks. Scoping experiments were performed to optimize procedures and establish the range of conditions over which to collect data. The simulant was dosed with 0.6 MGy at 90C in an O2/Ar atmosphere. In another experiment, the simulant was thermostatted at 90C for the same time (8 d) as the previous experiment, except it was not irradiated. Gas and condensed phase samples were retrieved from the reactor and analyzed. Based on these results, sampling intervals and temperatures were selected for the long-term aging studies. Experimental procedures and results are detailed in our year end report (32).

The schedule of experiments is provided in Table A-I of the Appendix. Most of the simulant irradiation experiments were run at 50, 70, and 90C with samples receiving radiation doses of 0.07 to 1.2 MGy (7 to 120 MRad) at a flux of 2660 to 3100 Gy/h (2.66x105 to 3.10x105 MRad/h). A few experiments were run at a much lower flux. Several duplicate experiments (runs 13-16, 18 and 21) were run to determine reproducibility. Most experiments (nos. 31, 32, and 34) had oxygen gas in the headspace of the reaction vessel at the start of the irradiations. However, three experiments were run without oxygen to learn how oxygen effects radiolytic and thermal reactions of the simulant. And, experiments (nos. 2, B1, B2, B3, and X1) were run in the absence of radiation to determine thermal conversion levels.

Sample irradiation experiments were performed within the g-Irradiation Facility at PNNL. The facility contains 37 stainless steel irradiation tubes positioned in a 2.13-m-diameter by 4.19-m-deep stainless steel tank. Two arrays of 60Co sources with a combined inventory of 1.184x1016 Bq are located near the bottom of the tank. For radiation shielding purposes, the tank is completely filled with water, and a concrete wall,

1.1 m in height, surrounds the top of the tank. The irradiation tubes, which are sealed on the bottom, vary in length from 4.9 to 5.5 m, and in diameter from 4.6 to 5.1 cm. The irradiation fluxes of the tubes range from 2 to 2x104 Gy/h (200 to 2x106 Mrad/h). The uniform flux region varies from 15.2 cm for the tubes closest to the sources to greater than 30.5 cm for the tubes farthest from the sources. All flux measurements of the tubes are traceable to the National Institute of Science and Technology (18).

Vessels of the simulant are manually lowered into the irradiation tubes to the desired flux where they are left in the tubes for a specific length of time to attain the required exposure. No nuclear activation products are associated with the g irradiation, thus the samples and sample vessels can be transported to other facilities for examination after being removed from the tubes.

Other ancillary equipment at the g-Irradiation Facility increases the repeatability of thermal control and sampling procedure. The gas manifold system is used to connect multiple reaction vessels to the initial headspace gas source, pressure monitoring equipment, and a port for headspace vapor removal. A gas sample is removed using an evacuated bulb sampler for analysis by mass spectrometry (MS) at another location. The data logger controls both the automated temperature and pressure measurements of the reaction vessels.

Reactor Vessel

The reactor vessel for this study was made from 1.59 cm inner-diameter 316 stainless steel pipe. One end of the pipe was sealed by welding a plate of 316 steel to the pipe. The other end was welded to a flange, 1.59 cm inner-diameter by 3.49 cm outer-diameter. The reactor cover has two openings. One opening was fitted with a 0.16 cm swagelok fitting (Swagelok Company, Solon, OH) to connect to the gas manifold via small-diameter stainless steel tubing. The other opening was fitted with a 0.32 cm swagelok fitting through which a K-thermocouple was inserted 15.2 cm into the body of the reactor from the top of the flange. The volume of the vessel to the top of the flange was approximately 30 mL.

The reactor vessel was heated to operating temperature by heating tape that had been wrapped around the outside of the vessel. The internal temperature of the reaction vessel was maintained by feedback control to minimize thermal gradient problems (18). Reaction vessel temperatures and pressures were monitored using a data logger.

Filling and Placing the Reactor

The simulant was removed from storage at ~4C, equilibrated with room temperature, and resuspended by mixing with the high-shear mixer. An aliquot of 15 mL was measured out and transferred to the reactor. The reactor was then sealed and transported to the g-Irradiation Facility. The staff at the g-Irradiation Facility performed the irradiation experiments according to a set procedure 32. A mixture of 79.5% argon and 20.5% oxygen was used as the gas phase, instead of air, to facilitate analyses of nitrogen gas. The vapor phase of the reaction was removed at the end of the irradiation time via an evacuated bulb sampler at temperature. The vapor was later analyzed by PNNL staff according to the procedure by Goheen (33). Total moles of gases produced were calculated based on the measured pressure, temperature, and known volume (24 mL) of the gas phase of the reaction vessel plus gas manifold lines. Concentrations (mol%) of individual components in the gas sample were determined using mass spectrometric analyses. The quantities of specific

gases generated were calculated from the total moles of gas generated and the mole fraction data.

No attempt was made to extract product gases that were either dissolved in the simulated waste mixtures or present as gas bubbles. For all product gases except ammonia, this approach should result in negligible errors. Ammonia has significant solubility in the concentrated simulant mixtures (34). Solubilities of gases such as H2, N2, and N2O are known to be quite low in concentrated brines as compared to their solubilities in pure water (35,34). Gas bubbles will contribute to the measured pressure in the test vessel, much as if the bubbles were brought to the slurry surface and eliminated, provided that simulant surface tension does not significantly compress the bubbles.

Condensed Phase Analyses

After the gas sample was removed, the reactor was cooled to below 35C, removed from the manifold, sealed, and transported to another laboratory to retrieve the condensed phase. The lid was removed and the mixture was stirred with a spatula and separated into two samples of ~7.5 mL. One sample was combined with other washings from the reactor to a total volume of ~20 mL and stored at -2C. The other sample was analyzed for the organic constituents by PNNL's Advanced Analytical Methods Development Group.

Continual modifications and improvements to the analyses of the simulant waste samples have been made. Three different analyses are being used to date to obtain quantitative information on the organic starting components. In addition, some of the degradation products from heat and gamma irradiation treatment have been identified.

Gas Chromatography/Mass Spectrometry (GC/MS) is used to quantify dodecane, stearic acid, hexone, and TBP. This procedure requires neutralization of the caustic sample with phosphoric acid followed by extraction of these four organics with dichloromethane (CH2Cl2). Two surrogates, dodecane-d26 and palmitic acid, are added prior to extraction to track the efficiency of the sample preparation procedure. Following extraction, diazomethane is added to methylate stearic acid to enhance its volatility for GC/MS analysis. Analyte recoveries for the CH2Cl2 extraction and sample preparation procedure were determined by spiking known amounts of the organic analytes of interest into the inorganic portion of the simulant recipe. Triplicate experiments gave recoveries of 102% for dodecane, 96% for TBP, and 98% for stearic acid.

Recoveries for dibutyl phosphate with this sample preparation method averaged only 6%. Therefore, this is not a viable method for this analyte. Alternatives for analyzing DBP are being pursued under the Advanced Organic Analysis task. Precision of the GC/MS analysis, determined through replicate sample injections had a standard deviation of less than 2% for all analytes.

Ion-Pair Chromatography (IPC) is the method of choice for the analysis of EDTA. The aqueous layer remaining after extraction of the organics by CH2Cl2 for GC/MS analysis is used for EDTA analysis. An aliquot of the aqueous layer is diluted to a concentration within the dynamic range of the IPC technique. Copper sulfate is added to enhance EDTA detection.

Error estimates for sampling and analysis were performed under the Flammable Gas Safety Organic Analysis Task, but can be applied here as well. The simulant waste samples have similar inorganic matrix recipes. Replicate sample preparations and instrument injections produced standard deviations of 5% for control simulated waste samples and up to 10% for

heated and irradiated simulant waste samples. The increased error in the stressed samples can be explained by possible incomplete separation of degradation products in the stressed samples.

Ion chromatography (IC) with conductivity detection is used for citric acid analysis. This technique was developed under the Flammable Gas Safety Project for detection of low molecular weight organic acids in waste tank matrices. The simulant waste sample is diluted with water. Dichloromethane is added to remove non-polar organics that could interfere with the IC. Recovery studies of citric acid spiked into the simulant inorganic matrix gave a recovery of 97%. Triplicate weighing and sample preparation gave sampling error (standard deviation) of 9%. Multiple injections of the same sample gave a precision of 1%. Therefore, the total error for the IC analysis was determined to be 10%.

RESULTS OF WASTE AGING STUDIES

Gas Phase Analytical Results

Results of headspace gas analyses have been completed. The analytical data are contained in the Appendix (Tables A-II and A-III) of this paper. The quantities of gases found were calculated from reactor head pressures and mass spectral analyses of headspace gases drawn from the reactor. Figure 1 shows a plot of the reactor headspace gas pressures as function of radiation dose and temperature. The pressures in the figure have been corrected for the pressure increases due to raising the reactor temperature from ambient to the actual run temperatures. Control experiments B1, B2 and B3 that were run in the absence of radiation showed little gas increase after correcting for temperature changes. Accordingly, the pressure increases are due to radiolytically-induced gas-producing reactions.

Fig. 1

Induction periods for gas production were observed. For example at 50C, the rate of gas production increases significantly after the simulant has received a dose of about 0.2 MGy (20 Mrad). Gas pressures increase linearly with dose after the induction period indicating that gas production obeys zero-order kinetics, p=kt. Induction periods decreased with increasing temperature.

Hydrogen, nitrogen and nitrous oxide are the major gaseous products produced during g irradiation of the simulant. The yield of individual gases showed behavior similar to that for total gas produced except scatter in the data is greater. Figures 1 and 2 show plots of the production of these gases at 50, 70 and 90C. Yields in Table A-III and Figs. 1 and 2 are in mmol. For reference, the headspace above the simulant initially contained 752 mmol of argon and 192 mmol of oxygen in a volume of 24 mL. Linear fits to the yield data in Figs. 1, 2, and 3 provide zero-order rate constants for production of H_2, N_2 and N_2O gases. Figure 4 plots these rate constants vs. inverse temperature in K.

Fig. 2

Fig. 3

Fig. 4

As shown in Figs. 2 and 4, production of H_2 gas increases markedly when going from 70 to 90C. Scatter in the data is also much greater at 90C. This behavior is attributed to the onset of thermal pathways for generating hydrogen from organic degradation products. Meisel et al. (14-16) found that preirradiation of simulated wastes containing glycolic, EDTA, HEDTA, and citric acids showed enhanced production of H_2 during thermal treatment. Meisel et al. (14-16) suggested that radiolytic

degradation of chelators produced formaldehyde and glyoxylate, which decompose thermally producing H2.
Figure 3 shows that, after an induction period, production of nitrogen-containing gases obeys a zero-order kinetic rate law, m=kt, where m is moles of a gas in the headspace. N2O is produced in greater amounts than N2. The ratios of N2/N2O yields approach ~0.5 at higher does. The ratios decrease with increasing temperatures consistent with a larger Arrhenius activation energy (Fig. 4) for N2O production compared to N2 production. The activation energies are 73 kJ/mol for N2O and 13 kJ/mol for N2. The smallness of these barriers are in accord with the radiolytic mechanism offered by Meisel (16) in which both N2 and N2O gases derive from a common radiolytically-generated intermediate (Eqs. 1-5).
Eq. 1
Eq. 2
Eq. 3
Eq. 4
Eq. 5
According to this mechanism, the combined yields of N2 and N2O are dependent on the production of oximes; whereas, the relative amounts of N2 and N2O depend on how hyponitrous acid (HO-N=N-OH) partitions. Meisel 16) proposed that the oximes originate from combination of radiolytically-generated NO and organic radicals.
Several control experiments were run to examine the role that organic compounds in the simulant might play in promoting the generation of gases. Table II compares results for these runs with results for SY1-SIM-94C runs with and without irradiation. Experiments B4 to B6 were run with the inorganic portion of the simulant which contained only a very small amount of citrate that had been added as zirconium citrate. Its concentration was only 1/70th the molar amount of the total organic constituents in the SIM-SY-94C simulant. Levels of nitrogen containing gases in runs B4 to B6 were four to ten times smaller compared to the SIM-SY-94C runs with irradiation. Hydrogen gas yields were similarly reduced. Runs B1 to B3 with SIM-SY-94C and no radiation produced very little gas. These results clearly show a synergism between radiation and organic compounds in promoting gas production.
Table II
Oxygen gas in the headspace above the simulant was consumed, both in the presence and absence of radiation (compare runs 11 to 13 with B1 to B3 in Table II). Figure 5 plots headspace oxygen levels in mmols as function of dose and temperature. As evidenced by the plot, oxygen levels in the headspace decreased to a non-zero steady-state during irradiation. The rate at which the steady-state was achieved increased markedly with temperature. Furthermore, the steady-state level increased with temperature from 5% at 50C to 30% at 70C, and to 40% at 90C. Fitted rate constants and equilibrium O2 levels are 1.1 MGy-1 (0.011 MRad-1) and 9.8 mmol at 50C, 3.2 MGy-1 (0.032 MRad-1) and 57.9 mmol at 70C, and 18.8 MGy-1 (0.188 MRad-1) and 77.1 mmol at 90C. These equilibrium levels correspond to 5%, 30%, and 40% of the starting amount of O2 in the reactor headspace (~15 ml).
Fig. 5
In the case of a simple reversible reaction, AB, the steady-state level corresponds to the level at which the rates of consumption and rates of production are matched. Runs 31, 32, and 33 (Table II) starting without oxygen gas in the headspace were run to test if steady-state O2 levels

could be reached by starting without O2. Oxygen was produced in these runs, although the rates of production were much slower than the rates of consumption from runs starting with 21% oxygen in the headspace. Perhaps it should not be surprising that the chemical kinetics are much more complex than that for a simple reversible reaction, given the complexity of the simulant and of radiolytic processes in general.

Additional evidence that the oxygen steady-state levels are due to radiolysis was obtained from experiments run in the absence of radiation. The concentration of oxygen decreased well below the steady-state concentration reached during irradiation. The degree of consumption for a given interval of time is dependent upon the temperature of the reaction as can be seen in experiments B1 to B3 in which the only variable was the temperature. Approximately 84% of the initial oxygen was consumed at 90C in 11 days. This length of time is equivalent to the irradiation time for sample to receive a dose of 0.80 MGy (80 MRad) in the g facility.

Small quantities of "other gases" (see Table A-II in Appendix) were detected in the mass spectrometric analyses, but the ions could not be unambiguously assigned. To learn the identities of these gases, headspace gases from run 4 were analyzed using GC/MS according to the procedure by Stromatt (37). The gases were found to be a mixture of butanol and hexone. Butanol was present at 0.2 g/L, and hexone at 4.4 g/L, in the vapor phase. Butene and acetone, the possible products of TBP and hexone decomposition, which have been identified in the vapor of Tank 241-C-103 (24), were evidently below detection limits.

Condensed Phase Analytical Results

Organic analyses for most of the irradiated samples have been performed. Table A-IV contains the analytical data. Only the data for EDTA and citrate lend themselves to kinetic analysis. These data will be analyzed below. Data for the disappearance of hexone, dodecane and stearate are inconsistent and scattered. Data plots for these compounds appear in the FY95 year end task report (32). Control experiments indicate that excellent recoveries of these compounds can be obtained from the unirradiated simulant using the current analysis methods. These compounds are not expected to be soluble in the aqueous phase of the simulant. Therefore, sampling errors caused by sample inhomogeneity may account for the scatter.

Destruction of TBP was near complete at all radiation times. In fact, we observed its to be disappearing even while stored at 4C. The simulant was made up to contain approximately 16 mg/g TBP. Initial analyses found 18.6 mg/g. But, subsequent analyses at 3 weeks, 11 months, and 16 months found respectively 8.9 mg/g, 6.4 mg/g and 4.9 mg/g of TBP in the simulant. The rate of disappearance in the simulant is much faster than would predicted based on literature rates (1) for hydrolysis of TBP in 1 M NaOH. Similarly, results in Table A-IV show it to be disappearing at least 10 times faster than expected when the samples were held in the gamma facility at 50 and 70C even when the samples received no radiation dose. The reason for this surprising result is not known. It may be due to greater interfacial surface area of the emulsified simulant and a catalytic effect of metal oxides, e.g. Zirconia (38,39), or other multivalent cations. Emulsification of the simulant has lead to a significant increase in the effective concentration of TPB and Dodecane solubility and rate of hydrolysis. These two solvents have minimal solubility (10-3 M) in aqueous solutions (1). Work is currently under way to quantify the increased rate of hydrolysis due to cation and mixing.

Recoveries of DBP were only 6% from the unirradiated simulant. Therefore, quantification of DBP in irradiated samples was not attempted. The radiolytic and thermal reactions of DBP are being investigated in a separate study using a less complex simulant and 31P nuclear magnetic resonance detection.

The disappearances of EDTA and citrate in the condensed phase fit the first-order kinetic rate law, A=Aoe-kt. Plots of the data are furnished in Figs. 6 and 7. The radiation doses necessary to reduce the concentration of EDTA in the simulant by one-half were 0.63 MGy (63 Mrad) at 50C, 0.51 MGy (51 MRad) at 70C, and 0.41 MGy (41 Mrad) at 90C. Citrate was less reactive requiring about 5-times more radiation to reduce its concentration by one-half. The greater reactivity of EDTA relative to citrate is consistent with the greater number of C-H bonds in EDTA compared to citrate. Statistically, we would expect an EDTA:citrate reactivity of 3:1. Also, 8 of 12 C-H bonds in EDTA are adjacent to both carboxylate and amine functionalities. This should enhance their reactivity relative to C-H bonds adjacent to carboxyl groups in citrate. Scheme 1 shows a phenomenological model that is consistent with radiolytically-induced consumption of organic substrates in the simulant.

Scheme 1

The scheme assumes that radiolysis intermediates HO, O-, NO3, and H (13,14) attack the organic compounds producing organic free radicals which are subsequently converted to products. According to this scheme, the reactivity of an individual substrate depends on the rates at which it is attacked relative to the rates at which other compounds including inorganic compounds such as NO2- are attacked. Therefore, the rate of disappearance of a substrate depends on its concentration, which leads to the observed pseudo first-order kinetic behavior. Nitrite ion exhibits high reactivity towards e-, H, O-, and NO3: 4x109, 7x108, 3x108, 1x109 M-1s-1, respectively. And, it is present in high concentrations (3.65 M). Therefore, it probably is the major scavenger of attacking intermediates. Figure 8 plots the disappearance rate constants vs. inverse T (C) for EDTA and citrate. Least-squares fits of the standard error-weighted rate constants yielded activation energies of 9.72.5 kj/mol for EDTA and 18.98.7 kj/mol. The smallness of these activation energies is consistent with attack on these compounds by radiolyitically-generated nonselective reactive intermediates such as H, O-, and NO3.

Although data for dodecane, stearate and hexone do not reduce as well as the data for citrate and EDTA, qualitative comparisons of reactivity are possible. For stearate, the 90C data reasonably fit the first-order kinetic model. The fitted rate constant is 0.60.1 MGy-1 (61kRad-1). This value is ~40% of the EDTA rate constant and about twice the citrate rate constant at 90C. Forceda fits to dodecane data yield rate constants that are comparable to citrate. And, similar fits to hexone data yield rate constants that are larger that those for EDTA. Thus qualitatively, the relative reactivities are hexoneEDTA>stearate>citratedodecane.

The first-order rate equation was assumed and the fit was forced through the starting concentration of substrate.

Fig. 6

Fig. 7

Fig. 8

Several solution-phase products have been identified: dodecanones, heptadecane, isobutyrate, succinate, oxalate, formate, and glycolate. Table III lists amounts of glycolate, formate, oxalate and succinate by

IC analyses of the irradiated simulant. We suspect that docecanones are from dodecane, heptadecane is from stearic acid, isobutyrate is from hexone, and oxalate, formate, and glycolate are from EDTA and citrate. Significant quantities of succinate are found in runs B1, 27, 28 that received little or no radiation. Of the organic components in the simulant, TBP was consumed consistently and totally during these runs. Tributylphosphate rapidly hydrolyzes under the reaction conditions producing butanol and DBP. Thus, butanol is a plausible source of succinate.

Table III

Generally, our studies use high radiation fluxes to simulate waste aging over times of 1 to 14 days that are experimentally and programmatically practical. A few experiments have been run and are in progress to assess the effect of flux times that are short compared to the age of tank wastes. In these experiments, similar doses were delivered over times ranging from 90 to 104 days. Only two experiments (29 and 6) run at 90C are directly comparable. The simulant received a dose of ~0.6 MGy (~60 MRad) in each experiment although the flux differed by a factor of 12. Similar levels of EDTA, the kinetically best behaved of the simulant organic compounds, was observed for high vs. low flux: 5.8 vs. 7.5 mg/g. Stearate and hexone showed similar levels in both runs, but citrate was smaller in the low flux run (8.4 vs. 3.4 mg/g) which would be consistent with a thermal contribution (Ashby (15) to its consumption at longer times. The gas data correspond, as well. The combined yields of N2 and N2O are just 24% greater for the low flux run, while H2 runs are 135%. Greater thermal contributions to H2 yields are expected (15,22). These meager results provide tentative evidence that the radiolytic consumption of organics compounds scale linearly with flux. If attack on reactants or generation of a product requires reaction of two radiolytically-generated radicals in a single step (for example, Eq. 1), then quadratic scaling is expected. If additional data confirm that N2 and N2O yields do not scale quadratically then Eq. 1 may not be key in producing of nitrogen-containing gases. The result would be more consistent with the key step being reaction of organic radicals with nitrite. In summary, the results show that thermally- and hydrolytically-stable waste components, EDTA, citrate, hexone, and NPH undergo radiation-induced degradation. Based on the observed gaseous and condensed-phase products and supporting literature information (15,26,22,29), the organic compounds in the tank wastes are degrading by oxidative processes. Degradation rates depend mainly on the strength (flux) of the radiation fields to which tank wastes have been exposed and, to a lesser extent, on tank waste temperatures. If average radiation fluxes and temperatures for various tank wastes can be estimated, then ballpark estimates of the lifetimes (i.e., 1, 10, or 100 years) of organic tank wastes can be made.

CONCLUSION

Many of the organic compounds thought to be in tank wastes have been tested to determine which and what concentrations of the compounds may react with nitrate/nitrite salts in propagating reactions (37). Surprisingly, many of the energetic organic compounds, such as NPH, TBP and DBP, did not sustain propagating reactions because they either volatilized away from the heated initiation zone or decomposed to volatile compounds and gases that escaped before propagation temperatures could be reached. The compounds at most risk for propagating reactions were organic carboxylate salts and complexants. However, oxalate and

formate have too little energy content to propagate at TOC levels found in the tank wastes.

In summary, the total energy content of tank wastes should be declining with time. However, aging processes appear to be converting high energy, nonpropagating organic compounds such as NPH, TBP and DBP to lower energy, propagating carboxylate salts. Thus, the level of risk depends on how rapidly carboxylate salts of moderate energy content degrade to low energy oxalate and formate, and in turn how fast these degrade to CO2/CO32-. Work is in progress that will address this and related subjects.

Under the TWRS program plan, we plan to examine the reactivities of butanol (a possible source of succinate and butyrate), of organic carboxylates and chelator fragments, and of DBP in tank waste simulates, as well as addresses other questions relating to waste aging. To date, much of our work has been with a SY-101-type simulant. Future experiments are planned to examine aging in salt cake simulants and the effects of moisture content on products and rates. We plan to correlate our simulant aging results with currently available and future tank characterization data. We will examine the technical feasibility of making estimates of organic compound lifetimes in the tank wastes in an attempt to predict the degree of waste aging based on historical information (10,9). To support this effort, waste contents and radiation fluxes of individual tanks need to be estimated from Hanford Tank Content Estimates.

Additional study of simulant aging is needed to determine kinetic behavior of intermediate aging products such as chelator fragments and low molecular weight acids. Study is also needed of how radiolytically-induced aging scales with radiation flux. For cases such as Tank 241-SY-101, in which organic analyses of tank waste core samples exist, we plan to learn the actual extent of aging that has occurred. We expect that a coherent picture of aging will emerge from the planned work, which will include integrating the results of Tank Safety Waste Aging Studies at a programmatic level.

ACKNOWLEDGEMENT

Pacific Northwest National Laboratory is operated for the U.S. Department of Energy by Battelle Memorial Institute under contract DE-AC06-76RLO 1830. We thank Dr. H. Babad (Westinghouse Hanford Company) for many helpful discussions.

APPENDIX

Table A1

Table A2

Table A3

Table A4

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Session 29 -- WIPP-2
Co-chairs: Warren Bodily, WEC
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29-1
SUMMARY OF SCIENTIFIC INVESTIGATIONS FOR THE WASTE ISOLATION PILOT PLANT*
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ABSTRACT
The scientific issues concerning disposal of radioactive wastes in salt formations have received 40 years of attention since the National Academy of Sciences (NAS) first addressed this issue in the mid-50s. For the last 21 years, Sandia National Laboratories (SNL) have directed site specific studies for the Waste Isolation Pilot Plant (WIPP). This paper will focus primarily on the WIPP scientific studies now in their concluding stages, the major scientific controversies regarding the site, and some of the surprises encountered during the course of these scientific investigations.
The WIPP project's present understanding of the scientific processes involved continues to support the site as a satisfactory, safe location for the disposal of defense-related transuranic waste and one which will be shown to be in compliance with Environmental Protection Agency (EPA) standards. Compliance will be evaluated by incorporating data from these experiments into Performance Assessment (PA) models developed to describe the physical and chemical processes that could occur at the WIPP during the next 10,000 years under a variety of scenarios.
The resulting compliance document is scheduled to be presented to the EPA in October 1996 and all relevant information from scientific studies will be included in this application and the supporting analyses. Studies supporting this compliance application conclude the major period of scientific investigation for the WIPP. Further studies will be of a "confirmatory" and monitoring nature.
INTRODUCTION
In 1974 the Atomic Energy Commission (AEC) began exploratory investigations for a nuclear waste repository in the salt beds in the southeast corner of New Mexico. Since January 1975 SNL has pursued scientific investigations with the ultimate goal of providing the knowledge base necessary to establish confidence in the ability of the site to safely isolate Defense Transuranic Waste and conform with all applicable waste disposal standards. Some early in-situ tests also evaluated disposal of Defense High Level Waste (DHLW) in salt beds because salt beds in Texas were one of three disposal options. Those

studies were terminated when the Congress focused High Level Waste Repository investigations on the Yucca Mountain tuff in Nevada.

During these investigations SNL has been assisted by the U.S. Geological Survey (USGS) and by private contractors, university researchers, and consultants too numerous to individually recognize. It is important to acknowledge the contributions to the WIPP scientific studies provided by the NAS and by the New Mexico Environmental Evaluation Group (EEG). Both organizations have provided valuable scientific oversight and critical review of the scientific studies since 1978. Many of the studies and approaches to the issues were generated by their thoughtful examination of the technical concerns.

Thanks to the efforts of these organizations and many dedicated SNL WIPP staff, the information developed over the past 21 years allows an informed and confident assessment of WIPP's ability to safely isolate transuranic waste from the biosphere and to comply with EPA repository standards.

The concept of geologic isolation for disposal of radioactive wastes, and in particular, utilization of salt formations, received a primal endorsement from the NAS with the publication of their report in 1957 (1). That recommendation stimulated laboratory and field studies by and for Oak Ridge National Laboratory (ORNL) culminating in the Project Salt Vault experiments (1963 to 1967) in bedded salt at Lyons, Kansas. These tests utilized irradiated nuclear fuel elements, supplemented by electric heaters, to study heat and radiation effects from large-scale experiments in salt. These tests established that salt was an acceptable host rock for radioactive waste disposal but local factors, both technical and potential, resulted in abandonment in 1972 of an AEC proposal to construct a repository at the Lyons, Kansas site.

Subsequently, ORNL and the USGS evaluated existing knowledge regarding salt deposits in the United States and selected the northern portion of the Delaware Basin in New Mexico as having the best prospects of meeting the Site Selection Criteria. By June 1974, two boreholes had explored two corners of the proposed site area when site characterization was suspended in favor of the Retrievable Surface Storage Facility (RSSF) concept. The RSSF was soon abandoned because of objections from intervenor groups and the predecessor to EPA during the environmental hearing process. The concern was that problems created by this generation were being postponed to future generations.

When the Delaware Basin site characterization was resumed in early 1975, the AEC asked SNL to assume responsibility for managing continued site characterization, facility conceptual design,National Environmental Policy Act (NEPA) studies and scientific studies related to waste disposal in salt beds.

The remainder of the paper will summarize the major geotechnical issues and scientific studies that have occurred within the WIPP program over the past 21 years. The discussion will be divided into the major categories of 1) site selection and characterization, 2) facility seals and rock mechanics, 3) fluid flow in the geologic system, and 4) waste room interactions, (including gas generation). This paper will not address the equally interesting history of the political and policy shifts over this period of time even though they often impacted the scientific program.

SITE SELECTION AND CHARACTERIZATION

SNL formally commenced WIPP studies in January 1975 and received funding in March 1975 to begin exploratory drilling activities and NEPA-related field studies. The first task was to "complete" site characterization by exploratory drilling of the remaining two corners of the 11/2x2 mile site inherited from the earlier program. The first of these proposed boreholes, ERDA6, was drilled between June 13,1975 and September 23,1975. This exploratory boring unexpectedly encountered steeply dipping and displaced strata in the Salado and Castile formations and a geopressured brine reservoir in the uppermost fractured Castile anhydrite. This unexpected geologic structure and the inability to predict acceptable repository conditions throughout the site area led to an early disqualification of that site. Site selection investigations then focused on finding a more acceptable site within the Delaware Basin (Fig. 1).
Fig. 1
Intensive evaluation of existing geologic and geophysical information that had been acquired by the potash and petroleum industries was initiated. This examination was allowed by these companies on a "company proprietary" basis. These studies established that the evaporite deformation zone encountered in ERDA6 was primarily confined to a belt about five miles wide paralleling the buried Capitan Reef. Drilling data indicated that away from this zone, toward the interior of the Delaware Basin, the Salado Formation was relatively undeformed. The renewed site selection activities, continuing to focus on the Northern Delaware Basin but using more restrictive site selection criteria generated from the evaluation (and a relaxation of borehole stand-off to one mile allowed by improved knowledge of the hydrology), resulted in the same prime alternative site being independently identified by Sandia and USGS teams in December 1975.
The center of this proposed site, about 5 miles southwest of the first location, was core drilled (ERDA9) by June 1976 and the analyses confirmed the desired bedded salt properties, the anticipated stratigraphy and relatively flat dip. This initial tentative site identification led to a broadly-based geotechnical investigation focused on this region. Between 1975 and 1980, 57 boreholes were drilled and cored to provide basic geologic and hydrologic data, potash resource information, and to aid in interpretation of geophysical surveys. These latter surveys consisted of gravity, aeromagnetics, resistivity, and seismic reflection and refraction surveys. Seismicity evaluations established that the area was basically aseismic and concluded the majority of seismicity in the Central Basic Platform area of west Texas was induced by water-flooding activity. The nearest "active" fault, i.e. exhibiting evidence of motion in the last 100,000 years, was determined to be 65 miles to the west on the western slope of the Guadalupe Mountains. Surface mapping and drilling examined features suspected of being breccia pipes, and determined that breccia pipes occurred only over the Capitan Reef and were not, therefore, to be expected at the WIPP site.
Resource evaluations for potash and oil/gas were conducted based on WIPP drill holes and seismic data coupled with knowledge of oil/gas field development and potash exploration. No other significant resources were identified. With this information the site was located to avoid major known petroleum trends and minimized conflict with potash resources to the extent possible. Complete avoidance of potential conflict with

hydrocarbons and some potash ore was known to be unattainable in this portion of the Delaware Basin.

Studies conducted by the USGS concluded that salt dissolution rates would not breach the WIPP for millions of years. Surface geology, hydrologic studies, and selected drilling and geophysical surveys ruled out point dissolution and the existence of karst geohydrology at the site.

All this geotechnical information was summarized in a site characterization report in 1978 (2). A generalized geologic cross-section is shown in Fig. 2. These characterization studies were conducted prior to the promulgation of EPA standards for nuclear waste repositories. Consequently, the site characterization criteria which governed the studies were based upon the breach scenarios, especially involving natural processes, which were generally regarded by the scientists in the national and international repository programs as being significant to repository integrity and necessary to predict the behavior of a nuclear waste repository in salt. WIPP's self-imposed criterion for the time duration of repository integrity was 250,000 years, approximately 10 half-lives of Pu-239.

Fig. 2

Certain site characterization issues became the focus of much oversight scrutiny during the '70s and early '80s. Chief among these were the salt dissolution aspects of which there were four. Regional dissolution of salt, both at the top of the salt section and within the salt beds themselves was one of the first issues addressed. The former was addressed by geologic studies conducted by the USGS determining both horizontal and vertical rates of dissolutioning over the past million years. This information established that the WIPP would not be threatened by this process for a far longer time than the 250,000 years then used as a criterion. The inter-bed locus for salt dissolution, particularly within the Castile and at the base of the Salado, had been suggested as a cause of some of the local lithologic and structural features in the Castile salt beds and as the cause of Castile salt bed thickness variations. This issue was directly addressed for suspect areas near the WIPP site by borehole drilling and core examination. This evaluation showed no evidence of dissolution features and, together with regional isopach and structural interpretations, led to the present concept of anhydrite foundering in the salt to explain the Castile variations in salt thickness. The third dissolution issue addressed "point source" dissolution as evidenced by features termed "collapse breccia pipes". These breccia pipes were cylindrical collapse structures, about 1000 feet in diameter, caused by dissolution of soluble rock at depth, ultimately resulting in collapse of overlying rock into the solution cavity with the chimney action progressing to the surface. The concern for the repository was that since these features were known to occur north of the site, they might also be developed at the WIPP, providing a permeable waterflow path through the repository horizon and leading to a breach of repository integrity. Studies by the USGS, based on drilling of known breccia pipes and other suspect features, surface mapping and one underground mining intercept, concluded that breccia pipes were not a threat to WIPP because: 1) they occur only over the Capitan Reef--a source of water for dissolution, and 2) the examination of existing breccia pipes revealed they were not a long-term permeable path for water flow.

The fourth dissolution issue dealt with the issue of karst hydrogeology. Karst was known to be well developed to the west of WIPP, in the Nash

Draw topographic depression. This karst expression was postulated by some critics to extend to the WIPP site with the associated consequence of extremely rapid hydrologic transport of any radioactivity release to the Pecos River. Subsequent geologic studies, geophysical surveys and expanded hydrologic testing failed to find any evidence of karst development over the WIPP site or along the indicated hydrologic flow path. Peer reviews of the extensive hydrogeologic database, conducted by the EEG and the NAS WIPP Panel, concluded that karst development was not a threat to the WIPP.

The final stage of the site characterization phase commenced with the Site and Preliminary Design Validation (SPDV) in July 1981. This underground excavation explored the core of the site area and enabled extensive geologic mapping for one mile in the north-south and east-west directions as well as mapping of the exploratory and exhaust air shafts. As a result of this geologic investigation and the accompanying mineralogical analyses, it was concluded that the site met the previously established siting criteria (3). DOE accepted the site and proceeded to full facility construction in July 1983.

FACILITY SEALS AND ROCK MECHANICS

A primary attribute of a salt repository is the ability of the salt to deform or creep over time thereby sealing man-made access and intrusions into the repository and preventing natural processes from causing permeable fractures or faults which could allow entry of water. This creep attribute is utilized in the sealing of repository access shafts and in predicting closure of the waste rooms and resultant encapsulation of the waste placed therein. To allow quantification of this behavior an extensive rock mechanics program has been carried out in the laboratory and in-situ at the WIPP site. Additionally, evaluation of numerous materials, both cementitious and native materials (salt and/or clay), which are candidate seal materials, has been completed.

Prior to gaining access to the WIPP underground, laboratory tests were conducted on rock salt core obtained from surface drilling. The constitutive data thus acquired were used in existing rock mechanics models to predict salt creep in the WIPP. The SNL models SANCHO and JAC were then benchmarked (4) in a program that compared nine different codes from throughout the National and International communities. This exercise revealed that similar results could be obtained from the more sophisticated codes but only if great care was given to assuring the same problem specification. Code operator set-up of the problem could significantly alter the results.

Prior to implementing in-situ tests, an in-situ test plan (5) was prepared which outlined the tests to be conducted in the WIPP underground. The rock mechanics tests in the WIPP ranged in size from single borehole to full simulation of disposal room, configured respectively for TRU and DHLW disposal. Defense High Level Waste experiments in salt were conducted to provide early information about issues that could face a proposed High Level Waste repository in a west Texas salt site.

Measurements were begun in 1983 and the final tests have only recently been terminated. These tests, which span a decade in some instances, relied primarily upon strain and displacement measurements although some tests required extensive thermocouple data. Some stress gauges were emplaced but difficulty in interpreting their results because of the "plastic" nature of the rock limited their utility. At the peak of the

in-situ test program, about 5000 channels of data, most of it for rock mechanics, were being continuously recorded. Figure 3 displays the location of the WIPP in-situ tests.
Fig. 3
The early in-situ observations of salt creep revealed that strain rates and room closure were proceeding at about three times the rate that had been predicted by pre-test modeling. This accuracy was not sufficient for sealing and room-closure prediction requirements. Several laboratory and modeling studies were implemented to establish what modifications were required to our predictive model and to our understanding of the creep process to better predict the observed behavior.
To be certain that errors were not introduced into the two-dimensional model by the geometric abstractions necessary to represent the 3-D world, a large (108-foot diameter) circular test room (Room H) with a cylinder of undisturbed salt 36 feet in diameter at the center was constructed. After acquiring data at ambient temperature for about one year (1985), strip heaters around the pillar were activated to raise the pillar temperature from 50 to 70C, near the center and at the pillar wall respectively. This greatly accelerated the creep rate so that greater strains could be accumulated during the measurement span than would otherwise be possible but assured the strain mechanisms would not be changed. Laboratory tests on salt core containing impurities and from the anhydrite marker beds permitted the model to incorporate a more detailed description of the stratigraphy within and above and below the excavation. Other improvements to the model were incorporated with the realization that transient strain could not be neglected, even long after excavation, because the continuing creep and consequent stress changes, although small at long times after excavation, were enough to cause continued transient creep behavior. The other major change was to the stress generalization, from Von Mises to a Tresca formulation. This change was supported by results from hollow cylinder, salt core tests in the laboratory. Incorporating these modifications to the model and database resulted in agreement between prediction and observation to within 10 to 15 percent--satisfactory for repository modeling purposes.
Full-scale simulation of DHLW, as well as a thermal overtest, were conducted in two test rooms, A and B. Room B and Room D provided data for identical geometry excavations, thermal effects in Room B being the only difference. In a later test a 36-inch diameter borehole 100 feet long, drilled between two-mined drifts, was instrumented to be certain that the creep phenomena observed in the large room tests and that the model adaptations to predict their behavior, were not the result of scale effects. The newly developed model has shown it can predict different scales, geometries, heated and unheated, with equal precision. Prediction of salt creep is now considered to be a resolved issue (6).
Discreet rock mechanics phenomena, such as fracture development and healing, have presented a challenging problem. The healing of salt fractures at stresses typical of the repository has been demonstrated in laboratory experiments. These experimental results support the logic that the disturbed rock zone that forms in salt around excavations will return to nearly in-situ values of density and permeability. This is an essential element in the design of effective long-term seals for the repository. Modeling has developed to the point that stress conditions and locations conducive to fracturing can be predicted as a rough function of time as an excavation deforms. Precise prediction of the time

of failure is still not possible from modeling although deformation rates obtained on failed rooms gives reasonable time estimates for rooms of the same geometry.

Response of the salt beds to internal room gas pressures caused by decomposition of the emplaced waste is still another concern to be addressed by the rock mechanics program. In-situ hydrofracture tests revealed the salt, remote from the excavation, behaves isotropically. The induced fractures don't show a preferred direction on this relatively small scale. Since the salt is layered on a larger scale, with the interbeds of anhydrite containing pre-existing fractures and with clay partings at the base of the anhydrites, these layer features will provide paths of least resistance to propagation of fractures on a large scale. These horizontal bedding features prevent fractures from developing upward through the salt, but they must be considered as potentially more permeable paths when evaluating propagation of fluids horizontally to the site boundary. Detailed calculation of discrete fracture development over large distances is not yet realistic because of natural heterogeneities in the rock. Consequently bounding approaches are taken to model this issue.

Sealing Systems (7)

Sealing systems proposed for the WIPP depend strongly upon the knowledge obtained from the rock mechanics program. The principal seals, those upon which isolation depends, will be in the access shafts into the WIPP and will consist of physically different modules which are designed to be most effective over different time scales (Fig. 4). WIPP in-situ experiments have demonstrated that cementitious seals can be very effective as soon after emplacement as the disturbed rock zone (DRZ) heals. Materials evaluation has permitted development of tailored cements that are relatively compatible with a salt environment and with the non-halite rock of the overlying aquifers. However it is difficult to prove that interface degradation will not occur over the millennia that are required of seals. Thus the WIPP will use materials natural to the WIPP environment, i.e., salt and clay, which will not be chemically foreign to the surroundings and will therefore remain stable for as long as the salt beds survive.

Fig. 4

Utilization of natural materials requires compaction to near in-situ salt densities. Experiments demonstrate that compacting crushed salt to >95% of natural density will achieve the desired permeability. This desired compaction can be initiated by tamping the emplaced material. Tests show that densities of about 90% can be attained by tamping but final compaction will rely on additional compression obtained over time from natural creep of the salt. Densification of crushed salt by applied pressure has been quantified in laboratory testing. Another advantage of tamping to achieve high emplacement density is that it decreases the time interval required to reheal the DRZ since resistive forces to creep closure build more quickly. Shaft seals relying on natural materials can be emplaced in the WIPP which will provide satisfactory seals within 100 to 200 years. During this time span, while natural material seals are becoming effective, sealing will be provided by cementitious seals at other locations higher in the shafts. Conservative properties attainable for the WIPP seals are presented in Repository Seals Program baseline position paper (7).

A third component, clay, either separately or in conjunction with crushed salt has also been evaluated. Clay may be useful to provide low permeability when water may be expected before salt creep has made the salt seals effective. Compacted clay will expand when wetted and intrude into openings and crevices, helping to reduce permeability. Another attribute of clay (but for which the WIPP does not take credit in calculating performance assessment) is its ability to sorb radionuclides should they be present in any fluid moving through a clay seal component. Should it become desirable to enhance sealing of individual waste rooms or panels or to maximize sealing of the water bearing shaft intervals, a special injection technique was developed and demonstrated in the WIPP Marker Bed 139. This development utilized a specially ground microfine cement and fluidizers to allow penetration into microfractures. The demonstration test defined the degree of penetration and the reduction in flow by comparing pre- and post-grouting flow tests.

Fluid Flow in the Geologic System

The hydrology of the WIPP site has been a subject of study since the initial site selection. Ground water flow is a possible dissolution threat as well as the primary transport mechanism for any breach scenario which may be postulated. Consequently it is important that the hydrologic system be known well enough to predict not only its present behavior but the response to future natural variations in climate as well.

Fluid Flow in the Rustler Formation (8)

The earliest studies, in the late 1970s, established that the Culebra Formation, with an average thickness of 7.7m, is the principal aquifer overlying the WIPP site and the aquifer of concern when modeling radionuclide transport. Numerous hydrologic test holes indicated the transmissivity over the WIPP area varied by several orders of magnitude with transmissivity generally decreasing to the east. Further testing which employed larger scale, long-term pumping tests, detected a "channel" of higher transmissivity (1-20 ft2/day), running north-south, on the east side of the site. This feature altered the water flow patterns somewhat but the general flow path from the repository to the site boundary is still generally southward. There has been much speculation about the cause of the general trends of transmissivity over the area and for the origin of the higher transmissivity channel. The most commonly accepted explanation for the general trend is the striking correlation with the extent of apparent salt dissolution from below the Culebra in the Rustler Formation. The more salt missing the greater is the observed transmissivity. The eastern edge of the WIPP site has all the salt still present in the section and the transmissivity is very low . The explanation for the channel is less obvious--there are no apparent structural or stratigraphic features that provide an explanation. Resistivity surveys help to provide additional spatial definition of this feature. Since geologic processes which could extend this feature or cause similar new ones are unlikely over 10,000 years, it is not critical that its origin be understood--only that we can incorporate its effect into our hydrologic model (Fig. 5).

Fig. 5

One of the most useful tools in the hydrologic test program was the implementation of the large-scale, multiwell pump test. This test could be implemented with or without the utilization of conservative tracers to accomplish several objectives. As a purely hydraulic test it has been used to establish effective transmissivity over intervals between pumping

and observation wells that range from tens to thousands of feet. By performing these tests at selected WIPP locations a much better approximation of the transmissivity field has been established. When used as a tracer test it has provided additional information on dispersivity, fracture characteristics and matrix diffusion, all important parameters in modeling transport of radionuclides.

Other observations, such as examination of the Culebra in core and in the air intake shaft, have provided insight into the nature of the porosity and fracturing. The physical variations observed in the Culebra have led to tracer studies where the different Culebra zones are isolated by packers to see which horizons contribute most to the transport and to what extent these horizons are isolated from each other.

Some chemical aspects of the Culebra saline brines south of the site are difficult to explain given the current flow model and the assumptions of steady state and confined aquifer conditions. An early interpretation, utilizing stable isotope and carbon-14 data coupled with geologic interpretations of past climate and recharge, led to an hypothesis that present conditions are not steady state but are recovering from recharge during a prior pluvial period. Current studies, using three-dimensional modeling capability, are examining the confined aquifer premise, allowing some small but perceptible level of vertical recharge to the Culebra. This concept of the hydrologic system can also explain the observed geochemistry. While it is not necessary to understand the origin of the chemical disparity to do transport modeling, a logical explanation lends confidence to our total understanding of the Culebra hydrologic system.

In addition to the field studies, extensive tests have been run to establish retardation parameters for actinides of interest. Many of these tests have been batch tests for Kd determination. Some have looked at the dolomite matrix while others have examined the clay which lines many of the Culebra fractures. A more realistic laboratory evaluation uses long cores taken horizontally in the Culebra by coring from the Air Intake Shaft. These cores, some with intact fractures, were used in flow-through tests using Culebra brine and actual isotopes. These studies provide not only a measure of chemical retardation but the extent of matrix diffusion or physical retardation. Even this best laboratory effort is at a small scale compared to the actual transport one wishes to simulate. Presently the Project does not plan on an imminent implementation of an in-situ field test with sorbing tracers. Such a test could require several years and, if conducted, should be viewed as an experiment to confirm that the conceptual transport model is not violated by observations and to establish beyond doubt that retardation does in fact occur at the field scale.

An extensive hydraulic and non-sorbing tracer test is now underway at the H-19 well complex. This test employs seven (7) wells to investigate such transport issues as dispersivity, fracture spacing and matrix diffusion. Flow in different stratigraphic layers of the Culebra will also be examined. The best interpretation of these results currently indicates that the Culebra aquifer is quite heterogeneous, and different portions of the aquifer are best modeled as either single porosity or double-porosity two-layer flow.

Fluid Flow in the Salado Formation (9)

Salt has always been considered a favorable host rock for a geologic repository because of its propensity to creep and recrystallize under stress load. This characteristic, plus the lack of observed dissolution

within the Salado, led to the assumption that water flow through the Salado would not play a significant role in repository integrity. Because of the potential for gas generation from the wastes, drill stem tests (DSTs) in the Salado were conducted from the ground surface in the late 1970s to establish whether formation permeability would confine the gas. At that time the precise stratigraphic location of the repository had not been established and the DSTs could only evaluate large intervals of the Salado. These tests indicated very low, but non-negligible permeabilities, which were sufficient to dissipate gas before build-up to high pressures. As a result of this finding, gas generation studies were terminated. Access to the underground in 1983 allowed additional observations and tests to address the issue of fluid flow in the Salado units in the immediate stratigraphic interval of the WIPP excavation. A limited number of early in-situ permeability and flow measurements in the salt indicated a disturbed rock zone of increased permeability extending a couple of meters outside the excavation and a "free field" permeability much lower (<10-20M2) than determined from the surface DSTs. Another observation was the seepage of brine to the surface of the excavations. This seepage was observed to vary lithologically, spatially, and to apparently decrease with time. Both observations led to extensive testing--to establish flow parameters of the Salado to explain and model the brine seepage and to implement gas generation experiments to better predict the potential for large volumes of gas. This was now necessary since the indicated low permeability would not allow for adequate escape of the generated gases. The added regulatory requirement (RCRA) to consider the transport of hazardous gases imposed still further incentive to understand both liquid and gas transport in the Salado.

Extensive testing of the Salado has provided considerable permeability data. The "pure" halite intervals exhibit permeabilities at or below the limit of our ability to measure--(10-23 M2). The argillaceous salt exhibits low but measurable permeabilities, one to three orders of magnitude greater than pure salt. Brine seepage has been studied by collecting accumulated brine volumes from many "sealed" boreholes. While the effects of the DRZ on these observations are impossible to rule out, the observations seem consistent with the conclusion that brine seepage occurs from argillaceous halite--not pure salt--and that interbeds and their associated clay seams are also a primary source for and transport of brine.

Despite the study devoted to this area, there is still dispute over the fundamental mechanism controlling long-term brine seepage. One view is that in the undisturbed free field, the little permeability that exists is not interconnected and therefore continued flow into the manmade porosity is not possible and brine seepage will decrease and eventually stop as the brine in the DRZ is depleted. The other view is that, although the permeabilities of the interbeds and argillaceous halite are very small, d'Arcy flow will govern brine seepage from these units, and brine will ultimately fill all porosity until internal room pressure balances the free field pore pressure. Because the operating mechanisms for either process are so slow, and the interference of the DRZ so difficult to segregate, the existing data can be explained by either model within the range of reasonable parameter selection.

A major effort to resolve this issue was to bore a 350-foot drift and seal the entrance with an effective airlock. Pore pressure data outside the walls and brine collection inside the room were obtained. Resistivity

measurements monitored the redistribution of brine in the DRZ around the room. Despite efforts to overcome the shortcomings of earlier, smaller-scale studies, the results of this test are inconclusive for the same reasons as stated above. Consequently, for purposes of performance assessment, the quantity of brine is established by assuming d'Arcy flow limited only by the hydrologic parameters appropriate to the situation.
Another aspect of the fluid flow in the Salado is the transport by fluids outward from the waste rooms driven by gas pressures in the rooms which potentially exceed the pore pressure in the formation. This is complicated by the need to consider the creation of fractures or separations at bedding planes if pressures exceed lithostatic pressure. The orientations of these openings are expected to be horizontal due to the presence of clay partings at the base of anhydrite interbeds and the presence of pre-existing fractures within the anhydrite. Both of these features may provide a minimum stress path for fracture propagation due to the lack of tensile strength across these features.
The geometry of propagation within these horizontal units is more difficult to establish. It is unlikely that a single fracture would propagate to the distance of the site boundary. Efforts of the petroleum industry to create horizontally extensive fractures has shown how unlikely this is. There may, however, be a tendency for a fracture network to develop in a generally preferred direction--perhaps influenced by such factors as regional dip. This may be accommodated in the modeling by assuming a range of "flaring angles" which are more restrictive in geometry than uniform radial propagation. Present expert opinion, however, supports the view that on a large scale, i.e. one to two kilometers, the fracturing and flow are best represented by a two-dimensional radial symmetry. It is not practical to implement an experiment which would be of sufficient scale to effectively examine this issue. The natural heterogeneity of the interbeds make extrapolation of small scale results of dubious value.

WASTE ROOM INTERACTIONS (10,11,12)

The starting point for all the WIPP performance assessment calculations is the waste disposal room and it is essential to have a good understanding of the possible range of physical and chemical conditions that control the source term and behaviors that can occur in the event of human intrusion. The physical process of room closure, fluid flow and chemical interactions are all closely coupled and can significantly affect one another.
The physical condition of the room due to creep closure can be reliably predicted as a function of time if the nature of the room contents (backfill, waste) is known. If the room is backfilled, the room closure will approach its final state of closure before internal gas pressures can build enough to provide much opposition to closure. The studies performed on consolidation of backfills such as salt or clay/salt mixtures show that they will compact to high densities and low permeabilities within 100-200 years. Therefore the nature of the waste form itself, which will vary over time with degradation, is the major uncertainty in determining the physical (and hydrologic) parameters of importance within the waste room. Experiments which assume various physical waste properties have examined the compacted properties such as strength and porosity, and indirectly, permeability. This range of parameter data is used by performance assessment to input such values as

the amount of spalled and entrained waste into flow up a human intrusion borehole (10).

The degree of porosity remaining after compaction by room closure determines the maximum amount of brine that can enter the room and cause metal corrosion and gas generation, and also establishes the amount of liquid available to solubilize the actinides.

The chemistry within a waste room will be a determining factor in the solubility of actinides in brine. Consequently, extensive laboratory tests have been conducted to establish solubility for the possible oxidation states of critical actinides at different values of brine pH. In the last few years, the question of colloid formation and their role in contributing to source term and transport have been examined. Studies indicate most colloids will not be stable in the brines present for the WIPP. Solubility, rather than colloid concentrations, is expected to be the major factor for establishing the waste room source term (11).

The source term predictions will be based on models developed using information from laboratory studies, but both solubility and colloids will be the subject of tests using real transuranic waste. These tests are of liter and drum size scale and are still underway at Los Alamos National Laboratory (LANL). The results will be used to lend confidence to the predicted range of values and, where necessary, will be factored into the model parameters.

The information obtained from the solubility program provides the knowledge base to tailor waste room conditions to lower the solubility of critical actinides. For example, backfill with a cement component would assure basic conditions for which plutonium has been shown to be less soluble.

GAS GENERATION (12)

As previously mentioned, gas generation studies were resumed when permeability studies indicated gas might not escape rapidly enough to prevent build-up of high pressure in the repository. This study received additional impetus when it became apparent that the WIPP would have to comply with the no-migration variance aspect of RCRA. While volatile organic compounds are small in volume, they could be carried towards the unit (site) boundary by the much larger volume of gases derived from waste degradation.

Gases will be generated in the WIPP primarily by anaerobic corrosion of iron and aluminum and by microbial decomposition of organics, principally cellulose. Radiolysis has been determined not to be a major gas generation mechanism relative to the other two. Recent studies confirm the gas production rates and potential established in the late 1970s. Provided sufficient brine is present, there could be enough hydrogen from metal corrosion and CO2, (primarily) from organics to generate enough gas to pressurize the repository, closed by creep, to pressures above lithostatic. Since pressure above lithostatic is not a realistic long-term pressure condition in halite, the repository would increase its volume and reduce its pressure by either expansion of rooms or creation of fractures as discussed in the prior section on rock mechanics. Studies also examined conditions which might lead to a decrease in gas production rates or volumes. Passivation of iron corrosion in the presence of high partial pressures of CO2 was considered for a time to be likely, but continued testing has shown this not to be the case. Experiments continue to show, however, that liquid phase water is necessary for corrosion to proceed--water vapor alone is not conducive to significant anoxic

corrosion rates. If sufficient brine is present for gas generation to proceed at its optimum rate, a few hundred years would be required for pressures to reach levels similar to lithostatic pressure. As gas pressure builds it will decrease, and eventually stop and reverse, the brine inflow into the room. Thus the brine-gas relationship is very interactive and potentially self limiting so that a saturated repository and lithostatic pressures may never be realized. There are factors which complicate this scenario and increase uncertainty such as the presence of a one-degree dip to the beds. This could allow brine to flow in at the bottom gas exits at the top. This is being examined in modeling calculations and will be incorporated in the final performance calculations.

Gas generation studies in the laboratory have been completed. At the Idaho National Engineering Laboratory (INEL), tests of gas generation which employ actual TRU wastes are still ongoing. These tests will serve to provide confidence in the laboratory test results and in the bounding values of gas generation the average stoichiometry model can be used to predict. Some gas generation data will also be produced by the source term tests at LANL although this is not the primary focus of these tests, and the test conditions could result in misleading rate data due to experimental conditions such as agitation.

This summary of scientific studies does not touch upon every experiment and test or upon every issue, but does indicate the major areas of investigation. With the exception of the real waste tests at LANL and INEL, all the currently-planned experimental data are now available.. In some instances it has been impractical to provide total closure to an issue through experimentation. That is where other arguments such as bounding approaches or independent professional judgment are adopted to provide an acceptable, conservative input to performance assessment.

Finally, although this discussion has focused on the experimental tests and geotechnical studies, the area of performance assessment has been a major program development in and of itself (13,14). While performance assessment requires the input from the technical studies, it is the performance assessment result and the associated sensitivity studies that can determine which experimental activities are required to improve confidence in compliance with EPA standards. Equally important, PA can be used to establish when enough testing has been completed in a given area, based upon compliance needs and uncertainty bounds indicated by the analyses.

Several studies have been carried out within the PA arena to allow the EPA probabilistic approach to compliance to be implemented. Codes which run rapidly, but reproduce the results of mechanistic calculations, allow the hundreds of necessary discrete calculations to be performed. Other issues, such as the probability of human intrusion, are critical in implementing the long-term performance modeling and have received considerable study within the PA group. The only practical approach to quantifying input parameters in this arena is through the application of expert judgment since no experiment can be devised to resolve the issues.

SUMMARY

Twenty-one years of geotechnical, chemical and physics studies have provided a comprehensive database to support a confident assessment of the WIPP's long-term performance. Some studies have fully resolved issues--others have established bounds on a range of uncertainty. The limits of our knowledge in these areas have been defined, and these

limits can be used in developing a conservative approach to compliance. The program will continue to assess the results of continuing experiments and other sources of new information to assure the performance assessment inputs and assumptions are consistent with knowledge obtained from longer duration tests.

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SUMMARY OF THE SYSTEMS PRIORITIZATION METHOD (SPM) AS A DECISION-AIDING TOOL FOR THE WASTE ISOLATION PILOT PLANT*

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ABSTRACT
In March 1994, the Department of Energy Carlsbad Area Office (DOE/CAO) implemented a performance-based planning method to assist in programmatic prioritization within the Waste Isolation Pilot Plant (WIPP) project with respect to applicable Environmental Protection Agency (EPA) long-term performance requirements stated in 40 CFR 191.13(a) and 40 CFR 268.6. This method, the Systems Prioritization Method (SPM), was designed by Sandia National Laboratories (SNL) to: 1) identify programmatic options (activities) and their costs and durations; 2) analyze potential combinations of activities in terms of predicted contribution to long-term performance; and 3) analyze cost, duration, and performance tradeoffs. SPM results were the basis for recommendations to DOE/CAO in May 1995 for prioritization within the WIPP project. This paper presents a summary of the SPM implementation, key results, and lessons learned.

THE SPM APPROACH

The goal of SPM was to provide information about how potential activitiestwenty-one scientific investigations, three engineered alternatives, and two waste acceptance criteriawhen viewed singly or in combination, could contribute to a demonstration of compliance with EPA long-term performance requirements for the WIPP disposal system (1-4). For each activity set (a combination of activities), SPM calculated the probability of demonstrating compliance (PDC) if the activity set was implemented, along with the activity set's projected cost and duration. These performance measures (PDC, cost and duration) were contained in a decision matrix that was analyzed to find programmatic options that maximized incremental PDC while minimizing activity set cost and duration. SNL performance assessment models were used to estimate how the disposal system might perform if activities were implemented, and this evaluation was the basis for calculating an activity set's PDC. SPM analyzed roughly 46,700 activity sets. Probabilistic performance calculations for these activity sets resulted in over 1.3 million complementary cumulative distribution functions (CCDFs).

As applied to the WIPP, SPM can be described in terms of eleven key steps (Fig. 1):

1. definition of the performance objective (i.e., long-term performance in 40 CFR 191.13(a) and 40 CFR 268.6);
2. development of a technical baseline for SPM calculations;
3. performance modeling of the baseline;
4. determination of whether the baseline was predicted to succeed or fail in meeting the performance objectives;
5. (if the baseline failed to meet performance objectives), identification of activities that, if implemented, could improve a predicted ability to meet the performance objectives;
6. evaluation of the baseline combined with potential outcomes of activities (i.e., calculation of the probability of demonstrating compliance);
7. creation of a decision matrix containing the PDC, cost, and duration for all activities and subsequent decision analysis to develop final recommendations;

8. DOE/CAO programmatic decisions about which activities to implement, if any;
9. implementation of activities;
10. re-definition of the technical baseline with actual results from the activities, iterating the overall process as necessary until the baseline is predicted to meet performance objectives; and,
11. when the baseline is predicted to comply, final compliance calculations with approved data and models, etc.

Fig. 1

A key to understanding how SPM works is in the relation between the output of the performance assessment models, the regulatory performance requirements, and decision analysis methods used to analyze results (5). It is also important to understand the role of expert judgment in performance assessment calculations.

Performance assessment models are used by the WIPP project to produce performance measures that can be compared to regulatory requirements (4). One such measure is a CCDF, which represents the probability distribution of summed normalized releases from the disposal system to the accessible environment. The WIPP disposal system is predicted to be in compliance with 40 CFR 191.13(a) if no point on the CCDF submitted in the compliance certification application to the EPA exceeds the summed normalized release limits.

While the regulatory release limits are fixed, estimates of predicted performance of the WIPP disposal system is not; they are determined by a state of knowledge that changes over time.Changing the state of knowledge through scientific investigations, implementing engineered alternatives, or modifying waste acceptance criteria can alter the position of the CCDF with respect to the release limits. Our state of knowledge can be expressed, in part, through probability distributions. For example, while the solubility of plutonium in WIPP brines is not known accurately at the present time, a range of solubilities under various chemical conditions and based on many types of existing information can be postulated, thus defining a portion of the WIPP disposal system technical baseline.

Envision the design of scientific experiments to more accurately determine the solubility of plutonium in brine. The experimental design anticipates a range of possible outcomes based on both published information and expert judgment. For simplicity, suppose that the experimental outcomes can be classified into five ranges (really probability distributions), from lowest to highest solubility. Denote the event that the experimental outcomes are in the first range by x1, in the second range by x2, etc. Denote the five possible probability distributions corresponding to the five experimental outcomes by f1, f2, etc. After the experiment is complete, our knowledge of plutonium solubility changes to reflect new information produced by the study. All uncertainty, however, will not be resolved by the experiments. Uncertain repository conditions make it impossible to know with certainty what the solubility will be. Therefore, after the experiments are completed, there is still residual uncertainty about the solubility which can, again, be expressed through a probability distribution that reflects the new evidence and that incorporates new expert judgments.

Now, suppose that we use expert judgment to specify potential experimental outcomes xi and associated probability distributions fi before conducting the experiment and use these distributions in performance assessment models to estimate the corresponding CCDFs: CCDF1,

CCDF2, etc. In addition to providing the xi and fi, we also use expert judgment to specify the relative likelihood or probabilities of the various events (xi), denoted by pi. Suppose that performance calculations predict events x1, x2, x3, and x4 will indicate compliance with long-term performance requirements but that the event x5 will indicate non-compliance. The predicted probability of successfully demonstrating compliance for the five events x1, x2, x3, x4, and x5viewed prior to conducting the experimentis then p1 + p2 + p3 + p4. This process is the fundamental basis for calculating the PDC of an activity set, the key measure of programmatic value in the SPM method. For the WIPP, this technique applies to any activity that can be expressed in terms of effects on WIPP performance assessment components.

Although this discussion has been restricted to compliance with 40 CFR Part 191.13(a), WIPP must also comply with the RCRA regulatory requirements in 40 CFR 268.6. The SPM criterion for success was that the CCDF is at all points less than the release limits and that the RCRA soil concentration limits are not exceeded. The compliance indicator (CI) for each activity set outcome indicates whether the 40 CFR Part 191.13(a) regulatory release limits and the 40 CFR 268.6 soil concentration requirements are met by the activity set. If both requirements are met, the compliance indicator is equal to one; otherwise it is zero.

For example, suppose an activity set composed of activities x1 and x2, each with two possible outcomes, and suppose that performance results show that quantitative performance requirements are satisfied only if activity x1 has outcome OA2 and activity x2 has outcome OB2. The compliance indices for each of the four possible activity set outcomes would then be equal to zero for all but the outcome consisting of both OA2 and OB2, which would have a compliance indicator equal to one. Once the compliance index (CI) values have been determined for each activity set outcome, PDC is calculated by summing the probabilities for all activity set outcomes where CI is 1. The PDC for the activity set consisting of x1 and x2 would then be calculated as follows:

Eq. 1

Thus, because all terms (outcomes) with a compliance indicator not equal to one would drop out of the PDC calculation, the PDC would equal PA2xPB2.

Because of the multiple possible outcomes of SPM activities, activity sets can have anywhere between two and nearly 60,000 possible outcome combinations, each of which corresponds to a CCDF and a compliance indicator. Thus, the PDC for an activity set represents a logically straightforward but very computationally intense set of calculations.

SPM-2 RESULTS

The first iteration of SPM (SPM-1), the prototype of SPM, was completed September 1994. It served to develop the tools needed for the second iteration (SPM-2), which was completed in March 1995 for programmatic decision making. SPM-2 used technical positions derived from WIPP project technical staff, stakeholders, and oversight groups as a starting point for establishing a baseline. Technical teams also defined proposed activities and were elicited on the predicted outcomes of those activities. Trained elicitors external to the WIPP project worked with the technical teams in a formal, structured process to elicit the parameters and models to describe the activity outcomes and the probabilities of those outcomes. Activity cost and duration estimates completed the activity descriptions. DOE/CAO and the Westinghouse Waste

Isolation Division provided information with regard to engineered alternatives, potential changes to waste acceptance criteria, and other programmatic guidance.

Potential outcomes were initially elicited for thirty-seven scientific investigations, eighteen engineered alternatives, and three waste acceptance criteria. These were screened to twenty-six discrete activities for the final SPM-2 analysistwenty-one scientific investigations, three engineered alternatives, and two waste acceptance criteria. SPM-2 used existing WIPP performance assessment computer codes, with modifications required to model the baseline and activity sets, to calculate CCDFs of potential radionuclide releases. SPM-2 evaluated more than 600,000 possible activity sets. Activities that had no performance impact were removed from the decision matrix, reducing the number of activity sets in the decision matrix to roughly 46,700. Because each activity set had multiple outcomes, approximately 1.3 million CCDFs were needed to complete the SPM-2 analysis.

For activities in the decision matrix, SPM-2 showed that the probability of demonstrating compliance generally increased, as expected, with increasing activity set cost and duration. Figure 2 shows the overall structure of the results in terms of the probability of demonstrating compliance versus activity set cost. The large cluster of diamond-shaped points (each one corresponding to an activity set) on the far left includes only scientific activities. Activity sets near the top of Fig. 2 all include one or more engineered alternative. Activity sets with a PDC of zero are not shown in Fig. 2 for reasons of clarity, but are contained in the SPM-2 CD-ROM, an information management tool produced as part of the SPM project (6). Programmatic dependencies were also apparent from general trends in the data and are discussed in the statistical regression portion of this paper.

Fig. 2

The SPM-2 baseline calculation predicted release of radionuclides in violation of 40 CFR 191.13(a) but compliance with respect to 40 CFR 268.6. About 40% of the SPM-2 activity sets also had a probability of demonstrating compliance of zero i.e., with no predicted value in supporting a demonstration of compliance. Of the remaining 60% of the SPM-2 activity sets, one half had a probability of demonstrating compliance equal to one. When conducted alone, no single activitywhether a scientific investigation, an engineered alternative, or a waste acceptance criterionhad a non-zero PDC.

Activity sets with a PDC of 1.0 included the scientific activity for colloids investigation and one of two engineered alternatives: either, (1) backfill and a pH buffer to control actinide solubility (EA1), or (2) an engineered backfill (such as clay) in combination with waste form modification (EA2). (Note that engineered alternatives and waste acceptance criteria were assumed to be optimally effective and were assigned a 100% probability of yielding the predicted performance. Subsequent sensitivity studies investigated the impact of this assumption on the final decision.) Two waste acceptance criteria (WAC) were analyzed by SPM-2. In the WAC-1 activity, steel drums used to store the waste were replaced with noncorrodible materials. WAC-1 added costs to the program and slightly reduced the probability of demonstrating compliance. WAC-2, elimination of all high-molecular weight organic compounds (such as soils) from the waste, had no discernible impact on the probability of demonstrating compliance.

The sensitivity of SPM-2 results to the probability of engineered alternative performance was straightforward to evaluate (see Ref. 3 for details). The DOE/CAO had a preliminary decision to make, either:

1. depend on a program consisting of engineered alternatives and minimal scientific investigations to provide a basis for the final compliance calculations; or
2. reserve engineered alternatives for possible use in providing assurance and depend on the scientific investigation to demonstrate compliance.

In May 1995, DOE/CAO chose the second option. Additional work has been conducted on engineered alternatives since the completion of SPM and the final balance between predicted performance of the geologic system, engineered alternatives, and waste acceptance criteria will be described in the compliance certification application to the EPA (7).

The final programmatic recommendations made to DOE/CAO in May 1995 considered the SPM-2 results along with existing information such as the 1992 WIPP PA Sensitivity Analysis (4) and some sensitivity and uncertainty analyses. The sensitivity and uncertainty analyses did not alter the recommended series of activities. Other issues that were considered in using the SPM-2 results for decision-making were:

1. The technical baseline was for SPM use only. The final project technical baseline that will be used for preparing the WIPP compliance certification application will incorporate information from the activities completed subsequent to the SPM-2 effort.
2. The results were based on calculations using mean values, and were therefore valid for discriminating between activities intended to shift a mean value for a parameter but not for discriminating between activities intended to reduce uncertainty about a mean.

ANALYSIS OF RESULTS

The SPM-2 project generated roughly 46,700 unique activity sets. In order to understand the structure of the probabilities of demonstrating compliance among these activity sets, a statistical regression analysis was conducted. This analysis employed a logit regression methodology. A logit regression assumes that a probability, p, (or other number bounded by zero and one) is related to several independent variables through Eq. 2:

Eq. 2

In Eq. 2, xi are indicator variables (0,1) and bi are regression coefficients to be estimated. Here, p is the probability of demonstrating compliance. Because the left side of the equation is unbounded at p=0 and p=1, the probability of demonstrating compliance values were decreased slightly towards 0.5 as shown in Eq. 3:

Eq. 3

An initial inspection of activity sets in the decision matrix revealed two very strong relations. First, if neither colloid activity (NS 8.1 nor NS 8.2) was included in an activity set, the probability of demonstrating compliance was zero. Second, if either NS 8.1 or NS 8.2 was in an activity set, the probability of demonstrating compliance was one as long as an engineered alternative (EA 1 or EA 2) was also in that activity set, and less than one otherwise. Both of these relations were always true and thus the first relation provided a sufficient condition for creating a probability of demonstrating compliance to equal zero. The second relation provided a condition that was both necessary and sufficient for probability of demonstrating compliance to equal one.

These two relations logically limited the PDC of activity sets without EA 1 or EA 2 to 0<PDC<1.

In the absence of EA 1 and EA 2, what scientific programs should be undertaken to achieve a high probability of demonstrating compliance? This question was important because the predicted performance of EA 1 and EA 2 did not account for the possibility that an EA might prove less effective than assumed. Moreover, there were reasons to believe that the system-wide costs of EA 1 and EA 2 might ultimately be larger than initially estimated. For these reasons and to better understand the cost/benefit tradeoffs for the scientific program, a statistical analysis was limited to those activity sets where: both, (1) NS 8.1 or NS 8.2 was present, and (2) neither EA 1 nor EA 2 was present.

Using the logit model and excluding from the data set those activity sets without either NS 8.1 or NS 8.2 and excluding those having some combination of colloid activity with EA 1 or EA 2, regression coefficients were obtained. Based on regression results, activities are ordered from those with the greatest impact to those with the least impact, creating a series of activities such that as activities are added to the series, the PDC continues to increase but at a decreasing rate (see Fig. 3). If the costs of the activities are similar, it is, in principle, possible to build a concave, monotonically increasing function that maximizes incremental PDC gained while minimizing incremental costs as more activities are added to the series. Two such activity series are shown in Fig. 3 (the two curves on the left-most side of the graph) but they are not fully concave. The far-left curve is unconstrained by duration while the middle curve is constrained by a 19-month duration. The reason that these curves are not fully concave is that there are both thresholds and interactions (synergies) among some activities. The right-most curve in Fig. 3 is a sub-optimal activity series that ultimately reaches nearly the same PDC as the pareto-optimal series but without the same ability to maximize incremental PDC per dollar at every point in the series.

Fig. 3

For both the duration-constrained and unconstrained activity series in Fig. 3, no improvement in the probability of demonstrating compliance was obtained by performing NS 8.1 by itself. (Here NS 8.1 was chosen over NS 8.2 because of equal impact on the probability of demonstrating compliance and lower cost for NS 8.1.) However, for the duration-constrained series, the addition of NS 2 and NS 4 increased the probability of demonstrating compliance to 0.56. Addition of NS 7 further increased the PDC to 0.82. As Fig. 3 shows, the addition of AST 1.2 did not increase the probability of demonstrating compliance. However, AST 1.2 was necessary to gain the PDC improvement provided by the combination of RM 1, SL 4, and DR 2. In fact, without first performing AST 1.2, the addition of RM 1, SL 4, and DR 2 produced a decrement in PDC. The same unexpected behavior occurred when one switched the order of the activities. One therefore concludes, that some interaction is taking place between AST 1.2 and the collection of three activities. Addition of any other activity to the series only brings minuscule improvements. A probability of demonstrating compliance of 0.96 is achieved from the duration-constrained pareto-optimal series.

The two left-most series are both considered pareto-optimal, that is, the series that cannot be bettered simultaneously in both cost and probability of demonstrating compliance. Faced with programmatic options

limited to the scientific programwithout engineered alternatives or waste acceptance modificationsboth the duration-constrained and unconstrained activity series appear to be logical programmatic choices. However, the duration-
constrained series, which eliminated two scientific activities (NS 3 and NS 5), resulted in virtually the same probability of demonstrating compliance as the unconstrained set and with lesser cost. The duration-constrained series was selected for implementation by the DOE/CAO.

DISCUSSION

Lessons Learned

The SPM-2 decision matrix yielded valuable information for identifying: 1) activity sets necessary to achieve a given PDC; 2) activity sets that give the maximum PDC; and 3) activities that have minimal impact on the PDC. Moreover, the use of quantitative analyses balanced with expert judgment was essential in developing insights about decision options in a highly nonlinear system. The SPM project required a significant commitment of human and computational resources but numerous improvements could be made to increase efficiency.

Information needed for the SPM analysis was acquired as expert elicitation from individuals directing the various activities and those proposing new activities. Adequate time for training participants in an expert elicitation process is essential.

Concerns were raised that the SPM baseline was excessively conservative and would not produce a useful basis for evaluation of activities. A management review was held to assure that the baseline was, in fact, appropriately balanced and integrated and that it was acceptable as the basis for performance calculations. Review is recommended of both the baseline and activities prior to performing calculations to assure appropriate consistency and integration of information elicited from many different individuals.

Side efforts (also known as side bar calculations, or side calculations) were also important in being able to keep the probabilistic calculations tractable and in preventing unnecessary conservatism in the baseline. Side efforts refer to confirmatory evaluations required to address certain technical positions embedded in the SPM-2 baseline. These confirmatory evaluations included scenario screening work, literature searches, bounding calculations, and some computer modeling.

There are computational limitations to probabilistic calculations underlying SPM. Suppose, for example, that m activities are to be considered and each activity has k potential outcomes. The number of endpoints to be evaluated is Si=1,m kim!/[(m-i)!i!], which becomes very large, very quickly. Clearly, not all combinations of activities can be evaluated. But this is where judgment and an understanding of disposal system performance can be used to create reasonable sets of activities for evaluation. Other computational schemes, such as sampling certain computational intensive parts of the performance assessment model, should be explored. In addition, multi-attribute utility analysis techniques (5) could be useful for up-front screening and focusing an initial large set of potential activities into a smaller set that require quantitative evaluation.

The usefulness of an SPM-like method depends upon the quality of the elicited information about activity outcomes, their probabilities, and the state of knowledge about system parameters and conceptual models. Retrospective analyses of SPM results can assess the degree to which

actual outcomes were consistent with elicited predictions. Bayesian updating methods could be used if SPM were applied on an iterative basis.

SUMMARY

SPM identified viable combinations of programmatic options (activities) that, if implemented, were predicted to lead to a positive demonstration of compliance with long-term performance requirements. Moreover, analysis of the results also indicated that optimal programmatic pathways existed and that these activity series could provide useful insights into which activities to cut or add if budgets changed. Indeed, the analysis indicated that a positive demonstration of compliance with the long-term performance requirements could be anticipated within the DOE/CAO schedule.

SPM focused on work to achieve compliance with long-term disposal system performance requirements and helped eliminate concerns that activities would merely contribute to scientific knowledge. SPM utilized the existing performance assessment codes to calculate the expected results of various programmatic options. Use of quantitative performance assessment tools for prioritization was essential in gaining insights into the behavior of a highly coupled, nonlinear disposal system. SPM built upon the power of both performance assessment and decision analysis techniques, providing insights for decision making.

The general method could be applied to other complex issues in the environmental and waste management arena that need to clearly focus scientific and engineering activities on specific (and measurable) objectives within cost and schedule constraints. Because SPM combines decision analysis methods with quantitative analyses, it is conceptually applicable to any complex problem for which performance objectives, performance measures, and options to achieve the performance objectives can be defined. Projects that would likely benefit most are those with a complex set of technical issues and decision options that would benefit from planning based on calculated performance, rather than expert prediction alone. Projects with significant stakeholder involvement or with multiple participants might also benefit. Finally, probabilistic techniques used to treat uncertainties in the physical system could also be used to treat uncertainties in the cost or duration of programmatic alternatives (8).

ACKNOWLEDGMENTS

This paper is a summary of work done by many people who are specifically acknowledged in a number of reports now in preparation. The authors gratefully acknowledge their participation and important contributions to this project. We would like to specially acknowledge Walt Beyeler for his significant contributions to many aspects of SPM.

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NEW MODELS FOR PUBLIC PARTICIPATION EVOLVE WITH DOE'S PARADIGM SHIFT

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ABSTRACT
As the Cold War and its artifacts came tumbling down, DOE's veil of secrecy began lifting. In the early 1990s, DOE began implementing its new culture of involving the public in its decision making process. Most meetings held by DOE facilities were tied to a regulatory requirement or a specific milestone for which DOE was seeking stakeholder input.

The Public Affairs Office at DOE's Carlsbad Area Office, which manages the Waste Isolation Pilot Plant (WIPP), is undertaking a series of meetings that break out of the traditional meeting format and establish a new model for more meaningful involvement -- by both stakeholders and the DOE. It includes the following:

A stakeholder planning group to help plan the stakeholder forum;

An independent facilitator to serve as the focal point for organizing and managing the meeting;

A policy discussion on broad-ranging issues related to the WIPP; and

Recording the meeting to preserve the ideas generated and shared for use by current decision-makers and future generations.

Key in this new meeting model is a component for feedback to stakeholders to show how their input was or was not used.

This paper documents the evaluation of a meeting format and lessons learned by working with stakeholders in the design of public participation activities.

INTRODUCTION

When the Waste Isolation Pilot Plant (WIPP) was sited near Carlsbad, New Mexico 20 years ago, the U.S. Department of Energy's (DOE) standard procedure for interacting with those affected by any of its activities was somewhere between what the DOE has termed the "Control Paradigm" and the "Public Relations Paradigm." That is, the DOE believed that while it was no longer exempt from public scrutiny, its employees were nonetheless considered the "experts" and the public should trust and accept their decisions.

During that same general timeframe, more stringent environmental laws, such as the National Environmental Policy Act of 1969 (NEPA), the Resource Conservation and Recovery Act of 1976 (RCRA), and the Comprehensive Environmental Response, Compensation and Liability Act of 1980 (CERCLA), imbued the public with a new vision of what they should expect when it comes to protection of the environment and what they can do to protect themselves when it becomes contaminated. These laws required both public and private interests to become more accountable not only to the federal government for their impact on the environment, but to the public as well.

Over time, the result was that parts of the DOE in general and its Environmental Restoration and Waste Management (EM) Program in particular, began to adopt a Public Participation Paradigm. This paradigm shift allows for the views of "stakeholders", or parties interested in DOE decisions to be integrated into the DOE's decision-making process. Stakeholders include interested or affected individuals, organizations, state and local governments, Indian tribes, environmental groups and other federal agencies.

As the DOE's WIPP program progressed, so did its interactions with the public, which slowly changed to reflect this new paradigm. Scientific and philosophical controversies surrounding the project -- which were avoided or even ignored -- are now actively engaged. The DOE's Carlsbad Area Office (CAO), which has responsibility for the WIPP, recognizes that the public determines whether a risk is acceptable or not -- and that the public's concerns and perceptions of risk carry equal validity with risk as determined by scientific theory. In WIPP's case this would include questions such as whether radioactivity or hazardous materials will move beyond the boundaries of the WIPP for 10,000 years.

WIPP INVOLVEMENT STRATEGIES UNDER THE PUBLIC PARTICIPATION PARADIGM

The First Stakeholder Forum

In early 1993, the DOE's EM Program issued guidance on stakeholder involvement. Coincidentally, the DOE began to re-think its program to perform tests with radioactive waste in the WIPP underground. The tests were designed to increase confidence in the WIPP disposal system, including the handling, packaging, transportation and emplacement of waste underground.

Consequently, the DOE held a WIPP program review -- in actuality a stakeholder forum -- in Washington, D.C., with 45 participants and 30 observers from across the country to discuss issues regarding the WIPP Test Phase. Stakeholders were called upon to assist the DOE in assessing the current WIPP program strategy and exploring alternatives in light of budgetary constraints and the passage of the WIPP Land Withdrawal Act (LWA). Another purpose of the meeting was for the DOE to gather perspectives on how it can advance the program in a way that is acceptable to stakeholders.

This meeting was particularly significant because it was one of the first meetings DOE has had with citizens in which input was actually sought and incorporated into actions. For example, when Assistant Secretary Grumbly asked the group about the soundness of DOE's intention of performing tests with radioactive waste in the WIPP underground, he received significant, thoughtful response from stakeholders.

Two participants from the National Academy of Sciences (NAS) recommended that DOE not go forward with planned tests. They and several others suggested the tests with transuranic wastes in the WIPP underground lacked scientific justification and needed to be better designed. They said that the planned experiments would not attempt to confirm the entire WIPP performance assessment computer models, but only a part of them. Instead, they recommended, the DOE should perform short-term tests on key assumptions and hypotheses, and extrapolate long-term performance projections from the short-term tests.

Relatedly, stakeholders expressed concerns about the cost and budget of the project. Assistant Secretary Grumbly asked the group how they would trim the budget, given the opportunity. A member of a technical oversight group said he would focus the testing program on compliance issues and eliminate unnecessary tests.

Outcomes of the First Stakeholder Forum

In keeping with its new public participation guidance, DOE officials considered the concerns and suggestions raised at the meeting and three months later announced a decision to conduct radioactive waste tests in laboratories and re-direct WIPP's focus to showing compliance with the Environmental Protection Agency regulations set forth by legislation. In response to concerns expressed by local leaders, the Secretary also established the CAO in Carlsbad, directing that DOE staff working in Albuquerque on the project move to Carlsbad. These actions validated the DOE's willingness to involve stakeholders in the decisions the Department makes.

The Second Stakeholder Forum

In late September 1994, the CAO held a second stakeholder forum on the WIPP. This meeting included another evolutionary component: it was organized as a series of citizen presentations during which DOE officials listened and responded to questions as appropriate rather than make any formal presentations or defend its position on issues. The meeting immediately preceded a visit from the Secretary of Energy to Carlsbad the

following week. Stakeholder concerns aired at the forum were summarized and included in the Secretary's briefing book for the Carlsbad visit. In addition, stakeholders were invited to select a representative to make a presentation on selected issues during the Secretary's visit.

Stakeholders' major messages, representing divergent points of view, to DOE/CAO officials included the following:

The DOE should provide stakeholders with timely notice of meetings and adequate time for document review. DOE's compressed schedule may affect the quality of stakeholder interactions and relationships.

The disposal decision schedule is too inflexible; "schedule" should not drive good science.

The DOE needs to examine the ethics of permanently disposing of radioactive waste in the WIPP and evaluate potential impacts on future generations.

The DOE must change the way it works with Indian tribes and recognize each is a sovereign nation.

The DOE must further examine issues of waste inventory uncertainty, engineered alternatives, treating the waste before it comes to the WIPP, site characterization, and the WIPP's ability to contain the waste safely for 10,000 years.

The DOE should continue emergency response and hospital personnel training and provide public education programs on transportation risk.

The WIPP needs to open as soon as possible. It is a solution to a national problem, and continued delays are costly.

Despite coming from different perspectives on the WIPP, participants praised the meeting for bringing together divergent views and giving both project opponents and supporters a chance to speak their minds in an open forum.

Outcomes of the Second Stakeholder Forum

In response to these stakeholder issues, the CAO made some key changes. It sought to provide earlier meeting notice and longer document review time wherever possible. It stepped up its outreach program among tribes and pueblos. And it has been deeply involved in waste inventory definition; evaluation of engineered alternatives, including solicitation of stakeholder views on proposed alternatives; consideration of waste treatment options; and long-term performance assessment.

Three issues remained to be resolved: exploration of the ethical issues, the disposal decision schedule, and site characterization. A major goal of the stakeholder forum planned for 1995 had been to explore ethical issues related to permanent disposal of transuranic waste at the WIPP. Regarding the compression of the disposal decision schedule, the CAO is committed to respond positively to the equal or stronger pressures to maintain or accelerate the schedule. Later, in response to congressional pressure, the CAO has accelerated the disposal decision date from January 1998 to October 1997. Finally, site characterization has been completed, and without apparent need to do so, DOE does not plan to revisit that issue.

The oversight in CAO's response, however, was that the overall response was not documented and provided to those who attended the forum.

The Third WIPP Stakeholder Forum

At the conclusion of the second WIPP stakeholder forum, the CAO Manager invited stakeholders to take the lead in setting the agenda for the next stakeholder forum, which would take the WIPP's stakeholder meeting model to its next step. The CAO committed to holding another forum by April or

May 1995 -- but for a variety of reasons, the date was preempted and postponed. In late Spring 1995, the CAO held several meetings on the draft Compliance Certification Application (CCA) and the RCRA Part B application. In early Fall, scoping meetings for the second Supplemental Environmental Impact Statement (SEIS) superseded the forum.

In the interim, a former manager with the New Mexico Environment Department (NMED), a person with significant credibility and contacts within the advocacy community, was sub-contracted to support, among other things, CAO's efforts on the next stakeholder forum.

Part of the CAO's commitment at the September 1994 forum was that stakeholders would be integrally involved in the planning of the meeting, and that it would not be another DOE-planned meeting. Thus, in the Fall of 1995, the CAO's sub-contractor set about organizing a stakeholder planning group to develop plans for the next stakeholder forum, which the CAO hoped to hold before the end of the year.

The most surprising outcome of the initial planning meetings was that the planning group advised against trying to meet before February 1996. At the first planning meeting, stakeholder participants said the time was too short to meet without conflicting with previously planned events, such as tribal elections. In addition, they recommended that the CAO hold forums in smaller communities along the waste transportation routes, in addition to the proposed forum in Albuquerque.

Further, some members refused to participate in the planning meetings until the CAO provided them with the budget for the overall public outreach program and the proposed forum, and documentation of responsiveness to concerns expressed at previous meetings and forums. The overall public affairs budget inside the DOE generally had become an issue in this same time frame, focusing the attention of state advocacy groups on the WIPP as one office within the DOE. The issue accelerated when an Albuquerque paper ran a story about the number and make-up of the WIPP's outreach office, and the request for both the overall and forum budget numbers should not have been a surprise.

More of a surprise was the request for documentation of responsiveness. The CAO has prepared responsiveness matrices for most of the meetings in question. These matrices provided a thorough, organized way to track who was responsible for what, and individual action items were checked off one-by-one. What had not been done as tenaciously, however, was to summarize, at the time when all the action items had been completed, the CAO's response to issues brought up at the forum and to transmit its response systematically to stakeholders. We set about taking that crucial final step.

At a second planning meeting, the group recommended that the CAO not hold the Albuquerque forum in February at all, but rather, concentrate on the recommended smaller meetings and devote available funding to the outreach and involvement program for the second SEIS. They also asked that the CAO invite representatives of diverse points of view to participate in other meetings it has with the public. After further discussion, the group agreed that it would be helpful to the public for any of the concerned entities -- the CAO, the state, or specific interest groups -- to invite representatives of diverse viewpoints to participate in meetings they may hold. This way, stakeholders would not have to attend many meetings to get a balanced picture of the issues. The group also asked the CAO to publish diverse viewpoints.

LESSONS LEARNED

Lessons learned are always somewhat more apparent in hindsight than they are in the making. The lessons below guide our future interactions.

1. Working with stakeholders can have beneficial effects on other programs. Collaboration between the CAO and its stakeholders in advance of the planning of the stakeholder was beneficial to the overall outreach effort. In our pre-forum planning, we quickly identified unresolved issues that we took action to correct.
2. Don't get stuck in a rut. Just because something works well the first and second time does not mean it will work well the third time. We found that although stakeholders praised the forums of 1993 and 1994, it was not necessarily the right thing to do in 1995. Stakeholder interests and concerns should be viewed as an on-going "video," rather than a "snapshot."
3. Be flexible. The CAO wanted to hold another stakeholder forum, but stakeholders stated loudly and clearly that their needs would be satisfied better by smaller meetings in different locations; by meetings at the generator sites; and by a beefed-up hearing process for the SEIS. The CAO's response -- without acrimony -- was, "If you do not need this meeting, we do not want to waste resources organizing it."
4. Be innovative. The CAO's initial response to the "rejection" of a stakeholder forum could have been negative. Rather, it was taken as an opportunity to work with stakeholders to develop a new model for stakeholder interactions, particularly meetings. As of the writing of this paper, CAO intends to continue to work with stakeholders to achieve a model that fits their information needs and CAO's outreach and involvement goals.
5. Tie stakeholder interactions to decisions to be made. The stakeholder forum of 1993 was for the purpose of soliciting feedback on specific decisions the DOE had to make. The 1994 version had no similar "hook," which perhaps caused stakeholders to see it as less fruitful than its predecessor. The CAO did, however, make changes in response to that forum; they simply were not as far-reaching as those of 1993, nor did the CAO provide the clear message of its changes. Stakeholders might have viewed the forum as more successful had they had a clearer idea of its impacts. One lesson here is that action items the DOE agrees to take should be viewed as promises -- as commitments to be kept. The related lesson follows in #6.
6. Make sure responses to concerns are relayed to stakeholders as soon as possible. While the CAO had tracked and responded to stakeholder concerns, it had not completed the circle at the end of the process by preparing a summary of responses and providing them to all who attended the meetings in question. This lack of an "overall" response made the CAO appear to be unresponsive. In these changing times of "more is better," it becomes a refreshing view of a new reality to see stakeholders and the DOE working together to identify the best uses of limited and shrinking funds for the involvement of the public in waste management decisions.

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ACTIONS TO BE COMPLETED TO OPEN WIPP

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ABSTRACT

The Waste Isolation Pilot Plant (WIPP) is intended to be a repository for defense transuranic (TRU) waste in the bedded salt deposits of Southeastern New Mexico. Exploration for the WIPP site began in 1974 but the repository is not open yet. The reason for this delay is that the U.S. Department of Energy (DOE) has not yet completed the documentation to show that the TRU waste can be safely disposed in the repository. From 1974 to 1993, the DOE and its predecessor agencies planned to first emplace varying amounts of TRU waste at the underground repository for "Research and Development" or "Operational Demonstration", etc. Only since 1994 has the DOE focused on completing the scientific evaluation necessary to assess WIPP as a permanent repository. Much work still needs to be completed to prepare a robust document showing compliance of WIPP with the Environmental Protection Agency (EPA) Standards for safe disposal of transuranic waste. This document must then be certified for compliance with the disposal regulations by the EPA. The DOE plans to submit the documentation in the form of an application to the EPA in October 1996. The WIPP Land Withdrawal Act (Public Law 100-579) of 1992 requires the DOE to also demonstrate compliance with other relevant health, safety and environmental regulations. Finally, the WIPP facility must be operationally ready with radiation protection measures in place to start receiving waste for disposal and the DOE waste generators must be ready with appropriately characterized waste to start shipping the waste to WIPP.

INTRODUCTION

The Waste Isolation Pilot Plant (WIPP) has been constructed to be a geologic repository in bedded salt formation in southeastern New Mexico for permanent disposal of defense transuranic (TRU) waste. The repository is located 40 km east of Carlsbad, New Mexico, in the northern part of the Permian age Delaware Basin, at a depth of 655 meter below the surface. The geohydrologic setting of the site and a summary of the geohydrologic issues that affect the WIPP performance assessment are summarized in Chaturvedi (1993)(1). A summary of the unresolved performance assessment issues for the WIPP is provided by Chaturvedi, Lee, Silva and Neill (1995)(2).

The Environmental Evaluation Group (EEG) was established in 1978 as an interdisciplinary group of scientists and engineers to provide an independent technical evaluation of various aspects of the WIPP project to protect the health and safety of the people of New Mexico. The group is funded 100% by the DOE and has offices in Albuquerque and Carlsbad.

WHY IS WIPP NOT OPEN YET?

Site investigations for the WIPP in southeastern New Mexico began in 1974 following abandonment of the Lyons, Kansas site in 1972. The initial site for WIPP was located about 11 km northeast of the present site but was abandoned after a borehole (ERDA-6) indicated unacceptable geologic deformation, and pressurized brine was encountered in that bore hole at a depth of 826 meter. The U.S. Department of Energy (DOE) and its predecessors planned the WIPP to be a "pilot plant" and a "research and

development facility", and expected to justify emplacement of large quantities of waste as much as 200,000 drums of contact-handled transuranic (CH-TRU) waste in 1982 (3) before making the decision to use the facility as a permanent repository. This attempt to temporarily emplace some waste at WIPP continued until 1993 when it became clear that a scientific justification for experimenting with waste in an underground repository did not exist and the cost and engineering problems were too overwhelming to justify such an endeavor. The WIPP is not open for waste in 1996, 22 years after the site investigations began, because the DOE did not focus on demonstrating the suitability of the site and the facility for permanent disposal of TRU waste. Because of the lack of this focus, scientific and analytical work was not targeted towards the goal of making the decision to use the WIPP as a permanent repository until 1993.

PROCESS OF DECISION MAKING FOR A PERMANENT REPOSITORY: THE EPA STANDARDS

The purpose of WIPP is to isolate transuranic radioactive wastes that are hazardous due to the long-lived alpha emitting actinides. The WIPP waste will contain about 9 metric tons of Plutonium-239 with a half-life of 24,000 years. Thus, the decision to permanently dispose waste in such a repository has to be based on projection of the repository's integrity for several tens of thousands of years. A formal process specifically developed to conduct such an analysis is described below.

Beginning in 1979, both DOE and EEG did deterministic calculations of the radiological consequences of the radioactive waste returning to the biosphere. The 1980 DOE WIPP Final Environmental Impact Statement established that the peak discharge activity would occur at 1.3 million years. In 1985, the U.S. Environmental Protection Agency (EPA) promulgated "Environmental Radiation Protection Standards" (40 CFR 191) (4) for the disposal of high level and transuranic waste. Subpart A of these standards specifies maximum allowable radiation doses received by members of the public as a result of management and storage of transuranic waste, i.e. during operations. Subpart B of 40 CFR 191 specifies probabilistic limits of release of radionuclides to the environment for 10,000 years and requires demonstration of projected releases meeting these standards through probabilistic risk analysis called the "performance assessment". The following justification was provided by the EPA for selecting a period of 10,000 years for analyzing the integrity of a repository.

A period of 10,000 years was considered because that appears to be long enough to distinguish geologic repositories with relatively good capabilities to isolate wastes from those with relatively poor capabilities. On the other hand, this period is short enough so that major geologic changes are unlikely and repository performance might be reasonably projected. (Federal Register, Vol. 50, No. 182, pp. 38070-71, September 19, 1985).*

The EPA's disposal standards were legally challenged in 1987 on grounds, among others, that they were less stringent than the Clean Water Act of 1971, without providing any justification for the difference. The First Circuit Court of Boston remanded the Standards to the EPA for revision and repromulgation. The State of New Mexico entered into a formal agreement with the DOE within a few days to continue to evaluate WIPP against the vacated 1985 Standards because the standards were not expected to change significantly. The revised Standards were published 6.5 years later on December 20, 1993 (6). As directed by the WIPP Land

Withdrawal Act (P.L. 102-579, Sec. 8), amendments were made only to those parts of the Standards in which the Court had found fault, leaving the bulk of the Standards unchanged.

The process of assessing compliance with the EPA Standards and the status of the unresolved issues in demonstrating such compliance was described by Chaturvedi, Lee, Silva and Neill (1995) (2). The following section describes a history of this effort and a January, 1996 update on the status of the unresolved issues.

The criteria for compliance (40 CFR 194) with the standards for safe disposal (40 CFR 191) are scheduled to be issued by EPA in February, 1996.

EVOLUTION AND STATUS OF THE WIPP PERFORMANCE ASSESSMENT

The task of assessing the WIPP's compliance with the EPA Standards (4) was first assigned by the DOE to the WIPP Management and Operating Contractor, the Westinghouse Waste Isolation Division, in 1985, and then to the Sandia National Laboratories in 1986. Between 1986 and 1988, the DOE focused on making preparations to start receiving waste in October 1988. One of the 8 panels, consisting of 7 rooms, each 91.5 m x 10 m x 4 m (300 ft x 33 ft x13 ft), enough to store approximately 40,000 CH-TRU drum equivalents in the rooms, was excavated during this period. But the first "Preliminary Comparison with 40 CFR 191, Subpart B for the Waste Isolation Pilot Plant" (7) was not published until December, 1990. Two additional "Preliminary Comparison with 40 CFR 191" (8) and "Preliminary Performance Assessment" (9) were published in December, 1991 and December 1992. The WIPP Land Withdrawal Act of 1992 (P.L. 102-579) allowed the DOE to emplace up to 0.5% of the total planned capacity of the CH-TRU waste at WIPP for a test phase and also required the DOE to submit an application for certification of compliance with 40 CFR 191 Subpart B within 7 years after the first receipt of TRU waste at WIPP.

In less than one year after the promulgation of the WIPP Land Withdrawal Act, however, the DOE abandoned the idea of conducting experiments with actual TRU waste in the WIPP underground. EEG had recommended that this decision be taken and the reasons are described in Chaturvedi and Neill (8).

THE DOE SCHEDULE FOR THE DISPOSAL DECISION

After abandoning the plans to conduct experiments with TRU waste in 1993, the DOE announced an accelerated plan to demonstrate WIPP's compliance with the EPA Standards, 40 CFR 191 Subpart B. A "Systems Prioritization Method" (SPM) plan was instituted in early 1994 to determine the optimal set of "activities" (laboratory and field experiments, numerical simulations, analyses, etc.) that would demonstrate compliance. The SPM was the focus for the DOE compliance activity from March 1994 to May 1995. On June 2, 1995, the DOE announced the decision to conduct some experiments and not conduct others, based on the SPM analysis. The EEG's analysis of the SPM results showed that the particular activity set selected by the DOE was not unique for the assumed probability of demonstrating compliance, the cost, and the duration parameters. There appear to be 690 other activity sets that fulfilled those requirements. The DOE responded by stating that the decision involved a management judgement in addition to the objective analysis. In any case, it appears that the elaborate exercise further detracted the project from conducting activities that would help it demonstrate that the WIPP can be used as a permanent repository.

This is particularly significant because the DOE schedule to submit the compliance application to the EPA has become progressively tighter. According to Revision 2 (October 6, 1995) of the DOE's WIPP Disposal Decision Plan, the Compliance Certification Application will be submitted to the EPA in October 1996. To keep this schedule, the deadline for results of the experiments currently in progress for the "performance assessment" (P.A.) calculations is March 1996, and the P.A. calculations have to be completed by June, 1996. This schedule leaves no time for resolution of the performance assessment issues (see Chaturvedi, Lee, Silva and Neill, 1995) (2), and the certification process of the application may therefore be a difficult one.

OTHER ACTIONS TO BE COMPLETED

In addition to the Compliance (with 40 CFR 191) Certification Application, the WIPP Land Withdrawal Act requires other actions to be completed before TRU waste can be disposed at WIPP. Furthermore, the DOE has to make some decisions (e.g. the design of engineered barriers, shaft and panel seals, etc.) in the process of preparing the application and has to be operationally ready to start emplacing the waste. The following is a brief description of these requirements.

Waste Characterization and Inventory

The Waste Acceptance Criteria (WAC) for the WIPP waste are being changed. The generator sites have to prepare waste for shipment according to these criteria. The inventory of waste meeting the WAC is not yet clear. Because of the lack of waste characterization facilities for remote-handled transuranic (RH-TRU) waste, the RH-TRU waste is not scheduled for shipment until the year 2002 and that projection may be optimistic. Since RH-TRU canisters will be emplaced in the walls of the repository rooms before stacking CH-TRU drums and boxes, initial CH-TRU emplacement will result in lost design space for the RH-TRU.

Performance Assessment

Deficiencies continue to exist in the scenarios for breach that have not yet been analyzed, geohydrologic and waste characteristics data to support the conceptual models selected, quality assurance of data and computer codes, and justification of input parameter values in the absence of experimental data. These deficiencies will have to be resolved to conduct robust performance assessment for an acceptable compliance application.

Operational Readiness

A revised Safety Analysis Report (SAR) is being prepared to document the safety of the operations. Issues remaining to be resolved include: operation of Continuous Air Monitors (CAMs) in a heavy salt dust environment and the degree of reliance on CAMs, operational hazard studies, RH-TRU waste emplacement, and the procedures and practices to ensure radiation protection of the workers and the public.

Section 9 of the WIPP Land Withdrawal Act requires demonstration of compliance with Subpart A of 40 CFR 191 that specifies limits of radiation doses to members of the public during the operational period.

Design and Testing of the Panel and the Shaft Seals

Confirmation of the assumed permeabilities of the seals, used in the performance assessment analyses, has to be demonstrated through tests.

Engineered Barriers

Decisions to incorporate engineered barriers in the WIPP design, as required by the Assurance Requirements of the EPA standards, have yet to

be made. Backfill in the WIPP repository is required by the Consultation and Cooperation Agreement between the DOE and the State of New Mexico.

Plans for Decommissioning and Post-Decommissioning Management of the WIPP Site

These plans are required by the WIPP Land Withdrawal Act (P.L. 102-579, 1992, Sec. 7 (b)(2)).

Acquisition of Federal Oil and Gas Leases

Required by the WIPP Land Withdrawal Act, Sec. 7 (b)(4), unless the EPA Administrator determines that such acquisition is not required.

Comprehensive Recommendations for the Disposal of All TRU Waste

Required by the WIPP Land Withdrawal Act, Sec. 7(b)(5), because the WIPP is intended only for retrievably stored waste and the waste yet to be generated. About 80% of the existing TRU waste is buried in shallow ground at various DOE facilities.

Survey Identifying All TRU Waste Types at All Sites

Required by WIPP Land Withdrawal Act (Sec. 7(b)(6).

Compliance with the RCRA

The TRU waste destined for the WIPP will contain volatile organic compound (VOC) residues absorbed on metal, paper, cloth or other absorbent surfaces. The compounds present in quantities greater than 1 ppm include carbon tetrachloride (CCl_4), methylene chloride (CH_2CCl_2), 1,1,1-trichloroethane and 1,1,2-trichloroethylene (TCE). The WIPP is regulated by Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act (RCRA, 42 USC 6905 et seq) as a miscellaneous land disposal unit under 40 CFR part 264 Subpart X, and must demonstrate under 40 CFR 268.6 that VOCs will not migrate from the site. Absence of migration is demonstrated by a No-Migration Determination (NMD), to be made by the U.S. Environmental Protection Agency (EPA), and by a permit to be issued by the New Mexico Environment Department pursuant to the New Mexico Hazardous and Solid Waste Act. EPA made a No-Migration Determination for the now defunct test phase of the WIPP. Limits allowed by the EPA in the test phase NMD were based on data from headspace measurements made on about 220 drums (9). The test phase NMD conditions will not apply after closure, and potential emissions from the WIPP must therefore be recalculated.

Compliance with other Environmental Laws and Regulations

Section 9 of the WIPP Land Withdrawal Act also requires compliance with the Clean Air Act; the Safe Drinking Water Act; the Toxic Substances Control Act; the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA); and all other applicable federal laws pertaining to public health and safety or the environment.

SUMMARY

Many actions are yet to be completed by the DOE before a decision can be made to use the WIPP facility as a permanent repository for transuranic waste. Until these actions are completed, neither the Environmental Protection Agency nor oversight agencies can judge whether WIPP is safe for waste disposal.

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THE NATIONAL TRU PROGRAM MANAGEMENT OF THE COLD WAR LEGACY

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ABSTRACT

The United States Department of Energy (DOE) Carlsbad Area Office (CAO) has responsibility for developing and implementing the National Transuranic (TRU) Program (NTP). The mission of the NTP is to integrate the national TRU-waste system in order to ensure that all TRU waste under the purview of the DOE is effectively and systematically managed from its generation to its disposal. The mission encompasses the generation/storage/retrieval, characterization, certification, treatment, packaging/transportation, and disposal of TRU waste. Each element of the TRU system has varying degrees of involvement and guidance from the NTP. In keeping with the DOE goal of integration, the NTP is in the process of merging the individual generator site management efforts into one system. The touchstone of this effort is the preparation of the TRU Waste Management Plan. The benefits resulting from the Plan include savings in costs and efforts, consistency in meeting regulatory requirements, widespread use of developed technologies, and acceleration of site cleanup and permanent disposal schedules. The process for preparing the

Plan requires scrutiny of the entire TRU waste management system by means of utilizing a computer model which, by using differing scenarios, will determine the optimum configuration for TRU waste management activities. Effective integration and execution of this strategy will require centralized guidance from the NTP to the generator sites.

The DOE's goal is to eliminate temporary storage of TRU waste and achieve environmentally sound and permanent disposal at the Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico. In order to accomplish this goal, the National TRU system must prepare waste to support the initial opening of WIPP and provide waste for a consistent and efficient operation at the WIPP facility.

INTRODUCTION

For more than 40 years, the mission of the DOE has involved the production of nuclear weapons and conducting research with nuclear materials. These activities have generated significant quantities of radioactive wastes, which are currently stored at generation sites throughout the DOE complex. Over the past decade, national priorities have shifted, and the mission of the DOE now emphasizes environmental cleanup and safe, permanent disposal of radioactive wastes.

The NTP was established in the newly-created CAO in December 1993 to integrate and coordinate TRU waste management activities throughout the DOE complex. The mission of the NTP is to integrate the national TRU waste system to assure that all TRU waste under the purview of the DOE is effectively and systematically managed from its generation to its disposal.

TRANSURANIC WASTE

To ensure effective waste management and disposal, radioactive wastes are divided into four major categories: spent fuel, high-level radioactive waste, low-level radioactive waste, and TRU waste. The focus of this paper is on TRU and TRU-mixed waste. TRU waste is defined as radioactive material contaminated with greater than 100 nanocuries per gram of transuranic elements (elements which have atomic numbers greater than 92), with half-lives greater than 20 years. TRU waste is further designated as either contact-handled (CH) waste or remote-handled (RH) waste, depending on the dose rate at the surface of the waste container. If the dose rate is 200 millirems per hour or less, the waste is designated as CH-TRU; if the dose rate is in excess of 200 millirems per hour, the waste is designated as RH-TRU. TRU-mixed waste is TRU waste that also contains hazardous constituents (1). For the purposes of this paper, the term "TRU waste" includes both TRU and TRU-mixed waste.

Prior to 1970, there was no classification for TRU-contaminated waste. Waste contaminated with TRU radionuclides was commingled with other wastes and disposed of on-site in shallow landfill configurations. These wastes are referred to as "pre-1970 buried" wastes. In 1970, the U.S. Atomic Energy Commission (AEC) concluded that waste containing long-lived alpha-emitting radionuclides should have greater confinement from the environment. Therefore, all TRU waste generated since the early 1970s has been segregated from other waste types and placed in retrievable storage pending shipment and disposal in a geologic repository. These wastes are referred to as "retrievably stored" wastes. Until 1984 the minimum radionuclide concentration level for defining TRU wastes was greater than 10 nanocuries per gram; however, in 1984 the concentration level was revised to greater than 100 nanocuries per gram (1).

THE WASTE ISOLATION PILOT PLANT

In the 1950s, the U.S. Atomic Energy Commission requested the National Academy of Sciences (NAS) to study the permanent disposal of radioactive wastes. The NAS study concluded that disposal in deep geologic salt beds offered the greatest potential for safely isolating TRU waste from the environment. Consequently, the DOE began developing plans for a deep geologic repository. After evaluating several potential disposal sites throughout the country, the DOE selected a site in southeastern New Mexico which met all selection criteria. The WIPP, located 26 miles east of Carlsbad, New Mexico, is the first DOE facility designed to demonstrate the safe disposal of TRU waste in deep salt beds (1).

The WIPP was authorized by the DOE National Security and Military Applications of Nuclear Energy Act of 1980 (Public Law 96-174) for the express purpose of "providing a research and development facility to demonstrate the safe disposal of radioactive waste resulting from the defense activities and programs of the United States exempted from regulation by the Nuclear Regulatory Commission" (2).

On October 30, 1992, the U.S. Congress passed the Waste Isolation Pilot Plant Land Withdrawal Act (LWA) (Public Law 102-759), which transferred control of the federal land on which the WIPP is located from the U.S. Department of the Interior to the DOE. In addition, the LWA established a set of environmental, safety, and health requirements that must be met before waste can be shipped to WIPP, and required the EPA to establish disposal standards for TRU waste (3).

The WIPP facility, scheduled to begin receiving CH-TRU waste in 1998, has a design capacity of 175,600 cubic meters of CH-TRU waste and 7,080 cubic meters of RH-TRU waste. The WIPP disposal horizon is located at a depth of 655 meters below the ground surface. The waste disposal area will consist of eight panels, each of which will contain seven rooms. Each room will have nominal dimensions of 91 meters long, 10.1 meters wide, and 4 meters high. The storage rooms and panels will be excavated in stages, coordinated with the scheduled arrival of waste (1).

NATIONAL TRU PROGRAM

In order to accomplish its mission of ensuring effective, integrated, cradle-to-grave management of TRU wastes throughout the DOE complex, the NTP is developing a National TRU Waste Management Plan. The Plan will provide long-term guidance for the coordination and integration of TRU waste activities throughout the DOE complex. The Plan will also identify specific TRU waste management initiatives and provide a systematic prioritization of the projects needed to meet the principal NTP waste management objective - the permanent disposal of all TRU waste. The benefits of such a plan include savings in preventing duplication of costs and efforts, consistency in meeting regulatory requirements, expanded utilization of developed technologies, and acceleration of sites' cleanup and permanent disposal schedules.The process for developing a comprehensive National TRU Waste Management Plan is comprised of several major activities. The first involved establishing a National TRU Waste Program Management Objective, which is the cornerstone upon which all of the NTP's goals and initiatives are based. This objective is as follows:

In the interest of ensuring the health and safety of the public as well as protecting the environment, the National TRU Program will pursue the most efficient and effective means, in terms of schedule and budget, for the permanent disposal of all TRU waste (4).

The second major activity is to identify initiatives or special studies and associated milestones for each of the six functional elements of the TRU waste system. These elements are the generation/storage/retrieval, characterization, certification, treatment, packaging/transportation, and disposal of TRU waste. Each of the elements in the TRU waste system are discussed in further detail later in this paper.

Development of the National TRU Waste System Model (NTWSM) is the third major activity in developing the NTP Waste Management Plan. The NTWSM is a computer simulation model designed to evaluate the preparation and flow of TRU waste from generation and storage locations in the DOE complex to final disposal at the WIPP. The NTWSM will integrate the program initiatives, evaluate various TRU waste system configurations, and provide information for decision making and program prioritization. The first product of the NTWSM will be the evaluation, documentation, and acceptance of a baseline system configuration scenario. This baseline scenario will describe the current TRU waste system configuration and capabilities and will be used to compare the effectiveness of all other alternative system configuration scenarios. After the baseline scenario is established, multiple alternative scenarios will be evaluated to determine their effectiveness in terms of cost and schedule improvements relative to the baseline. Once optimal configurations are identified, they will be further developed, evaluated, and documented as part of the Management Plan, which will be used to provide complex-wide programmatic configuration recommendations for future fiscal years. Successful implementation of the plan will result in the maximum amount of TRU waste possible being removed from the generator sites and permanently disposed in a safe, cost-effective, efficient manner.

Stakeholder involvement will be incorporated throughout all stages of the development process, including data collection, scenario development, modeling, and post-modeling evaluations. The acceptance and implementation of the National TRU Waste Management Plan will involve many organizations, including the DOE, regulatory agencies, the National Governors Association, the DOE Federal Facilities Compliance Act (FFCAct), Policy Coordination Group (PCG), and a number of other stakeholders.

Almost all of the TRU waste in the DOE complex generated since 1970, and future TRU waste to be generated, are destined for disposal at the WIPP. Therefore, the WIPP facility is a key component of the NTP disposal strategy.

NATIONAL TRU WASTE SYSTEM

To ensure that generator/storage sites throughout the DOE complex are prepared to ship their TRU waste to WIPP in 1998, the NTP has developed initiatives in each of the six functional elements of the National TRU Waste System. Figure 1 illustrates the TRU Waste System Process in terms of these six elements and identifies significant milestones to be accomplished in developing the National TRU Waste Management Plan. A summary of each functional element, along with a description of current initiatives for each element, is provided in the following sections.

Fig. 1

Generation/Storage/Retrieval

The generation of TRU waste has evolved over time as the DOE mission has changed. In the past, DOE's major role was research, development, and production of nuclear weapons, which produced large quantities of TRU waste. Over the past decade that role has decreased and emphasis has been

placed on decontamination and decommissioning (D&D) of DOE facilities as well as the environmental restoration of the DOE nuclear weapons complex. Both of these activities are expected to add large volumes of newly-generated TRU waste to the current inventory.

CH-TRU waste is currently stored in a variety of facilitiesbermed pads, open storage, air support buildings, expandable engineered sprung structures, and conventional buildings. RH-TRU waste storage facilities consist of a mixture of shielded compartments for the remote handling of highly radioactive wastes (hot cells), covered tanks, and concrete culverts.

Storage capacity at the generator sites for CH-TRU waste is currently being built to ensure RCRA-compliant storage. If the opening of WIPP is significantly delayed beyond its 1998 scheduled opening, additional CH-TRU storage facilities will be needed. With regard to RH-TRU waste, all but two generator sites have adequate storage facilities to accommodate their existing and projected waste. The sites which do not have adequate storage capabilities are the Hanford Reservation (Hanford) and Oak Ridge National Laboratory (ORNL). These two sites have the vast majority of RH-TRU waste. ORNL has plans to construct two new facilities which, if completed as scheduled, will adequately meet its future storage needs. Hanford has the largest demand for RH-TRU waste storage. Assessments have identified an unfinished reactor facility as a possible location for storage and treatment of radioactive wastes, but it would require extensive retrofitting. Another option would be to build a facility specifically designed for RH-TRU waste storage.

According to Revision 1 of the Waste Isolation Pilot Plant Transuranic Waste Baseline Inventory Report (WTWBIR), there are approximately 72,598 cubic meters of CH-TRU waste and 1,165 cubic meters of RH-TRU waste currently in storage at the 10 major facilities (large-quantity sites) and more than 20 smaller facilities (small-quantity sites) located throughout the country. Locations of the major facilities are shown in Fig. 2. In addition, the WTWBIR indicates that approximately 50,553 cubic meters of CH-TRU and 3,650 cubic meters of RH-TRU waste will be generated from D&D and environmental restoration activities through the year 2023 (5). Table I lists volumes of stored and projected TRU waste, by site.

Fig. 2

Table I

The NTP has identified several initiatives within this functional element. One is to develop waste minimization programs at all generator sites. In addition, the NTP is preparing revisions of the WTWBIR, which will describe the WIPP inventory by waste stream and identify important waste parameters of both existing and projected waste volumes at each DOE facility. These data will provide important input to the Performance Assessment and Compliance Certification Application, which is the document demonstrating WIPP's compliance with EPA standards. The NTP will also define documentation and characterization requirements for newly-generated waste in order to preclude the need to re-characterize drums of newly-generated waste prior to disposal at WIPP.

Waste Characterization

Developing a complex-wide waste characterization program is an essential part of the NTP mission. This includes identifying waste characterization requirements and developing the required technology and methods to obtain waste characterization and inventory data that support regulatory compliance and systems planning activities. In order to ship waste to the

WIPP, the generator sites will need to characterize the waste to ensure that the WIPP Waste Acceptance Criteria (WAC) are met. The WIPP WAC document identifies and consolidates existing criteria and requirements which regulate the safe handling and preparation of TRU waste packages for transportation to and long-term emplacement in the WIPP (6). Below is a list of the major documents and regulations which form the basis for the WIPP WAC.

WIPP Land Withdrawal Act (Public Law 102-579)
Resource Conservation and Recovery Act
- Part B (40 CFR 264)
- EPA No Migration Determination (40 CFR 268)

WIPP Safety and Analysis Report
TRUPACT Safety and Analysis Report for Packaging
Compliance Certification (40 CFR 191, 40 CFR 194)

The WIPP WAC provides the specific criteria each waste package must satisfy regarding content, characterization, and documentation. Although some revision to the WAC may be necessary due to the EPA's final WIPP certification and the State of New Mexico's RCRA permit for the WIPP, waste characterization can be performed now. To ensure that the generator/storage sites comply with the WIPP's waste characterization requirements, the Carlsbad Area Office prepared a Transuranic Waste Characterization Quality Assurance Program Plan (QAPP) (7). The QAPP requires site-specific sampling plans, describes sample collection and custody procedures, and establishes the quality assurance objectives for each method used to characterize TRU waste. The QAPP requires that CH-TRU waste streams be characterized using acceptable knowledge, radioassay, radiography, headspace gas analysis, visual examination, and destructive analysis. Each generator site must develop a site-specific Quality Assurance Project Plan (QAPjP) which documents how the site will meet each of the required elements of the QAPP.

An NTP characterization initiative which was recently completed was the development of a Waste Characterization Systems Analysis, which evaluated various configurations of equipment and facilities to determine the optimum configuration for meeting TRU waste storage, transportation, and disposal characterization requirements. In addition, the NTP recently developed the Mobile Waste Characterization Systems Analysis. This analysis was an important element of the Waste Characterization Systems Analysis, but was developed separately. The mobile analysis evaluated various levels of mobile characterization capability, ranging from characterizing waste to meet only transportation requirements, to characterizing waste to meet all disposal requirements. The systems analysis determined operational requirements, defined potential alternatives to enhance current fixed characterization facilities, and identified needed characterization capabilities at TRU waste sites which do not have characterization facilities.

The NTP is currently preparing Revision 5 of the WIPP WAC document, which will reflect several significant developments since the previous WAC was last issued, such as the creation of the Carlsbad Area Office, the elimination of the previously planned Test Phase, and the submittal of several permit and compliance applications. Feasibility studies at Hanford and ORNL for alternate characterization programs and facilities, including privatization, have also been conducted.

Certification

Waste characterization activities and data are subject to certification to meet the acceptance criteria of various regulatory agencies. A key component of certification activities are the various Performance Demonstration Programs (PDPs) which assess and approve the performance of analytical facilities supplying TRU waste characterization services by applying a series of recurring performance tests to specific analytic techniques.

Major certification initiatives of the NTP include the development of PDPs for headspace gas sampling of waste drums, radioassay of waste drums, and chemical analysis of sludges, and all with periodic laboratory re-certifications. Certification initiatives also include conducting site visits, surveillances, and audits at all DOE sites that have TRU waste characterization programs to verify compliance with the WIPP WAC, QAPP, and site QAPjP.

Treatment

Treatment processes are designed to physically, chemically, or radiologically alter waste forms in order to comply with specific waste acceptance standards. The FFCAct required the Secretary of Energy to submit Site Treatment Plans (STPs) for the development of treatment capacity and technologies for treating mixed waste for each facility at which DOE stores or generates these wastes. The DOE submitted Final STPs to the EPA or, where applicable, the state regulating agency; and, by October 1995, Compliance Agreements or Consent Orders had been approved or issued for most sites.

In addition, the Carlsbad Area Office conducted a technology study to fulfill the requirements of Section 19 of the LWA, which stipulates:

> Within three years after the date of this Act, the [DOE] Secretary shall submit to Congress a study reviewing the technologies that are available and that are being developed for the processing or reduction of volumes of radioactive wastes. The study shall include an identification of technologies involving the use of chemical, physical, and thermal (including plasma) processing techniques (3).

In response to this requirement, the Radioactive Waste Processing and Volume Reduction Technology, issued in October 1995, presents background information and summaries of 35 categories of waste processing technologies. Within the 35 categories, 219 processes were reviewed. Of the processes reviewed, the vast majority apply to hazardous waste or to the hazardous component of mixed waste. Few processes presently exist at the commercial level for processing radioactive waste. Many processes that are in the conceptual or pilot stage may apply to radioactive waste; however, their availability for future use is uncertain. Development of these potential applications will depend on research funding, potential cost effectiveness relative to other technologies, and their ability to comply with current and future environmental regulations (8).

The current DOE treatment strategy consists of implementing the treatment necessary to meet the WIPP WAC. However, for waste not designated for disposal at the WIPP, treatment to other standards may be necessary. Initiatives within this element included treatment feasibility studies which examined the benefit and cost effectiveness of treatment processes provided by private companies for TRU and alpha-low-level waste at the Idaho National Engineering Laboratory (INEL) and a treatment/characterization comparison study to determine if the INEL treatment privatization feasibility study results can be extrapolated and applied to other facilities within the TRU waste complex. In addition,

data collected from the Mobile Waste Characterization Systems Analysis and the INEL feasibility study will be combined to determine the efficiency of using mobile characterization units to characterize waste for transportation to a potential privately-owned treatment facility in Idaho or at regional treatment facilities across the DOE complex (9).

Packaging/Transportation

The packaging and transportation element consists of the systems and equipment necessary to safely transport TRU waste between facilities within and across state boundaries. Packaging and transportation capabilities must meet NRC, Department of Transportation (DOT), and other applicable regulations. In addition, transportation systems supporting the WIPP have other requirements such as emergency response and generator site training to assure the health and safety of the public and the environment. A certified Transuranic Package Transporter (TRUPACT-II) and related mobile loading equipment are presently available.

The TRUPACT-II is an NRC-approved Type B packaging for the transportation of CH-TRU waste within the United States. The TRUPACT-II has a maximum payload capacity of 2,835 kilograms, including pallets, slip sheets, and waste packed in either 14 208-liter drums, two 1.9 cubic-meter standard waste boxes, or one ten-drum overpack (1). The DOE currently owns 15 certified TRUPACT-IIs. The current packaging design for RH-TRU waste is the RH-72B Cask. The maximum payload capacity of each RH-72B is 3,629 kilograms. The payload will consist of RH-TRU waste in 114- or 208-liter drums contained in a canister. The canister is a DOT 7A Type A carbon steel single-shell container, with an outer diameter of approximately 66 centimeters and overall length of 3.1 meters. Each canister is capable of transporting three 208-liter waste drums (1). The RH-72B Cask has been designed to meet NRC Type B requirements. An application has been submitted to DOE-Headquarters which will, in turn, forward the application to the NRC for a Certificate of Compliance for the RH-72B cask.

One of the NTP packaging and transportation initiatives is to find ways of expanding the TRUPACT-II Envelope of Performance by conducting tests to allow for an increased fissile gram loading limit and an increased wattage limit, which is related to the isotopic activity and gas generation rates in the TRUPACT-II during transport. The NTP is also finalizing the CH-TRU Package Optimization Study, consisting of a cost/benefit analysis of developing a new CH package for waste configurations which cannot meet TRUPACT-II requirements versus retaining the TRUPACT-II and thereby requiring some sites to repackage the waste to meet current TRUPACT-II limits. Initiatives for RH-TRU waste include completing the final certification process for the RH-72B Cask and investigating alternate packaging configurations.

Disposal

Disposal consists of the permanent, non-retrievable disposition of all DOE TRU waste. The initial disposal priority is to use the WIPP facility for TRU waste generated after 1970, the date retrievable interim storage began. Disposal recommendations for the waste volumes which are not currently designated for disposal at the WIPP are being developed in conjunction with WIPP disposal operation plans. In addition, the DOE has management responsibility for the limited volumes of TRU waste at numerous smaller facilities which can be consolidated to minimize the disproportionate waste management costs prior to disposal.

The DOE CAO is in the process of developing comprehensive disposal recommendations to be submitted to Congress pursuant to LWA requirements. The major activities necessary to support WIPP's opening are identified in the WIPP Disposal Decision Plan (DDP), which provides milestones, deliverables, and coordination points for all WIPP activities. Revision 1 of the DDP, issued October 6, 1995, shows CH disposal operations beginning in April 1998; RH operations are planned to begin in 2002.

A major initiative supporting the disposal system element is to complete the Comprehensive Disposal Recommendation Study required by the LWA. Included in the study is the portion of waste which is not currently designated for disposal at the WIPP, such as any which may exceed the WIPP capacity, pre-1970 buried waste, and non-defense waste. In addition, the NTP recently completed the RH-TRU Waste Disposal Strategy document, which established the management and disposal strategy for RH-TRU waste and satisfied a WIPP DDP milestone. The NTP is also conducting an RH-TRU Disposal Alternatives Study Assessment, which will analyze RH-TRU waste disposal alternatives for the WIPP and will make recommendations for the most cost-, schedule- and safety-effective alternatives.

CONCLUSION

Since the NTP was established in the CAO in late 1993, it has made extensive progress toward accomplishing its goals. However, a significant number of activities remain to be completed. The most important is the completion of the National TRU Waste Management Plan. This Plan will establish what needs to be done at each of the generator sites as well as establish a schedule for completing the tasks. The Plan will identify initiatives and alternatives that will vary input data into the system model. The result will be the best combination of scenarios to achieve the most efficient national system and provide the basis for resource allocation, consistent with DOE commitments (i.e., compliance agreements with states). Given the current and future resource constraints, it is imperative that the diverse TRU waste programs across the DOE system be focused toward a consistent, integrated, and long-term national management program. Once completed, the Management Plan will provide the necessary guidance and prioritization for ensuring an integrated and effective TRU waste management program consistent with the ultimate objective for the safe and efficient management and disposal of all TRU waste.

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WIPP WAC REV. 5 APPLICABILITY
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ABSTRACT
The Department of Energy (DOE) is preparing for disposal operations at the Waste Isolation Pilot Plant (WIPP) in 1998. WIPP is a deep geological repository designed for the safe and efficient disposal of transuranic (TRU) wastes. The Waste Acceptance Criteria (WAC) for WIPP were initially developed by a DOE steering committee in 1980. Revision 5 reflects the latest negotiations and permit requirements from the Environmental Protection Agency (EPA), the State of New Mexico Environment Department (NMED), and the Nuclear Regulatory Commission (NRC). The regulatory requirements are combined with the requirements derived from the WIPP safety analysis performed for disposal operations and the original criteria established for safe waste handling operations.
The WIPP WAC provides a comprehensive overview of the requirements and basis for developing waste acceptance criteria to meet today's rules and regulations for transportation and disposal of TRU wastes. The authors believe that it is a comprehensive criteria and a guidance manual for generator/storage sites who must characterize and certify TRU waste for disposal at WIPP. It also provides valuable insight to future projects that may develop their own waste acceptance criteria.
The WIPP WAC presents the requirements from the following sources:
1) Resource Conservation and Recovery Act (RCRA) Permit Application
2) Land Disposal No-migration Variance Petition
3) 40 CFR 191 Draft Compliance Certification Application
4) Certificate of Compliance (C of C) from the NRC for a Type B shipping container
5) Federal Land Withdrawal Act for WIPP
6) WIPP Safety Analysis Report
7) WIPP System Design Descriptions (SDDs)
The WIPP WAC combines operations and nuclear safety requirements with transportation and hazardous waste regulatory requirements to provide a comprehensive set of criteria and requirements that ensure the safe disposal of TRU waste.
INTRODUCTION
In 1978 and 1979, the WAC for the Waste Isolation Pilot Plant were initially developed by a U.S. Department of Energy Steering Committee. The WAC Steering Committee generated a May 1980 report, DOE/WIPP-069, titled "Report of the Steering Committee on TRU Waste Acceptance Criteria

for the Waste Isolation Pilot Plant." The purpose of the original WAC was to provide performance requirements to ensure public health and safety as well as the safe handling of TRU waste at WIPP.

A series of revisions consistent with the intent of the original DOE/WIPP-069 were subsequently published to incorporate the results of ongoing project activities and the comments and suggestions from others. Revision 1, September 1981, reflects consultations between the Environmental Evaluation Group (EEG) and the Albuquerque Operations Office WIPP Project. Revision 2, September 1985, reflected continued interactions with the EEG and other TRU program participants. Revision 3, January 1989, incorporated other requirements such as those in the Resource Conservation and Recovery Act (RCRA) and the TRansUranic PACkage Transporter-II (TRUPACT-II) Certificate of Compliance (C of C) from the U.S. Nuclear Regulatory Commission (NRC). Revision 4, December 1991, added specific requirements for the Test Phase. Revision 5 of the WAC reflects the organizational restructuring of the DOE, deletes the Test Phase requirements and updates other requirements instituted since the issuance of Rev. 4.

DOCUMENT LAYOUT

WAC Rev 5 defines current criteria and requirements for characterization, certification and acceptance of TRU and TRU mixed waste at the WIPP. The derivation of the criteria defined in this WAC is shown in Fig. 1.

Fig. 1

As depicted by Fig. 1 the WAC now has its basis or justification in a new set of documents that were not previously published. Because the WAC criteria can now be referenced to WIPP-specific requirements documents the justification section of each criterion was deleted from the current revision.

The criteria and associated requirements needed to ensure safety and compliance are listed in Table I "TRU Waste Acceptance Criteria Summary". The "CRITERIA" column lists the individual criterion addressed in the WAC and the "REQUIREMENTS" columns list the limits or controls applied to each criterion.

Table I

In the WAC document the criteria and requirements for waste certification and acceptance are subdivided into 3 areas, WIPP Operations and Safety Requirements, Transportation Requirements, and Environmental Compliance Requirements. Each subsection lists its own requirements for each criterion. The WIPP Operations and Safety Requirements ensures TRU waste is received and disposed in a manner which protects the public and WIPP personnel. The requirements that limit criteria under this area are derived from the WIPP Safety Analysis Report (SAR) and the WIPP System Design Description (SDD) for waste handling. The Transportation Requirements ensure no release of the payload contents from the packaging, if and when subjected to extreme accident abuse. The requirements under this area are derived from the TRUPACT-II and draft RH 72B Cask Safety Analysis Reports for Packaging (SARPs). Environmental Compliance Requirements ensure that wastes comply with the limits set by the Environmental Protection Agency (EPA), New Mexico Environmental Department (NMED) and Congress. The requirements under this area are promulgated in several documents. These documents are the:

- WIPP RCRA Permit Application
- draft No-Migration Variance Petition
- Waste Isolation Pilot Plant Land Withdrawal Act, Public Law 102-579

It is the generators responsibility to ensure that the waste meets all of the WAC requirements. For ease of understanding and displaying the criteria they were grouped into five categories. The categories are:

Container and Physical Properties
Nuclear Properties
Chemical Properties
Gas Generation, and
Data

The WAC requirements have in the past caused some confusion for generator sites because of the number of requirements required for each criterion and the concept of most limiting condition. In WAC, Rev 5, the requirements have been combined in a summary table which lists the minimum requirements which must be met to demonstrate full compliance with each criterion. To further help the generator/storage site a compliance description section is written for each criterion. The compliance section describes the activities or actions that must be taken to demonstrate full compliance with each criterion. In addition to the organization of the criteria and requirements, the contact-handled (CH) criteria are presented separately from the remote-handled (RH) criteria.

TECHNICAL CONTENTS

WAC, Rev 5 has several technical changes from the Rev 4 version. Some of the more extensive changes are as follows:

The criterion for particulate size and quantity was deleted. This criterion was based on reducing the severity of a dropped drum accident. The latest analysis in the WIPP SAR concluded that accident results are not dependent upon the amount of particulate in the waste form. Neither the TRUPACT-II SARP nor RCRA addressed particulate in the waste, so the criterion was discontinued.

The requirements for liquids was rewritten to clarify the position on free liquids detected in the payload containers. The criteria now clearly states that it is acceptable to have up to 2 liters of liquid total in a 55 gallon drum (~1% of the drum volume). This is not a change from Rev 4, but a clarification of what has always been the intent of the criterion

The ten year life of the bar code labels on drums has been deleted. The requirement for WIPP operations is that it is readable at the time of receipt at WIPP.

The criterion for removable surface contamination on payload containers has been revised to bring it in line with the DOE Rad Con Manual. This is actually a reduction in allowable surface contamination from the Rev 4 requirements. It is not expected that this lower requirement will impose a hard ship on generators because the limits are the same as are currently required by the DOE Rad Con Manual.

The requirements for the payload container (i.e. 55-gallon drum, etc.) has been rewritten to reflect the current configuration of WIPP waste handling operations equipment, the TRUPACT-II and what has been described in the RCRA permit application. There are numerous other payload configurations, but currently, the only containers that can be characterized, shipped in a TRUPACT-II or RH 72B cask and have been described in the permit applications are the DOT Type A 55-gallon drum, TRUPACT-II Standard waste Box (SWB) and the RH canister described in the RH 72B Cask SARP.

The CH TRU drum loading limit for acceptance at the WIPP was reduced from 1000 to 80 Plutonium Equivalent Curies (PE-Ci). The allowable drum

loading activity of 80 PE-Ci resulted from a resent reanalysis of the TRU waste inventory and calculated off-site dose form postulated waste handling accidents. The new drum loading limit of 80 PE-Ci was determined using acceptance criteria for off-site doses to the general public at the WIPP exclusion area boundary rather than at the WIPP Land Withdrawal boundary. Drums exceeding 80 PE-Ci may still be accepted for disposal at the WIPP; however, additional analysis will be required to determine if additional safe guards are necessary to protect the public, workers and environment.

New requirements limiting the levels of drum headspace VOC concentrations was added. The new limits are based on Back-Calculations from the Health-based levels in the draft No-Migration Variance Petition.

New requirements for waste characterization and documenting the characterization techniques was added to the data requirements. This caused a new form called the Waste Stream Profile Form to added to the WAC. The new form will provide WIPP with documentation of actions taken to characterize waste for disposal in the WIPP. This completed form and selected reports from the WIPP waste information system will be part of the records for waste characterization that will be kept at the WIPP.

CERTIFICATION

WAC, Rev 5 furnishes guidance for the preparation of deliverables necessary for implementation of site and waste certification. Site certification and waste certification are imposed on TRU waste generators in lieu of sampling waste at the WIPP. This is because sampling waste at WIPP is neither practical nor economical. Site certification involves generator sites developing written programs that ensure wastes are properly characterized and certified to the requirements required for transportation and disposal of TRU wastes at the WIPP. Site certification is the written acknowledgment by the DOE/Carlsbad Area Office (CAO) that a generator site has the capability to meet the requirements of the TRU Waste Characterization Quality Assurance Program Plan (QAPP) and the WIPP WAC. The CAO ensures these programs are in place and functioning properly by reviewing and approving key generator site documents and performing periodic (annual) audits of the generator sites for evidence that the programs are functioning as described in the site specific documents. Waste certification involves determining and documenting that the waste meets the requirements of the QAPP and the WAC. Waste Certification is the generator sites written assurance that waste containers shipped to and disposed of in the WIPP meet the requirements for transportation and disposal. The WIPP periodically reviews characterization, certification and shipping data maintained by generator sites to verify compliance. Generator/storage sites (sites) must characterize their waste on a waste stream basis to site specific and WIPP approved plans. After characterization of individual payload containers a TRU waste data package is transmitted to the WIPP via the WIPP Waste Information System (WWIS). WIPP personnel review the data package for completeness and acceptability and provide appropriate notification to the Site. When sufficient data from a particular waste stream have been submitted, the site prepares a summary of the waste stream information and reconciliation with the Data Quality Objectives (DQOs) defined in the TRU Waste Characterization Quality Assurance Program Plan (QAPP). This summary is compiled in a Waste Stream Profile Form. The form is completed by the generator/storage site and approved and maintained on file by the WIPP for acceptance information on future waste shipments waste. Waste in

payload containers from approved waste streams are certified for disposal in the WIPP in accordance with the WIPP WAC. When enough payload containers have been assembled to form a shipment they are certified for shipment in the TRUPACT-II for the CH wastes or the RH 72B Cask for RH wastes. The disposal and shipping certification data is transmitted to the WIPP for approval using the WWIS.

SUMMARY

The WIPP WAC is a comprehensive document containing the latest criteria and requirements for TRU mixed waste transportation and disposal. The WAC provides comprehensive operational safety, transportation and RCRA requirements presented in a simple user friendly format. In addition to the waste acceptance criteria and requirements, generator/storage site certification, characterization and data reporting requirements are established.

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STORED TRANSURANIC WASTE MANAGEMENT PROGRAM AT THE IDAHO NATIONAL ENGINEERING LABORATORY*

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ABSTRACT

Since 1970, the Idaho National Engineering Laboratory (INEL) has provided interim storage capacity for transuranic (TRU)-contaminated wastes generated by activities supporting U.S.national defense needs. Approximately 60% of the nation's current inventory of TRU-contaminated waste is stored at the INEL, awaiting opening of the Waste Isolation Pilot Plant (WIPP), the designated federal repository for permanent disposal of defense-generated TRU waste. A number of activities are currently underway to enhancing current management capabilities, conducting projects that support local and national TRU management activities, and preparing for production-level waste retrieval, characterization, examination, certification, and shipment of untreated TRU waste to WIPP in April 1998. Implementation of treatment capability is planned in 2003 to achieve disposal of all stored TRU-contaminated

waste by a target date of December 31, 2015, but no later than December 31, 2018.

INTRODUCTION

Transuranic waste is defined (1) as material that has negligible economic value and is contaminated with alpha-emitting radionuclides with an atomic number greater than 92 and a half-life greater than 20 years, and in concentrations greater than 100 nCi/g. Since 1970, the INEL has placed approximately 60% of the nation's contact-handled (CH) TRU-contaminated waste in interim retrievable storage pending establishment of a permanent disposal repository. Approximately 65,000 m3 of TRU-contaminated waste contained in about 129,000 drums and 11,000 boxes and bins are in storage at the INEL Radioactive Waste Management Complex (RWMC). These wastes are stored either in air-supported or pre-engineered metal buildings, or on earthen- or tarpaulin-covered asphalt pads. The total stored TRU mass is 823 kg with 500,000 Ci of activity.

This paper will discuss ongoing activities to: 1) enhance INEL TRU waste management capabilities, 2) conduct technical support projects supporting both local and national needs, 3) implementing full treatment of TRU-contaminated waste at INEL,and 4) identify future strategic direction.

BACKGROUND

Since the early 1980s, the INEL has been on a path to evaluate and improve TRU waste management capabilities. Characterization of various aged waste was performed in 1980-1985 to support development of real-time radiography (RTR) as a nondestructive waste certification technique, address issues related to radiolytic production of hydrogen in waste drums, and support development of the waste certification process (2,3).

The INEL Stored Waste Examination Pilot Plant (SWEPP) was constructed and placed into operation in August 1985. The SWEPP facility provides capability to weigh and nondestructively examine the contents of waste containers for compliance with WIPP Waste Acceptance Criteria (WAC) without container opening, to perform nondestructive assay (NDA) for determining fissile material content and total TRU activity of a waste package, and to perform ultrasonic examination of drums to verify metal wall thickness to meet transportation requirements. The SWEPP facility was placed in operational standby in late 1989 due to delays in opening WIPP and promulgation of final WAC. In 1991, limited operations were initiated to support the WIPP Experimental Test Program and waste characterization activities.

The INEL Drum Venting Facility (DVF) was designed, constructed and placed into operation in May 1987. The purpose of this facility is to remotely puncture a drum and install a carbon composite filter to allow pressure equalization and aspiration of radiolytic-produced hydrogen. Drums destined for WIPP will be vented and the headspace gas sampled. Vented packages are required for transportation if the Transuranic Package Transporter-II (TRUPACT-II) is used as the shipping container.

The INEL TRUPACT-II Loading Facility (TLF) was designed and constructed in October 1988. The facility was brought to operational status in November 1990 to provide shipments of TRU-contaminated waste to Argonne National Laboratory-West (ANL-W) to support preparation of experimental bins for the underground radioactive tests currently at WIPP. Currently, shipments are made for characterizing waste to support WIPP efforts to demonstrate compliance with regulations governing disposal of TRU mixed waste. Use of the TRUPACT-II was implemented in April 1992. Over 35

shipments using the TRUPACT-II have been completed to support characterization efforts.

PROGRESS TOWARDS READINESS FOR DISPOSAL

For the past several years, the INEL has initiated projects to enhance operational capabilities to manage the stored TRU-contaminated waste, to evaluate and upgrade existing waste certification systems to meet new quality assurance objectives for waste characterization, and to conduct technical support activities that meet INEL, U.S. Department of Energy (DOE)-Carlsbad Area Office (CAO) National TRU Program (NTP), and WIPP needs.

Facility Enhancements

In 1989, two Line-Item Construction Projects (LICP), consisting of several subprojects, were initiated to enhance the existing facility infrastructure and increase operational capability to support waste storage, retrieval, and characterization needs. Figure 1 provides an overview of key construction activities.

Fig. 1

Construction has been completed on eight new storage buildings. A Type I Storage Module was constructed over the existing DVF and primarily supports activities performed prior to waste examination at SWEPP. These activities include a warming area for waste containers, drum venting operations, and an aspiration area used to store vented drums for up to eight weeks to aspirate hydrogen. This storage module is a 3790 m2 pre-engineered metal building, 1340 m2 of which is heated to support container warming and aspiration operations.

Seven Type II Storage Modules have been constructed to support waste storage operations. The modules were designed to comply with Resource Conservation and Recovery Act (RCRA) requirements for permitted storage. The modules are approximately 2675 m2 and will contain about 16,000 drums or 2000 waste boxes. The modules are pre-engineered metal buildings with Type K concrete floors and 6-inch berms to meet RCRA containment requirements. The modules are unheated and include a forced-air ventilation system. Two storage modules are currently operational and are receiving waste from air-supported buildings to meet a 1992 Consent Order with the State of Idaho on storage of TRU-contaminated mixed waste. The remaining five modules will be operational by March 1996. The storage modules will be operated in accordance with the RWMC RCRA Part B permit issued by the State of Idaho in November 1995. One storage module will be used only for storage of environmental-remediation-derived waste from activities being performed at the RWMC on pre-1970 buried TRU-contaminated waste.

An Operations Control Building (OCB) was constructed and placed into operation in April 1995. The OCB is a single story pre-engineered metal building approximately 2230 m2. The OCB provides main access control to the RWMC, a changing area for operations personnel, office space for managerial and technical staff, a personnel training area, and conference rooms. The OCB will also serve as the Emergency Control Center for ensuring rapid and efficient response to abnormal operating situations.

Several site improvement projects have been completed to support new facilities. These improvements have included upgrades to the power supply line to meet anticipated power demands, providing fire and potable water to new facilities, a new sewer system and treatment lagoon, and a new communications/alarm system to support the additional facilities.

A Retrieval Enclosure (RE) is currently being constructed over TRU-contaminated wastes stored on asphalt pads under earthen- or tarpaulin-cover. The purpose of the RE is to reduce weather-related degradation of stored waste containers by providing an enclosure over the storage pads to allow year-around retrieval of waste containers and to provide the necessary utilities and systems to retrieve stored waste. The RE, which is weather-tight, has a nominal free span of 366 m by 61 m for one pad, and 146 m by 49 m for the second pad. The total area enclosed is about 29,170 m2. The retrieval work area will be defined by two moveable partitions or shrouds to provide ventilation control and maintain a negative pressure with respect to outside pressure. Ventilation system exhaust air is filtered by a baghouse and High-Efficiency Particulate Air (HEPA) filtration system. The shrouds support selected utilities needed in the retrieval area such as ventilation ducts, lighting, and electrical connections. Heating of the active work space is provided for year-around operation. A dry-pipe fire protection system is used. An oil-free breathing air distribution system is provided to support off-normal retrieval activities, such as retrieval of a breached container. Equipment used to support retrieval operations will include a mobile vacuum system for cleaning off soil overburden on waste containers, forklifts, an excavator called T-Rex with interchangeable end-effectors for removal of soil overburden and retrieving waste containers, and an isolation unit for use when dealing with contaminated containers. Construction of the RE will be completed by July 1996.

A Waste Characterization Facility has been designed to provide an isolated and controlled environment for opening, examining, and sampling the contents of waste containers. These activities will be performed to meet RCRA characterization requirements specified by final WIPP disposal criteria and for storage by the RWMC RCRA Part B permit. Features of the WCF include capabilities for opening drums and boxes in a glovebox environment, coring capability for sampling sludges and other solidified process residues, remote waste sorting, and repackaging capability. Other functions that WCF could perform include repackaging waste that does not meet WIPP WAC into a form that can be shipped to WIPP for disposal, providing characterization of waste forms that will be treated, and providing limited-scale treatment (e.g., solidification, amalgamation) of certain waste forms. Title II design of the WCF was completed April 1994. Initiation of construction has been deferred until decisions concerning privatization of characterization and treatment services are completed.

System Upgrades

Upgrades to several key existing systems are being made to either improve the operational capability, improve efficiencies, or implement new requirements defined in the TRU Waste Characterization Quality Assurance Program Plan (QAPP) (4).

The DVF is currently being modified to alleviate operational constraints imposed to ensure organic emissions do not exceed allowable limits, and to improve efficiencies and reduce costs associated with performing drum headspace gas analysis to determine the concentration of volatile organic compounds (VOCs)to meet WIPP WAC and transportation requirements. A VOC recovery unit consisting of an activated carbon absorption system has been installed to reduce organic emissions. A headspace gas sampling system has been designed and installed to automatically collect a gas sample during remote venting operations. A Fourier transform infrared (FTIR) spectrometer, developed to reduce waste characterization

analytical costs for determining drum headspace VOCs, will provide automated analysis of samples from each drum. A Residual Gas Analyzer has also been installed to determine hydrogen and methane concentrations in the headspace gas sample. The results of these analyses will be used to determine compliance with VOC concentration limits imposed by final WIPP WAC and the presence of flammable gases exceeding allowable facility safety or transportation limits.

Upgrades to the existing SWEPP NDA systems have been ongoing for several years. In 1993, use of high-resolution passive gamma spectroscopy was implemented to support passive-active neutron (PAN) assay measurements. Addition of the gamma spectroscopy system was necessary to determine the relative mass ratios of various isotopes contained in the nuclear source material in the waste. Although the primary contaminant is weapons-grade plutonium, other isotopes such as U-235 and AM-241 have been identified and must be accurately reported to meet waste characterization and transportation criteria. Current efforts are focused on upgrading hardware and software to improve data collection, data analysis, and operation. The PAN system electronics are being replaced with state-of-the-art electronics, such as shift registers. The improved shift register electronics will allow higher count rate data collection needed for wastes containing high (a, n) reactions. Software modifications are being developed to integrate the PAN and gamma spectroscopy systems to provide quantities of isotopes that must be reported, self-diagnostic checks to flag data inconsistencies and reduce manual review of data, and more user-friendly software for technicians operating the system.

Technical Support Projects

Several key technical support projects are being performed to implement new TRU Waste Characterization QAPP requirements to support future production-level waste examination and certification activities and ongoing NTP projects.

The Waste Certification Program is currently undergoing revision to ensure all aspects of the TRU Waste Characterization QAPP have been addressed and incorporated. Logic diagrams or process flow sheets depicting the cradle-to-grave waste management process have been developed, and activities and data that must be performed or collected to meet QAPP and final WIPP WAC have been built into the flow sheets. These flow sheets will form the basis for determining tasks that must be completed to effectively implement all requirements that allow certification and shipment of waste to WIPP.Evaluation of SWEPP NDA system performance and demonstrating compliance with TRU Waste Characterization QAPP requirements are among the most challenging and difficult aspects of future efforts to certify waste for shipment to WIPP. Calibration drums representing several major stored waste streams have been developed, nuclear sources fabricated to meet stringent quality assurance requirements are being procured, and programs are underway to demonstrate NDA system compliance with precision, accuracy, and minimum detectable concentration quality assurance objectives. Total uncertainty, which is the propagated uncertainty of all corrections and factors used by the NDA system during assay, is being performed on a waste form basis. Because of the wide variation in stored TRU-contaminated waste forms and the resulting affect on NDA system response, it is not economically practical to fabricate calibration standards to experimentally assess total uncertainty. Two approaches have been developed to address total uncertainty. One approach uses a neutron transport and temporal response

calculational method based on the Monte Carlo Neutron Photon (MCNP) code. Total uncertainty evaluations have been completed for graphite molds, and are underway for combustible, glass, and sludge waste forms. For solidified process waste forms, such as sludges, a statistical sampling method is being developed. During coring and sampling of sludge waste forms for RCRA analysis, representative samples will also be collected for destructive radiochemical analysis to quantify radioisotopes in the waste. The results from radiochemical analysis and the SWEPP NDA measurements for each of the drums sampled will be used to complete the total measurement uncertainty evaluation.

The Gas Generation Test Program consists of performing controlled tests with actual containers of CH-TRU waste to collect data and quantify the gas generation properties of the waste under simulated transportation thermal conditions. These tests will be performed due to concerns that hydrogen and methane concentrations in shipping containers destined for permanent storage could exceed their flammability limits. Waste to be tested will be placed in one of five heated enclosures which simulate temperature increases due to decay heating, warmed to 57C - 63C, and held until steady state is attained. An offgas collection system is attached to the carbon composite vent filter installed in the waste drum so that total gas flow may be measured, and samples can be directed to a process mass spectrometer for analysis. The mass spectrometer measures the concentrations of hydrogen and methane in the offgas at frequent intervals during the test. Figure 2 is an internal view of the gas generation test assemblies. The Gas Generation Test assembly is in the final stages of performance testing now, and will be operational in February 1996. The Gas Generation Test Program will also support the Matrix Depletion Program by measuring the concentrations of hydrogen and methane to verify that the rate of gas generation decreases as gas producing matrices (e.g.,combustibles, organic sludges) are depleted.

NTP Support Projects

Since 1990, the INEL has been actively involved with DOE-CAO efforts to demonstrate WIPP compliance with disposal standards and supporting NTP Office objectives. Key technical support projects have included: 1) development of the TRU Waste Characterization QAPP and Performance Demonstration Programs (PDPs) supporting characterization of TRU and TRU-mixed wastes, 2) characterization of stored TRU-contaminated waste to support WIPP data needs for regulatory compliance documents, 3) development of advanced waste characterization methods to reduce waste generator/storage site costs, 4) development of a method to determine internal waste package VOC concentrations without intrusive sampling, 5) developing a system-wide computer simulation model to support national strategic planning efforts, and (6) developing a program to reduce the impact of current transportation wattage restrictions on transportation by analyzing and modeling depletion of gas-producing waste matrices.

Data quality objectives have been developed to provide a uniform and complete suite of requirements that must be met prior to disposing of TRU waste at WIPP. The objectives, which are compiled in the TRU Waste Characterization QAPP, support ongoing efforts at WIPP to demonstrate compliance with regulations governing transportation and disposal of TRU waste. The QAPP has been reviewed by the external DOE agencies that have regulatory authority over WIPP. The intent of the QAPP is to provide generator/storage sites with a single document that identifies characterization requirements that will provide data of known quality to

meet various regulatory compliance needs. Additionally, specific guidance for sampling and analysis methods to be used during TRU waste characterization processes was developed and documented in a guidance manual (5). Several PDPs have been developed to provide an independent method for evaluating and ensuring a waste generator/storage site method for characterizing TRU waste meets quality assurance objectives established in the QAPP. To date, three PDP programs have been developed for DOE-CAO: 1) analysis of VOCs and gases (6), 2) nondestructive assay (7), and 3) RCRA constituent analysis of solidified wastes (8). Waste generator/storage site implementation of these requirements is a key step in achieving authorization to dispose of TRU waste at WIPP.

Fig. 2

Since 1991, the INEL has been actively retrieving and characterizing accessible stored TRU waste forms to supply information primarily supporting development of the No-Migration Petition for WIPP. Characterization activities, at a minimum, include headspace gas sampling, weighing, RTR, and radioassay. Intrusive sampling, completed to meet visual examination, inner bag sampling or collection of sludge samples for RCRA analysis,is performed at the ANL-W Waste Characterization Area (WCA). The WCA provides capability to open and examine waste drums and collect samples in a glovebox environment. Wastes are transported to ANL-W using the TRUPACT-II shipping container. Analysis of sludge samples for RCRA constituents is being performed by Oak Ridge National Laboratory. Quality Assurance Project Plans have been prepared to implement the requirements of the TRU Waste Characterization QAPP. By January 1996, 595 drums have been characterized and the results reported to DOE-CAO.

FTIR spectroscopy was evaluated as an alternative approach to analyzing headspace gas from TRU waste containers. Current practices involve collection of the gas sample in a SUMMATM canister followed by analysis of the gas at the INEL Environmental Chemistry Laboratory using gas chromatography/mass spectrometry. Use of the FTIR method offered a simple, rapid analysis that could be performed at the location the sample was collected. Significant cost savings could be realized over the conventional sampling approach. A two-phase study (9) was completed over about three years to evaluate FTIR capability to simultaneously quantitate the 29 VOCs required by the QAPP and then evaluate application of the FTIR on actual TRU waste drum headspace gas samples. Results of this effort have been successful, with FTIR meeting the TRU waste characterization requirements of 30% accuracy and 25% precision. As previously noted, this system is being installed on the Drum Venting system to support venting, sampling, and analysis needs.

A test program (10) has been completed to demonstrate that VOC concentrations in the void space of each layer of confinement in vented waste drums can be estimated using the measured drum headspace gas and a model incorporating theoretical diffusion and permeation transport principles. The conditional No-Migration Determination for WIPP requires that gas sampling of all layers of confinement within a container for flammable and nonflammable volatile hazardous constituents be performed unless DOE demonstrates this sampling is not necessary. A VOC transport model was developed and experiments completed to measure VOC concentrations in laboratory-scale simulated waste drums and actual waste drums. Both transient and steady-state modeling were completed. Results of this test program provided information that has demonstrated that

headspace gas sampling of inner layers of confinement is not necessary to characterize VOC concentrations in the void volumes of drums and to safely manage waste at WIPP. This provides a significant reduction in costs by not opening and sampling inner bags of waste. A position paper has been prepared and submitted to DOE for regulatory compliance discussions.

A National TRU Waste Model was developed by INEL to support NTP planning efforts. The model, developed using commercially available personal computer software, is a decision-making tool for evaluating the preparation and flow of CH-TRU waste from eight major storage sites to WIPP. The model is designed to predict the course and results of hypothetical actions, understand observed or projected events, identify inefficiencies and problem areas, evaluate alternative management approaches, improve understanding of variables that influence waste flow to WIPP, and support development of an integrated complex-wide plan for achieving disposal of TRU waste. This simulation model forms the core of a larger system being developed by INEL, Sandia National Laboratory, and the NTP to address other management aspects such as budgeting and remote-handled waste. Results of these efforts will be assimilated into a complex-wide National TRU Waste Program Management Plan by September 1996.

The Matrix Depletion Program (11) is a cooperative effort between the INEL, Los Alamos National Laboratory (LANL), and Rocky Flats Environmental Technology Site (RFETS) for DOE-CAO. The objective of this program is to investigate the phenomenon of matrix depletion as it affects the generation of radiolytic-produced gases over time, and to develop age-dependent gas generation values. The expected outcome of this program is the development of data that will support an application to the Nuclear Regulatory Commission for lower effective gas generation values and corresponding higher thermal wattage limits for wastes shipped in the TRUPACT-II. This will increase the amount of untreated waste that can be shipped to WIPP without repackaging or treatment. Radioactive tests containing various waste matrices are expected to begin at LANL in February 1996 to provide quantitative gas generation data and development of bounding effective gas generation rates. Predictive modeling will be completed for actual TRU waste drums, followed by sampling of this waste by INEL and RFETS, for determination of actual hydrogen content to verify experimentally-derived time-dependent gas generation rates are bounding. This program is expected to be completed in February 1998.

Procurement of Treatment Services

In 1992, efforts were initiated to evaluate the feasibility of procuring treatment services needed for TRU-contaminated waste from the private sector as an alternative to design, construction, and operation of a DOE-funded LICP. Results of these studies have indicated that significant cost savings could be achieved for DOE, and completion of waste disposal could be accomplished eight years earlier. The INEL has issued a Request for Proposal (RFP) for treatment services. It is expected that award of the contract will be completed in September 1996. Treatment of all TRU-contaminated waste will commence by March 2003. Until that time, untreated waste will be shipped to WIPP for disposal.

FUTURE STRATEGIC DIRECTION

In October 1995, a settlement agreement was reached between DOE, the State of Idaho, and the Department of the Navy concerning future receipt of spent nuclear fuel at the INEL. This agreement will result in

significant impacts to the future management program for INEL stored TRU waste. Although the stored TRU waste program was moving in a direction consistent with the general terms of the agreement, the establishment of specific dates and volumes of waste to be removed from the State of Idaho will result in acceleration of planned activities. Key agreement conditions affecting management of stored TRU waste include: 1) the first shipment of TRU waste out of the State to WIPP or another designated facility must be accomplished by April 30, 1999; 2) by December 31, 2002, 3100 m3 of waste (15,000 drum equivalents) must be shipped out of the State; 3) after January 1, 2003, a running three-year average of 2000 m3/yr must be shipped out of the State; 4) a mixed waste treatment facility contract must be established by June 1, 1997, construction completed by December 31, 2002, and operational by March 31, 2003; and 5) shipment of TRU out of the State must be completed by a target date of December 31, 2015, but not later than December 31, 2018. Failure to comply with these conditions could affect the receipt of DOE spent fuel at the INEL.

As originally planned, production-level waste examination and certification operations will be initiated in the summer of 1997 to begin building a backlog of certified waste for shipment to WIPP in April 1998. Receipt of final WIPP WAC is assumed by December 1996. Alternatives for achieving shipment of 15,000 drums by December 31, 2002, are currently being evaluated.

SUMMARY

Historically, the INEL has proactively and successfully enhanced management capabilities for stored TRU waste. These capabilities include facility, operational, and technical development and support activities. Initiation of shipment of INEL stored TRU waste to WIPP will begin once WIPP is opened and authorization to ship is received.

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8. "Performance Demonstration Program Plan for RCRA Constituent Analysis of Solidified Wastes," Revision 0, DOE/CAO-94-1077 (June 1995).
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Session 30 -- POSTER - ENVIRONMENTAL REMEDIATION
Co-chairs: Mike Mauzy, Roy F. Weston
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OFFICE OF ENVIRONMENTAL MANAGEMENT WORLD WIDE WEB SERVER

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ABSTRACT

The Department of Energy (DOE), Office of Environmental Management (EM), Office of Information Management (EM-14), established the EM World Wide Web Server (EMweb) to provide up-to-date information to DOE Headquarters, Operations Offices, stakeholders, contractors, and the public. Information on the server is kept current through a configuration management plan maintenance schedule that includes a weekly upload of new information.

EMweb supports EM's mission by making data available on demand to users throughout the world. The categories of information currently on EMweb include Public Information Resources, Regulatory Information, Waste Management, Environmental Restoration, Science and Technology, Facility Transition, Site Operations, Crosscutting Programs, and Other Servers. The server also provides features to view on-line usage statistics for all applications; capability to review draft documents before public release of information; a response mechanism through electronic mail for users to contact staff that manage the server or are responsible for information posted on the server; and search engines that let users search the EMweb server or servers throughout the Internet. Licensed access to Secretary of Energy Notices, the Code of Federal Regulations, the Federal Register, DOE Orders, state environmental regulations, and the Commerce Business Daily is also provided for staff members located at DOE Headquarters.

INTRODUCTION

The Department of Energy (DOE), Office of Environmental Management (EM) World Wide Web Server (EMweb) was established to provide up-to-date

information to DOE Headquarters, Operations Offices, stakeholders, contractors, and the public. The server is located in Washington, D.C., and is accessible over the Internet throughout the world. Parts of the server are available to all users, and other parts are secured through passwords or specific Internet protocol (IP) addresses. This paper will describe EMweb features and discuss some of the more widely used documents on the server. The uniform resource locator for EMweb is "http://www.em.doe.gov."

Information on EMweb is organized by subject categories and programmatic management authority. Because EMweb menu structures are updated whenever the information posted by DOE and its stakeholders is changed, the current configuration of the server may not match that shown below. The main menu is currently subdivided into

- Public Information Resources,
- Regulatory Information,
- Waste Management,
- Environmental Restoration,
- Science and Technology,
- Facility Transition
- Site Operations,
- Crosscutting Programs, and
- Other Servers.

The server also provides several features to ensure that users find the information they need and can communicate with the providers of the information. Server features include

- on-line usage statistics for all applications;
- a "limited access" review area for use by information providers before public release of information;
- a "feedback" mechanism that can be used to send electronic mail to those that manage the server and are responsible for information posted on the server;
- search engines that let users search the EMweb server or servers throughout the Internet;
- a "restricted area" that can be used to post information only of interest to internal DOE Headquarters staff;
- licensed access to Secretary of Energy Notices, the Code of Federal Regulations (CFR), the Federal Register, DOE Orders, State Environmental Regulations, and the Commerce Business Daily;
- a "What's New" feature that identifies documents or features recently added to the server; and
- a "Features" highlight that denotes new items added to the server.

EMweb has been registered with all major search engine and Internet directory providers so that users throughout the world can locate the information they need on the EMweb server. In addition, EMweb is managed to ensure that information added to the server meets the needs of DOE, its programs, and stakeholders. All information on the server has been processed through a rigorous configuration control procedure that ensures information accuracy and consistency. All information posted must conform to the DOE Office of Environmental Management World Wide Web Server (EMweb) Hypertext Markup Language Style Guide (DOE/HWP-164). After information is converted to the hypertext markup language (HTML) format, the information is reviewed and approved by the federal employee who is responsible for the information.

Every World Wide Web server provides access to on-line text and graphics. EMweb goes a step further to help ensure that users get the information they need. EMweb provides services to download large documents, debate issues with other EMweb users, and access data in legacy database management systems.

INFORMATION PUBLISHED ON EMweb

EMweb Public Information Resources

The EMweb Public Information Resources directory contains information of general interest to EM and its stakeholders. Two documents are currently featured. Committed to Results: DOE's Environmental Management Program, An Introduction, which is also known as the "EM Primer," describes the EM organization, its goals, and objectives. It provides a colorful picture story of EM and its programs. A second document, Closing the Circle on the Splitting of the Atom, characterizes the environmental legacy of nuclear weapons production in the United States and describes what DOE is doing to address the resulting environmental problems.

The EM newsletters EM Progress, EMformation, and Full Circle, DOE's Quarterly Solid Waste Reduction and Affirmative Procurement Bulletin are also featured in this section. Efforts are being made to ensure that publication on EMweb corresponds with hardcopy distribution to more effectively communicate the newsletters' contents to DOE and its stakeholders.

Other information in the Public Information Resources section of EMweb includes

- EM Fact Sheets;
- Environmental Management 1994;
- Environmental Management 1995;
- Anuario Hispano 1995 Hispanic Yearbook;
- Estimating the Cold War Mortgage: The 1995 Baseline Environmental Management Report (Executive Summary only);
- Resource Conservation and Recovery Act (RCRA) Corrective Action & Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) Remedial Action Reference Guide;
- Speeches;
- Integrated Risk Management Home Page;
- Information and Announcements;
- Press Releases;
- Public Participation Information;
- Conference and Meeting Information; and
- Job Listings.

EMweb Regulatory Information

The Regulatory Information portion of the server contains information for

- The Advisory Committee on External Regulation of DOE Nuclear Safety
- Regulatory News and Meeting Notices
- National Environmental Policy Act Implementation

EMweb Waste Management Information

The Waste Management portion of the server features the Integrated Data Base (IDB) Report, the Federal Facility Compliance Act (FFCAct) Bulletin Board, and the Waste Management Programmatic Environmental Impact Statement (PEIS) Bulletin Board.

The IDB reports for calendar years 1992 and 1993 are currently on the server. The 1994 edition is currently being prepared for publication. The IDB reports offer historic data on inventories and characteristics of both commercial and DOE spent fuel and commercial and U.S. government-

owned radioactive wastes. The report is published annually by DOE's Office of Civilian Radioactive Waste Management and Office of Environmental Management.

The FFCAct Bulletin Board contains information concerning

Federal Facility Compliance Act Approved Site Treatment Plans;
Overview of the Draft Site Treatment Plans;
Overview of the Proposed Site Treatment Plans;
Downloadable Documents and Files;
FFCAct Federal Register Notices;
FFCAct Issue Alerts and Fact Sheets;
Federal Facility Compliance Act of 1992;
Site FFCAct Points of Contact Listing;
DOE Mixed Waste Inventory by State;
Progress Report to Congress - 1994;
Progress Report to Congress - 1993;
Implementation Plan for the Programmatic Environmental Impact Statement for the DOE Environmental Restoration and Waste Management Program; and
National Summary Report of Draft Site Treatment Plans.

The Waste Management Programmatic Environmental Impact Statement (PEIS) Bulletin Board contains information concerning

Waste Management Programmatic Environmental Impact Statement (PEIS) DOE Update: September 1995;
Waste Management PEIS Site-Specific Fact Sheets;
National Results of the Waste Management PEIS;
An Overview of the Waste Management PEIS;
The Waste Management PEIS and Public Comment Opportunities;
Implementation Plan for the Programmatic Environmental Impact Statement for DOE EM; and
A description of the "Relationship Between the Environmental Management Programmatic Environmental Impact Statement and the Federal Facility Compliance Act."

EMweb Environmental Restoration Information

The Environmental Restoration portion of the server features the DOE Environmental Management Benchmarking Clearinghouse, the Generic Technical Management Plan, and the Environmental Restoration Strategic Plan.

The Benchmarking Clearinghouse section contains

Benchmarking Abstracts and Reports,
Current DOE Comparative Performance Data,
Benchmarking Data Sources,
Points of Contact for Additional Benchmarking Information,
Frequently Asked Questions About Benchmarking in Environmental Management Projects, and
Remedial Action Program Information Center.

The Environmental Restoration Strategic Plan was developed to guide decision-making in the planning and execution of DOE's National Environmental Restoration Program. The plan is designed to implement the Environmental Management Strategic Plan and to reflect the roles, responsibilities, and authorities of Headquarters and field organizations as defined by the Assistant Secretary for Environmental Management.

EMweb Science and Technology Information

The Science and Technology portion of the server features the complete text and graphics for the June 1995 Technology Summary Reports (Rainbow

Books) that are published by DOE EM's Office of Technology Development. The following Rainbow Book Reports are provided:

- Robotics Technology Crosscutting Program;
- Characterization, Monitoring, and Sensor Technology Crosscutting Program;
- The Landfill Stabilization Focus Area Technology Summary;
- Contaminant Plumes Containment and Remediation;
- Efficient Separations and Processing Crosscutting Program;
- Facility Deactivation, Decommissioning, and Material Disposition Focus Area; and Tank Focus Area Technology Summary.

Also included in the Science and Technology section are the Technology Information Exchange (TIE) Bulletin Board, the U.S. DOE Office of Environmental Management Oak Ridge Operations Technology Needs Database, and the U.S. DOE Scrap Metal Inventory Report.

EMweb Crosscutting Programs Information

The Crosscutting Programs section of EMweb is the fastest growing section, partly because the Internet has become an excellent mechanism for disseminating up-to-date information quickly and effectively across the DOE complex. Home Pages on the Crosscutting Programs section include

- Progress-Tracking System Information Home Page,
- Training and Education Home Page,
- Transportation Program Home Page,
- Facility Representative Bulletin Board,
- Pollution Prevention Home Page,
- National Decommissioning Program Home Page, and
- National Recycle Program Home Page.

ACCESS TO SPECIAL SERVICES

Because of the demand for on-line access to government-generated laws, regulations, and guidance documents; DOE EM has made arrangements with a publishing company for access to

- The Commerce Business Daily,
- The Code of Federal Regulations,
- The Federal Register,
- DOE Orders,
- Secretary of Energy Notices, and
- State Environmental Regulations.

Access to these services is controlled by Internet Protocol (IP) address. Currently, anyone within the Class B IP address scheme for DOE Headquarters can access these services. EM is attempting to acquire free access to these services. DOE is also negotiating with the service provider to widen the agreement to cover Operations Offices and contractors if free access cannot be provided. At the time of this printing, a contract has not been awarded.

SUMMARY

EMweb has been established as a resource that provides up-to-date information to DOE Headquarters, Operations Offices, stakeholders, contractors, and the public. The server is updated weekly, so new items are continuously available to users.

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CERCLA FEASIBILITY STUDY PLANNING AND INTEGRATION FOR REMEDIATION OF THE RADIOACTIVE WASTE MANAGEMENT COMPLEX SUBSURFACE DISPOSAL AREA AT THE INEL

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ABSTRACT

This paper summarizes the strategy and supporting activities for developing proposed alternatives for remediation of the Subsurface Disposal Area (SDA) at the Radioactive Waste Management Complex (RWMC). These activities are being described to illustrate one approach to tackling a complex, comprehensive clean-up decision using an interim action as a vehicle for focusing the technology evaluations. Prior to the start of this feasibility study the interim action was initiated to clean-up part of the SDA, Pit 9. The Pit 9 clean-up comprises remote retrieval under containment, physical separation processes, soil treatment, and stabilization via thermal treatment. The cost of the Pit 9 interim action is estimated to be approximately $260 million. While this is clearly a significant investment it represents only a small portion of the potential cost of remediating the SDA if ex situ treatment is needed. The feasibility study will include detailed evaluations using Pit 9 processes as representative processes for an alternative of retrieval and ex situ treatment, along with other alternatives that will make up the remainder of the evaluations.

BACKGROUND

The SDA is a large (38.85-ha) landfill located at the Idaho National Engineering Laboratory (INEL) that contains buried radioactive waste generated at the INEL and shipped in from other waste generators starting in 1952. The SDA is still being used for current low level radioactive waste disposal. A Federal Facility Agreement and Consent Order (FFA/CO) for the INEL was signed by the U.S. Department of Energy, U.S. Environmental Protection Agency, and State of Idaho Department of Health and Welfare to address past releases of radioactive and hazardous substances, including the migration of contaminants to the Snake River Plain Aquifer which underlies the INEL (1). This aquifer was designated as a sole source aquifer by the EPA in 1991. The agreement effectively moved the investigation and cleanup of specified releases at the INEL from a RCRA to a CERCLA process. Remediation of the SDA is included under this agreement, and development of a proposed remedial alternative for the SDA will be conducted under a CERCLA Comprehensive Remedial Investigation/Feasibility Study.

Much of the waste disposed in the SDA was TRU-contaminated waste generated at the Rocky Flats Plant and shipped to the INEL for burial from 1954 to 1970. Waste was buried in pits, trenches, and soil vaults, and historical records from the SDA indicate that prior to 1970 TRU-contaminated waste was mixed with non-TRU waste during burial (2). After 1970 the Atomic Energy Commission (AEC) issued a policy regarding segregation and storage of waste contaminated with TRU radionuclides

which led to the decision to cease all burial of TRU-contaminated waste at the SDA in that year. The general categories of waste buried at the SDA include construction and demolition materials, laboratory equipment and materials, process equipment and materials, nuclear reactor components such as fuel scraps and radioactive sources, maintenance equipment and scrap metals, decontamination materials, and miscellaneous waste. Miscellaneous waste includes such things as process sludges, animal carcasses, contaminated jet engine components, contaminated vehicles, batteries, etc.. Preliminary investigations evaluating human health risk drivers at the SDA (3) have identified contaminants of potential concern (COPCs), including 43 radiological contaminants and 44 non-radiological contaminants that will be evaluated under a Remedial Investigation/Feasibility Study. Plutonium and Americium are included in the list of COPCs. Additionally, the waste inventory at the SDA includes a large list of other hazardous constituents that do not pose a risk to human health or the environment in their current state. These must be evaluated for potential problems, such as potential worker exposure and interference with process equipment performance, under a scenario where there is intrusion to the buried waste as part of a remedial alternative. The breadth of the list of contaminants, variety of contaminated media, and potential for staggering clean-up costs make for a daunting task to remediate the SDA. In 1991 an Interim Action was initiated to clean up part of the SDA, Pit 9, using a private sector, turnkey approach to meet key Department of Energy (DOE) objectives. Several objectives for the Pit 9 Interim Action are directly applicable to the SDA remediation. The Pit 9 Interim Action is designed to be a near-term remediation of plutonium, americium, and other identified risk drivers in Pit 9 at a minimum cost to DOE, as well as a demonstration of private industry's capabilities for remediating TRU-contaminated mixed waste sites using innovative technologies (4). The remediation of Pit 9 includes excavation and segregation of buried waste with greater than 10 nCi/g TRU elements for input to the treatment process train. Treatment includes chemical extraction, physical separation, and/or stabilization to remove radionuclides and hazardous constituents. The remedial design for the Pit 9 remediation is scheduled to be complete April, 1996, followed by a Limited Production Test (LPT) which will provide early data on equipment performance and operational readiness prior to remediating the remainder of Pit 9. LPT data will be available for evaluation under the Feasibility Study (FS) for remediating the remainder of the SDA.

KEY OBJECTIVES AND INTEGRATION OF THE FS

A strategy for the FS was developed prior to detailed scoping between stakeholders. The strategy included documentation of key assumptions used for the FS (5). This assumptions document has been and will continue to be revised until completion of the FS. It provides some boundaries to the scope of work, and a starting point for stakeholder consensus-building. The strategy also included the development of some high level questions that needed to be answered as part of the FS. Using the Pit 9 Interim Action to answer some of these questions is a critical piece of the strategy. Key objectives for the FS include:

Identify and group contaminated media from the operable unit so that a reasonable number of response actions can be developed.

Gather early information on technology costs/complexities to identify areas of greatest uncertainty, and rough order of magnitude costs for

technologies that can be used for screening decisions and integration with risk assessment activities.

Identify areas where costs are most significant so that this information could be factored into the remedial decisions.

Develop a list of remedial alternatives that will be used to evaluate and compare grouped technologies in accordance with CERCLA criteria.

Some high level questions that need to be answered in this FS include:

1. Is it likely that a landfill cap will be protective for the period of time necessary to prevent groundwater risk from the SDA? To answer this question functional and operational requirements for a cap over the SDA need to be considered, particularly the required design life for a cap. Based on preliminary risk calculations, for example, some contaminants of potential concern pose risks to groundwater for thousands of years6.
2. Is it feasible to consider in situ or ex situ treatment of the entire SDA considering potential costs and complexities that arise from treatment of such a large volume of heterogeneous waste?
3. Do we have necessary data, primarily from waste inventory and shipping records, to define partial treatment scenarios that will reduce the volume of waste that would need treatment?
4. In the case of an alternative to retrieve and treat waste, will the treatment technologies work on heterogeneous waste such as will be found in the SDA?
5. Are treatability studies necessary to provide part of the technical basis for confirming the feasibility of a technology/alternative or are existing data adequate?
6. Can opportunities for technology development be identified in time to provide a window for bringing in new technologies that could be used to reduce complexity and/or cost of remediation?

The feasibility study will include the following conceptual response actions:

No Action

Occupational Period to prevent or limit access to contaminated areas

Containment In Situ Treatment

Retrieval, Ex Situ Treatment, Storage, and Disposal

Retrieval, Storage, and Disposal

The containment response action will include evaluation of composite capping technologies. It will likely use water balance caps similar to those developed at the Hanford Site as representative technologies. Generally these designs are being considered for applications with 500 - 1000 year design lives.

EARLY RISK ASSESSMENT INDICATORS

In preparation for conducting the remedial investigation a Preliminary Scoping Risk Assessment (PSRA) was completed in 1994 (6). The primary objective for this task was to provide early information regarding contaminants, exposure pathways, and other factors in determining potential adverse human health effects from the SDA. This activity provided some conclusions on key sensitivities for risk assessment of the SDA, and will be used to prioritize modeling and data-gathering efforts in preparation for the next major task of the remedial investigation, the baseline risk assessment, scheduled to begin spring of 1996.

The PSRA provided a list of contaminants that posed potential risk for all pathways under evaluation. The groundwater contamination pathway is considered the most influential in coming to a remedial decision for the SDA. Using conservative assumptions resulted in three inorganic

contaminants potentially presenting the highest risk to groundwater from the SDA: C-14, I-129, and Tc-99. Later evaluations of SDA inventories support a significantly lowered inventory for I-129, which will result in a lower risk to groundwater. The main transuranic contaminants in the SDA, Pu and Am, were retained as contaminants of potential concern but based on current assumptions these contaminants pose less of a risk to groundwater than the C-14, Tc-99, and I-129. There has been and continues to be public concern about the transuranic waste buried in the SDA and this concern will also be factored into the feasibility study. Hazardous organics also pose a potential groundwater risk based on early models of the SDA, and recent groundwater samples from monitoring wells indicate there may be an increasing trend in carbon tetrachloride concentrations. The baseline risk assessment will be used as the final tool for determining contaminants of concern, but until these results are known, PSRA results are useful indicators for what kind of remediation alternatives are likely.

According to the results of the PSRA, several parameters were very sensitive in influencing risk assessment results. Release rate was considered the most important of these parameters. Release rate generally models how contaminants move and at what rate they move from the disposed waste. It includes parameters that are derived from three main release mechanisms for SDA waste, namely, surface to water partitioning of sorbed constituents between solids and water, corrosion of metal waste with corresponding release of constituents, and equilibrium solubility limit applied for dissolution of liquids and solids in oxide, hydroxide, nitrate, particulate form, etc. Source inventory was another sensitive parameter and is dependent on our current ability to accurately estimate how much of a given contaminant was placed in the SDA. Inconsistencies and incompleteness in inventory and shipping records create uncertainty in many contaminant inventory estimates. Discussions with Rocky Flats Plant employees and others have been conducted to obtain better waste inventory information from process knowledge, but there have been no direct measurements to calibrate current inventory estimates with field data. Additional information that could be gained regarding release rates, source term inventory and other sensitive parameters would be of highest priority in completing the remedial investigation/feasibility study since input to these terms will greatly influence the results of the risk assessment, and therefore influence the complexity and cost of any necessary remedial action.

PIT 9 INTERIM ACTION

The Pit 9 interim action has the potential of answering many questions regarding how effective specific retrieval and treatment processes are for remediation of heterogeneous, buried waste. It also provides opportunities for learning how the contaminants in Pit 9 have changed and migrated over a period of about 25 years. The remedial action will include a containment building that can be repositioned along the length of the pit as excavation proceeds. Remote operation of the waste retrieval process will occur within this containment after the clean overburden is removed from Pit 9. The retrieved waste and/or soil will be assayed to determine whether it needs treatment or not. Waste and soil requiring treatment will be packaged and transported to either the thermal treatment or the soil treatment process. Treatment residuals from the soil treatment process will be fed to the thermal treatment process. Treatment residuals from the thermal treatment process will be packaged

and stored in preparation for ultimate disposal as transuranic waste at the Waste Isolation Pilot Plant (WIPP).

The Pit 9 clean-up specification calls for collection and analysis of approximately 114 soil samples during the remediation phase. These samples will be analyzed for VOCs, semivolatile organics, transuranics, Pu-241, Co-60, Sr-90, C-14, I-129, Cs-137, Tc-99, H-3, Ni-59, Ni-63, Hg, Pb, Be, Zn, As, Cr, cyanide, and nitrates. Whatever portion of this information that is available during the FS for the SDA will be used to understand what kind of contaminants the Pit 9 treatment processes are actually seeing in the feed, and what contaminants are present in different areas of the pit.

One area of significant uncertainty in conducting the FS for the SDA is evaluating the viability of technologies for treating waste considered infinitely heterogeneous. Performance of any treatment process is highly dependent on the waste feed, and evaluating technology performance using data on actual waste is the optimum situation. The Pit 9 interim action provides exactly this for the FS. While Pit 9 is not representative of all pits and trenches in the SDA, it does provide a relatively broad envelope of waste feed that will rigorously test technology performance. A list of other waste found in the SDA and not in Pit 9 is being compiled for the FS to determine if additional testing is warranted or if assumptions can be made on technology performance with these wastes, with acceptable uncertainty to the stakeholders. The vast majority of radionuclides known to be disposed in Pit 9 are transuranics, therefore, it is assumed that very little Tc-99 or C-14 will be processed during the interim action. Since these contaminants posed the highest risk to groundwater in the PSRA special consideration will be used to verify Pit 9 technologies are feasible for these waste forms.

Another area of uncertainty when evaluating technologies for waste treatment is estimating the cost of remediation, including operations and maintenance. Particularly for the case of plutonium operations, on-stream time and maintenance costs are extremely important parameters in gauging the success and cost of the treatment train. The Pit 9 interim action will demonstrate the cost of operations and maintenance during the limited production test and during the one year full-scale clean-up of the pit. This kind of cost information is far more accurate than estimates based on historical treatment of waste that may not be at all like that found in the SDA.

The safety of the operations will also be demonstrated during the Pit 9 interim action. A CERCLA feasibility study must include an assessment of the risk of implementing each remedial alternative. Retrieval and treatment of buried waste from the SDA does include some risk, including potential exposure of workers to hazards of the operation. The Pit 9 operations will provide data on the severity of these hazards, and the ability of the equipment and operational designs to accommodate all hazards that arise. This information is invaluable in convincing stakeholders of the viability of a remedial alternative.

CURRENT AND FUTURE ACTIVITIES

Technical activities for the FS are still in the early, data-gathering stage. Screening decisions will begin in 1996 as the results of the baseline risk assessment are compiled. Detailed evaluations of remedial alternatives will be finalized after the Pit 9 LPT results have been incorporated, tentatively as early as January, 1997.

Prior to screening, several parallel activities have been initiated. In preparation for the Pit 9 LPT, engineers responsible for the FS are reviewing Pit 9 design documents and test plans to identify the data most useful for the FS anticipated from Pit 9, as well as data that would be valuable but are not currently defined within the LPT. The list of valuable data not currently in the scope of the LPT will be evaluated to determine if additional testing should be done with Pit 9 processes to generate these data, or whether the stakeholders are willing to accept the uncertainty and forego additional testing.

Another activity that has been initiated is to answer the question previously raised on partial retrieval/treatment scenarios. While it is straightforward to identify pits and trenches that could not contain specific contaminants using available inventory records, there is much greater uncertainty regarding which pits and trenches contain which contaminants. The uncertainty rises again in estimating concentrations of contaminants for a given pit or trench. A preliminary look at which pits and trenches could not contain specific contaminants showed only a small percentage of the SDA that did not contain some of the contaminants that may pose a future groundwater risk (based on PSRA results). Less than 40% by area of the SDA does not contain Pu or Am, for example, according to inventory records. About the same percentage (though not necessarily the same pits or trenches) can be shown to not contain Tc-99 and C-14. VOCs were disposed in over 30% of the SDA by area according to records. It has been estimated that retrieval and treatment of the entire volume of contaminated media at the SDA will cost over $1.5 billion (7), excluding the cost of interim storage and final disposal of treatment residuals. One can easily infer from this that even small reductions in volume of waste that must be treated would reduce the cost of remediation by millions of dollars. One strategy for reducing the volume of waste that might require treatment is to go back to the inventory and shipping records to find information that could eliminate more pits and/or trenches from the list of known pits containing contaminants of concern. Information that could eliminate portions of a pit or trench would also be useful, but this is less likely to have been documented in historical records. Another avenue for addressing this uncertainty is to evaluate options for locating specific contaminants by measurement at the SDA. If these measurements could only be done by intrusion into the waste the cost of obtaining these data could be prohibitive and safety issues would need to be addressed beforehand. If, however, some contaminants could be located via non-intrusive measurements through the overburden then a window of opportunity exists for attempting these measurements before the FS is completed.

We are just beginning to ask and attempt to answer the hardest questions of this feasibility study. While the focus of this paper has been on the importance of the Pit 9 interim action, it should be noted that a significant portion of the FS will be devoted to other alternatives in addition to ex situ treatment. Containment issues, in particular, will require careful evaluation against the problem at hand, and making maximum use of existing data on capping technologies is a key step in accomplishing the technical objectives of this program.

Having access to a large interim action to support this FS will be a great asset to this work. Unfortunately, other feasibility studies have been completed with the benefit of an interim action and were not entirely successful. The usefulness of this work will be bounded by our

ability to distinguish what aspects of the interim action can be extrapolated to the waste that may need remediation in pits and trenches outside of Pit 9, and what judgment we apply to evaluating the treatment processes for contaminants, and a scale of operations not tested in the interim action.

ACKNOWLEDGMENT

This work was supported by the U.S. Department of Energy Office of Environmental Restoration under DOE Idaho Operations Field Office Contract DE-AC07-94ID13223.

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COST-EFFECTIVE APPROACHES TO SUCCESSFUL REMEDIATION OF UNEXPLODED ORDNANCE (UXO) AT THE IDAHO NATIONAL ENGINEERING LABORATORY (INEL)

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ABSTRACT

Unexploded ordnance (UXO) and explosively contaminated soils were remediated at the Idaho National Engineering Laboratory (INEL). A Geographical Information System (GIS) was used to map areas of contamination and a Global Positioning System (GPS) was used to identify the location of ordnance explosive waste (OEW). Using these systems has improved the accuracy of the data collected. A central demolition site was used instead of the traditional method of detonating unexploded ordnance in place. Field screening of explosive contaminated soils was performed instead of using a more expensive laboratory analysis. In addition, an Air Force remote excavator was used to remediate potentially sensitized ordnance disposal pits in a cooperative technical demonstration arrangement with the Department of Defense (DoD) and the Department of Energy (DOE). The use of these methods and technologies has resulted in an enhanced remediation process and significant cost savings.

The next phase of ordnance work at the INEL will involve using a computer-based site statistical sampling model for UXO estimation entitled SiteStats. This model will eliminate the need for labor-intensive field searches to estimate ordnance concentrations.

INTRODUCTION

Before the inception of the Idaho National Engineering Laboratory (INEL), military activities such as aerial bombing practice, naval artillery testing, explosive storage bunker testing, and ordnance disposal took place on a large portion of what is now the INEL. As a result of these past activities associated with the former Naval Proving Ground (NPG), numerous unexploded ordnance (UXO) devices have been discovered by INEL personnel. In addition to UXO and ordnance explosive waste (OEW), explosive agents such as TNT and RDX, released during partial detonation during NPG tests, have contaminated soils at the INEL. This paper will describe how: a) UXO has been remediated effectively and in a cost-effective manner, b) technologies have been used to enhance cleanup, and c) to save money sampling explosive contaminated soils.

Over the past 3 years, UXO and explosively contaminated soils have been removed from the INEL. Lessons have been learned on appropriate approaches to take to effectively and in a cost-effective manner remediate OEW. While the cost of remediating ordnance from former military installations continues to rise, efforts are being made at the INEL to reduce these costs using innovative technologies.

Beginning in 1993, work began to remediate ordnance and explosive contaminated soils at six INEL locations. This work was completed the same year. The specific mission of this action was to locate, identify, detonate, and dispose of UXO and associated shrapnel and to characterize, remove, and incinerate soils contaminated with explosive residues.

The first phase of the ordnance removal was to set up search lanes in each area to sweep for ordnance. Five-foot lanes were established by placing metal posts and running string to each point. A visual search was made first for UXO/OEW and debris on the surface. Each potential ordnance item was flagged by explosive ordnance disposal (EOD) technicians. This search was followed by a magnetometer search to check for subsurface anomalies to a 2-foot depth. On the second search, any additional items were marked. Each flagged anomaly was investigated to determine if UXO existed. Located UXO was collected and if determined to be safe, was transported to a central blast area for destruction.

Following the clearance of an area, a quality survey of the lanes cleared was performed. If any ordnance items were discovered during the quality check, the EOD team would reclear the lanes while in the area. This ensured that a thorough clearance was performed and saved project costs.

Historical U.S. Department of Defense (DoD) EOD publications were used to positively identify ordnance in the field. Some of these publications include: Bombs for Aircraft (TM 9-1980, 1944); General Ammunition Technical Manual (TM 9-1900, 1942); and U.S. Explosive Ordnance (NAVSEA OP 1664, 1947). Since ordnance deposited by the military during World War (WW)-II is outdated, the newer DoD publications usually do not identify the older ordnance. These historical publications assisted with identifying ordnance type, filler, and fusing. This reduces the expense of blowing up inert and nonhazardous ordnance in place. Another benefit of using these publications, eliminates the necessity of sending pictures, drawings, dimensions, and other features of the ordnance to the

EOD Technical Center for identification. Processing this information through the EOD Technical Center would slow field work and cause additional project expense.

Cost savings have been realized by transporting ordnance to a central disposal area. This eliminated the need to clear areas around ordnance that would be normally blown up in the field. Clearing areas around ordnance to prevent fire danger is time-consuming and labor-intensive. Also, if each item discovered in the field is blown-in-place, considerably more explosives will be used and each detonation introduces risk. Decreasing the number of explosive detonations decreases risk associated with each disposal. Additional time and money would also be spent filling in craters and reseeding areas if ordnance was blown-in-place.

During the EOD search efforts to locate UXO, locations of soil contaminated with explosive compounds were marked. Locations with pieces of explosive compounds or stained soil present were flagged by EOD teams for sampling and the level of soil contamination was identified using field screening methods. A field laboratory was set up to conduct the sample analysis. The Jenkins Method (1) was used to perform the onsite field screening for explosive contaminated soil. This approach proved to be much less expensive than traditional laboratorysample analysis. The field screening cost per sample was $25, where the EPA 3380 Method for analysis averages $600 per sample. An estimated $297,000 was saved by using the field screening method. Ten percent of the samples were sent to an offsite laboratory for the full EPA 3380 Method analysis for verification purposes. The verification samples indicated the field screening sampling to be accurate. (2) Future samples of explosive contaminated soils will use the field screening methods. The INEL has worked with the U. S. Army Corps of Engineers (Corps) in testing and improving the Jenkins field screening methods (1,3) originally developed by the Corps.

After the soils were characterized for concentrations of explosive materials, the decision was made to remediate 186 cubic yards of soil. The soil was excavated, containerized, shipped offsite, and incinerated. The cost to incinerate the soils was $1,000 per cubic yard. (4) The current preferred method for remediation of explosive contaminated soils has changed to bioremediation. Bioremediation of explosive contaminated soils has been estimated at between 200-$400 per cubic yard. (3) The INEL is exploring ways to begin bioremediation of explosive contaminated soils. This new approach will realize substantial cost savings compared to past incineration of explosive soils.

A second ordnance removal action began in 1994 and was completed in 1995. This activity included the removal of ordnance from 90 acres from the Twin Buttes Bombing Range and 40 acres and six disposal pits at the Naval Ordnance Disposal Area (NODA).

Collecting information to identify the location, type, and disposition of ordnance is an important part of ordnance removal projects. In the past, ordnance information was manually collected and location information estimated. During the ordnance removal action in 1995, use of a Geographical Information System (GIS) to map areas of ordnance concentrations and a Global Positioning System (GPS) to identify locations of OEW has aided and enhanced the remediation process. Using these systems has improved the accuracy and quality of the information collected. Cost savings have been realized by eliminating the need to

manually input field data. The field data collected can now be electronically transmitted to the GIS. The information collected will assist in future assessments of UXO at the INEL and provide a baseline for projecting further cleanup needs.

During cleanup of the NODA area, a potentially sensitive ordnance disposal pit was identified. To mitigate the potential hazards of this area, it was determined that a remote operation was necessary in this area. Costs to bring in a remote excavator were estimated at $200,000. An Air Force remote excavator (Track Catapiller 325) from Tyndall Air Force Base was selected, which was part of a previously scheduled joint demonstration project between the U.S. Air Force and the U.S. Department of Energy (DOE). The excavation of the potentially sensitive pit was included as part of the technology demonstration project. As a result of these unique activities, sites at the INEL were successfully remediated of UXO/OEW. Continued remediation will be performed in a cooperative arrangement with the DOE, DoD, and the Corps, who all work toward cost reductions for ordnance removal.

The next phase of ordnance work at the INEL will be to assess the overall levels of bothordnance contamination and the volume of explosive contaminated soils. As part of this assessment, a computer-based site statistical sampling model for UXO estimation will be used. The software entitled SiteStats, a methodology developed by the Corps, will be used at the INEL. This tool will reduce the number of labor-intensive field searches for estimating ordnance concentrations. SiteStats will allow a minimal number of ordnance investigations, which will result in cost savings.

Innovative and efficient ordnance removal techniques will save taxpayers' money. By using new technologies, the cleanup of ordnance can be enhanced. Such techniques as field sampling for explosive contaminated soils versus traditionally more expensive laboratory analysis, electronic collection of field information, cooperative demonstration projects, and bioremediation of explosive soils will greatly improve and make ordnance removal projects much more cost-effective. The INEL has developed an integrated ordnance remediation approach the can be applied at other DOE/DoD sites.

It is important to promote cost-effective methods especially during these times of Government downsizing and reduced budgets. There remains a number of former defense sites that will require ordnance removal and soil remediation. Sharing information among those organizations responsible for ordnance removal will result in substantial cost savings.

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"TOOLS" INTRODUCTION TO THE INEL ER PROGRAM MANAGEMENT TOOLS

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ABSTRACT

The INEL ER Program discovered early in the program that it needed two things: 1) an agreement with the State of Idaho and the EPA that supported working together to deal with the CERCLA problems at the INEL, and 2) tools to manage the program in a time of change. These tools needed to meet the needs of the program and the intent of the DOE Orders. This paper presents these tools in a condensed format developed to do many communicating tasks. Three of the most used applications of these tools are:

Training aid for new members of the staff

Communication tool for all of the organizations performing overview activities of the program

Background for new management joining the program.

These papers were developed in a standard format and were called the two-page papers. This title was derived from the requirement that all of the information provided must fit on the front and back of a single sheet of 8 1/2 11 sheet of paper. As you read these two-page papers you will discover that each paper provides 1) Background, 2) How the Process Works, 3) Who Could Use This Process, 4) Why Should the Process Be Used, and 5) INEL ER Program Contacts. This format has been very successful. The next step is to place these papers on the INEL internet home page so that everyone has access to this information. In developing this tool system we had strong support from many in the program but the brunt of the work fell on the point of contacts listed in the papers. If you have questions on the subject matter of any of the tools please contact one of the program contacts. If you are interested in the two-page paper process to outline tools please contact L. Hutterman at (208) 526-3647.

THE INEL EXPEDITED CERCLA PROCESS

Background

The Idaho National Engineering Laboratory (INEL) has developed, and successfully implemented, guidance for a structured process to address known and potential environmental restoration sites. Numerous environmental restoration sites are present at the INEL. Many of these are low-probability hazard sites, which are those with suspected or unknown quantities of contamination. Typically, these sites are characterized by an uncertainty about the existence, quantity, type, or location of hazardous substances.

The Federal Facility Agreement and Consent Order (FFA/CO) Action Plan defines these sites as Track 1 and 2 sites. A Track 1 site investigation involves compiling existing information combined with conducting a very conservative qualitative risk assessment. Track 2 investigations are similar, but they allow for limited data collection. Based on the very conservative assumptions applied by the Track 1 and Track 2 processes, remedial action project managers can make timely decisions concerning these sites with a relatively high degree of certainty.

Track 1 and 2 sites at the INEL are a result of disposal operations and waste generation from activities such as nuclear reactor testing and development, nuclear fuel processing, abandoned underground storage tanks, and solid waste landfills spanning over 45 years. Many of the processes and associated facilities that created these sites were

abandoned long ago or evolved over years, making it difficult to obtain accurate documentation and make accurate assessments. Site information is often nonexistent, limited, or takes months to compile. Personnel familiar with the past operations of these sites are either unavailable or unable to give factual information.

In an effort to better incorporate these sites into the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) process, Track 1 and 2 sites were deliberately separated from the standard Remedial Investigation/Feasibility Study (RI/FS) process; a subpathway to the RI/FS process was developed (see figure). The sites may still require additional data collection, but to a lesser extent. CERCLA provides a rigorous decision process for determining the need to perform remedial actions on hazardous waste sites. Remediation decisions are based on the concept of acceptable risk. Using the RI/FS process on the low-probability hazard sites would divert funding from areas where remedial actions are required because of known unacceptable risks. The documents, Track 1 Sites: Guidance for Assessing Low Probability Hazard Sites at the INEL and Track 2 Sites: Guidance for Assessing Low Probability Hazard Sites at the INEL, were developed as guidance for evaluating these low-probability hazard sites and developing documented, defensible decisions concerning whether the sites require a more quantitative risk assessment to determine the need for remedial action. However, the decision making responsibility of the remedial project managers is not diminished.

Another process negotiated and implemented at the INEL involves acceleration of the typical 40-month RI/FS schedule in cases where the contaminant types and concentrations were well known and exhibited unacceptable risks from the outset. During the scoping phase of the RI/FS process, it was determined that sufficient information was available to conduct a baseline risk assessment; therefore, the RI/FS work plan tasks could be eliminated. Additionally, using the team approach, the participating agencies significantly shortened document development and review cycles.

How The Process Works

Rigorous quantitative risk analysis as outlined by CERCLA is not appropriate for Track 1 sites. Funding for feasibility studies and interim remediation is more fittingly directed to known hazard locations. The Track 1 guidance was developed to provide defensible qualitative evaluation of low-probability hazard sites and to record compliance with environmental regulations. The Track 1 guidance was designed to direct project managers through a qualitative risk assessment and provide a compliance template for Track 1 sites. The guidance provides detailed instructions for performing the essential components of risk assessment, from hazard identification through a final recommendation. A conservative screening technique is used that accounts for both human health risks and environmental impacts, with humans as sensitive indicators for the environment. This technique, in conjunction with the organized collection of historical data and all other available information, is used to develop a qualitative risk assessment. The result is a compact document that contains a template, references, and information used in the decision process and presents defensible recommendations for the Track 1 sites under consideration.

The Track 2 classification was developed specifically for the INEL to use in streamlining the implementation of CERCLA. Track 2 low-probability hazard sites are not described in the National Contingency Plan or in

existing regulatory guidance. The goal of the Track 2 process is to evaluate such sites using existing qualitative and quantitative data to minimize the collection of new environmental data. The Track 2 guidance presents a structured format consisting of a series of questions and tables and qualitative risk assessment to provide the information necessary to make decisions concerning the remedial action to take at a site. The process is iterative and addresses a site from multiple perspectives (e.g., historical, empirical, process knowledge) in an effort to generate a reproducible and defensible method. This approach follows the data quality objective process and establishes a well organized, logical approach to use in consolidating and assessing existing data and establishing decision criteria. If necessary, the process allows for the design of a sampling and analysis strategy to obtain new environmental data of appropriate quality to support decisions. Finally, the guidance expedites consensus between regulatory parties by emphasizing a team approach to Track 2 investigations.
RI/FS schedule acceleration has almost always occurred within the scoping process for the site under investigation. Presentation of sufficient existing data for a well defined site has allowed INEL environmental restoration project managers to negotiate the elimination of RI/FS work plans. This can eliminate several months from the overall schedule. The team approach has also been developed with respect to document development and review. Shortened review cycles, and, in some instances, concurrent reviews, have been negotiated to further accelerate the schedule. These shortened schedules are developed and documented in the scope of work for the specific site undergoing the remedial investigation.

Who Could Use This Process?

The INEL's expedited CERCLA process is flexible enough that it could be used at any Federal or commercial facility with either low probability hazard sites or sites posing high risks. The expedited CERCLA process can focus the scoping phase of any CERCLA investigation.

Why Should The Process Be Used?

- Reduces cost and schedule of CERCLA investigations
- Provides a method for collecting and organizing information for decision making process
- Develops a team approach between agencies (encourages open communication)
- Affords an iterative/flexible process and addresses sites from multiple viewpoints
- Requires a rigorous approach following the Data Quality Objective and Observational Approach process
- Provides a well organized, logical approach for consolidating and assessing existing data
- Establishes decision criteria and acceptable uncertainty levels in a timely manner
- Allows for design of a sampling and analysis strategy that collects limited data of sufficient quality to support a remedial decision
- Saves valuable resources and refocuses these resources on the higher risk sites that require more in-depth study.

INEL ER Program Information

Additional papers that have been developed explaining the INEL ER Program and procedures are available on the Internet. The address for the papers is http://www.INEL.gov/environmental/tools/index.html. This address has

been developed as a clearing house for new ideas. Authors not from the INEL with innovative ideas are able to submit 2 page papers for inclusion in the briefing papers collection. Submission parameters are:

- The entire paper may not exceed 2 pages
- The topics are; Title, Scope, How it works, Who could use it, Why should it be used, and a contact name, phone number, address, and fax number.
- Type - use 10 point Gothic
- Submit papers in "HTML"

Submit your papers to:
Leonard Hutterman
P.O. Box 1625
Idaho Falls, ID 83415-3921
(208) 526-3647 or Fax (208) 526-4373

30-5

ENVIRONMENTAL MANAGEMENT AT BELGOPROCESS: THE CENTRALIZED WASTE TREATMENT FACILITY IN BELGIUM

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ABSTRACT

The services that Belgoprocess is offering in the field of radioactive waste treatment and storage, decommissioning of nuclear installations, or the sanitation of contaminated sites generate potential risks to the workers, the public and the environment. The company strategy is to keep the risks of harmful effects for people and the environment to an absolute minimum. Therefore Belgoprocess is working along a company wide environmental management program (EMP). This EMP consists of a set of combined measures (organizational, administrative, managerial and technological) to minimize the overall environmental impact according to the ALARA-principles and the application of BATNEEC.

The paper is focused on the detailed analysis of the key elements of the EMP:

internal measures

- combined safety, quality and environmental management program
- organizational structure with decentralized responsibilities and participation of all workers
- the use of stretched management goals
- internal communication and employee involvement
- technological measures, including the use of BATNEEC
- monitoring programs

external measures

- communication with the public
- communication with the clients
- information towards authorities
- environmental reporting.

COMPANY PROFILE

Belgoprocess is a private company, located in Dessel, Belgium, which was established in 1984. It is a subsidiary of the Belgian National Agency for Radioactive Waste and Irradiated Fuel (NIRAS/ONDRAF).

Nuclear energy accounts for more than half of the electricity produced in Belgium. Radioactivity plays a major role in health care and in industrial and scientific applications. As the last link, Belgoprocess has an essential role to play. The company ensures that radioactive waste is processed and stored in such a way that the safety of people and the environment is guaranteed at all times.

It is also extremely important that nuclear installations which are no longer used can be decontaminated and dismantled effectively. Belgoprocess is experienced in this field. On our site in Dessel, the Eurochemic installations are located. This reprocessing plant was closed down in the '70s. Belgoprocess was made responsible for dismantling the installations safely and responsibly.

ENVIRONMENTAL MANAGEMENT PROGRAM

The services Belgoprocess is offering in the field of radioactive waste treatment and storage, decommissioning of nuclear installations, or the sanitation of contaminated sites generate potential risks to the workers, the public and the environment. The company strategy is to keep the risks of harmful effects for people and the environment to an absolute minimum. Therefore Belgoprocess gradually implements a company wide environmental management program (EMP). This EMP consists of a set of combined measures (organizational, administrative, managerial and technological) to minimize the overall environmental impact according to the ALARA-principles and the application of BATNEEC. To stress the importance for an environmental care taking program Belgoprocess has signed the "Responsible Care Act", which has been set up by the Federation of Chemical Industries in Belgium, and which summarizes the management principles toward safety and environmental care taking programs. The text of this act is given in annex 1.

Environmental management is a complex issue. It is an integrated part of the company wide management policy and program which is governed by several external and internal forces. For industrial oriented companies Michael Porter developed a model to better understand and manage company strategies. The company strategy and underlying activities are well balanced taking into account the complex external field of forces, especially for the nuclear industry.

Fig. 1

Political decisions have a major influence on the development of the nuclear industry as a whole. Belgoprocess is acting at the tail end of the nuclear chain. Waste management and decommissioning of nuclear installations are the company domains of excellence. Even in a shrinking nuclear market, excellent services for waste management and decommissioning will be needed, and will play a fundamental role for the overall acceptance of nuclear industry. It is in this frame that Belgoprocess has set up its mission statement which illustrates the essential role of environmental management for the company. Also the influence of the other external forces, namely the economical factors, the social factors and technological factors are well positioned.

Table I

INTERNAL MEASURES

Safety, Quality and Environmental Management Program

Environmental management is not a separate and independent management program. It is part of the company wide management program, where quality and safety management, as well as production management, human resources management, financial management, etc. play an equal role. Company management is a holistic approach and all parts of it are interconnected and influenced by one another. Quality assurance, safety management and environmental protection form the well known triangle of managerial interrelationships.

Fig. 2

Belgoprocess has implemented an overall QA/QC management system. The process of treatment and conditioning of LLW is QA certified to ISO 9001 standards. The process of decommissioning of nuclear installations is prepared to be certified by the end of March 1996. Other processes will be gradually aligned to be certified as well in 1996.

The basic rules of QA can be summarized as follows:

- write how work has to be done
- do the work as written
- prove that work is done as written.

The implementation of a QA system leads therefore to the documentation of all work procedures. Prior to finalize these procedures, it is necessary to analyze and optimize them by incorporating safety analyses and environmental impact evaluation. The QA system driven approach will therefore optimize the company processes.

Safety management, quality assurance and environmental management implemented in this way, enforce one another and leads to an efficient organization. This is the reason Belgoprocess will review the actual QA-procedures in future and integrate measures for environmental impact limitation.

Organizational Structure

Environmental consciousness is a matter of every employee, whatever his role is within the company. Every employee has to consider himself or herself as responsible for reaching the goals the company has set for environmental impact limitation. It is by the combined action of all individual work done that services to clients are offered, and that the safety, quality and environmental impact goals are reached.

Environmental management is therefore not the sole responsibility of the manager. Its the managers duty to set up, implement and control the work being done according to the EMP. Its the employees duty to work according to the rules of the system. Every employee has to know the company goals, the management systems, and its responsibility. According to Belgian law it is not the company but single employees who can be prosecuted according to criminal law for infraction of environmental rules and regulations.

Belgoprocess is structured according to their two main processes: radioactive waste management and decommissioning. Supporting services are formed for administrative work, technical services, safety support and quality assurance. The manager of the safety support service has been appointed as environmental coordinator. He reports directly to the manager. In this way split-up responsibilities are overcome and it is clear that the company stresses the importance of implementing a safety culture in association with an environmental care program. The environmental coordinator is the initiator and controller of the environmental responsible care programs. Their execution is a matter of every employee. A working group is established within the company which

act as a think tank for environmental affairs. The environmental coordinator ensures that all legal aspects applicable for the company are well known, that the company is working according to the rules, and that the results of the responsible care programs are communicated to the authorities.

Internal Communication and Employee Involvement

Effective execution of the EMP supposes an open internal communication network and culture.

The manager has to set clear and stretched goals. This information has to be communicated to all employees. By continually repeating communication about environmental company goals and policies, employees know it is important for the company because the manager puts a lot of effort and time in communicating the message. Data concerning the environmental impact levels, analysis results etc. have to be known by the employees immediately after measurement. In this way employees can evaluate the results of their efforts towards reaching the company goals. This leads to real involvement and foster employees at all levels an individual sense of responsibility for the environment and the need to be alert to potential sources of pollution associated with the operations.

Effective internal communication has a positive impact on external communication as well. Employees are in close contact with the surrounding community and are thus very effective ambassadors.

Technological Measures

Environmental impact limitation is established by applying available technology. The use of suitable or best available technology out of a broad spectrum of available technology is stressed by applying the principles of ALARA and BATNEEC. These principles stipulate that aspects as costs are part of the justification process for the selection and application of technology. Reduction of environmental impact has to be expressed in terms of reduction of costs. The costs of the use of alternative technologies or the application of alternative working procedures for obtaining a lower environmental impact has to be of equal level, without it the alternative practice cannot be justified. ALARA-principle is mainly used for radiological impact optimization, whereas BATNEEC-principle is applied for other industrial sectors or contaminants.

Monitoring Programs

Measuring the performance and efficiency of the applied technology and working procedures is essential in good environmental care taking practices. Knowing the relevant parameters characterizing processes and practices permits the optimization (justified limitation) of residual environmental releases. Exploitation licenses contain release limits for aqueous and gaseous effluents. It is the company strategy not only to operate the technical installations at all times within the imposed release limits, but to excessively reduce the residual releases as long as economically justifiable. The laboratory infrastructure of Belgoprocess and the use of on-line measuring equipment permits the control of technical installations and processes.

EXTERNAL MEASURES

Communication with the Public

As part of management policy special attention is given towards communication with the public. The management principle is to provide the public with the information necessary to enable them to understand the potential environmental effects of the companies' operations and to be

prepared to respond positively to expressions of public concern. The headway made with the cleansing plan for the site and the investment in new equipment and working procedures are the instruments which we are using to establish a new relationship of trust and understanding between Belgoprocess and society. The nature and frequency of communication is adapted to factors such as the nature of the work activities, their impact on local amenities, the level of risk and the level of public interest in information.

The Isotopolis information center was inaugurated in 1993. Isotopolis is a forum from which information on radioactivity and radioactive waste products can be distributed to a wide-ranging public. From the beginning a large number of visitors have witnessed the efforts Belgoprocess is taking as part of the EMP to reduce the residual releases to an absolute minimum.

Communication with the Clients

The same open communication policy is practiced towards our clients. Belgoprocess is achieving transparency and operates as an open book. By strengthening the environmental care taking practices clients have trust in the good functioning of Belgoprocess and the quality services the company is guaranteeing to their clients.

Information Towards the Authorities

On a regular basis the measured residual releases into the environment are transferred to the controlling authorities. Every six months the management of the company informs the authorities about the internal functioning of the company, the planning of future programs, the measures Belgoprocess is taken to guarantee the fulfillment of the imposed limits, etc. On a local basis a contact group with the municipalities is institutionalized and on a regular basis they are informed about the company strategies, programs and results. This intense and open communication process convince national and local authorities that the services Belgoprocess is offering are acceptable on broad basis.

Environmental Reporting

Yearly an environmental impact report is published. The most important results are summarized as follows.

To determine the impact of the activities on the public and the environment, the liquid and gaseous releases and the direct radiation are measured. For these source-terms the federal and regional authorities have fixed limit values. Its Belgoprocess' duty as nuclear responsible operator to assure that releases always are lower than the imposed limits. Belgoprocess is further reducing the releases to minimize its impact as long reasonably achievable. Based on the measured radioactivity the dose-impact is calculated according to a model. For the release of treated wastewater in the river MolseNete the annual dose is calculated being lower than 0,6 mSv for the mean inhabitant. For the gaseous releases the model is based on the impact of the most critical person. The dose-impact for gaseous releases is lower than 0,6 mSv as well. The dose-impact for the mean inhabitant is even lower. The measurements are performed as part of an environmental control program for the surroundings of Belgoprocess. This program is imposed by the competent authorities. The results are independently controlled.

Liquid Releases

In 1994 Belgoprocess released 63615 m3 treated wastewater into the river Molse Nete. This is 15% of the limit value. The measured radioactivity of the released wastewater was 5500 GBq. Activity consists off 99.75% 3H

which can not be captured by the applied treatment processes and equipment. For the other radioisotopes present in the released wastewater, the technical purification limits are reached. The released wastewater contained a weighed radioactivity of 27 GBq which is 1.36% of the limit value. The predominant radioisotope 3H generates only 20% of the weighed activity. b-isotopes which are present for only 0.249% account for 79% of the weighed activity. a-isotopes are present for 0.0006% and account for 1% of the weighed activity. Also for the release of pure chemical substances the actual amounts were only a few % of the limit values. In total only 44 tons of chemical load was released, or 4% of the limit value. The three most important groups of substances were suspended materials and oxygen-binding substances, heavy metals, and eutrophying substances such as N and P. The dose-impact for the inhabitant along the Molse Nete was lower than 0,6 mSv. The release of the wastewater had no negative effect on the biological quality of the river water. The biological quality is expressed in term of the bio-index (a 1 to 10 scale: bad .. excellent). The index upstream the point of release was only 3, but increased up to 6 due to the modernization of the community wastewater treatment station. The index downstream the point of release was 5 to 6.

Gaseous Releases

The release through several chimneys consists off mainly ventilation air. This air is released after effective filtration. All installations keeping or treating radioactive substances are kept in underpressure through a ventilation system. In this way uncontrolled releases into the environment are prohibited.

Also process gases are released. The most important ones are gaseous releases from the evaporation process of high level liquid wastes, from the incineration of solid and liquid radioactive waste, and from the bitumization plant of radioactive sludges. Through the chimney of the combustion furnace, 720 MBq or only 1% of the limit value, was released. For all other chimneys the releases were even lower than 0.1% of the limit values. Through usage of heavy fuel with lower S-content, the release of SOx was further reduced.

Direct Radiation

The storage of radioactive waste on site is a potential burden of direct radiation. Waste is stored and treated in adequately shielded buildings. At the periphery of the premises of Belgoprocess the radiation levels are measured, controlling the affectivity of shielding used. Around site 1 of Belgoprocess radiation levels of 60 to 80 nSv/h are measured, which are natural background levels. Around site 2 radiation levels up to 600 nSv/h are measured. They are due to the presence of radiation sources with limited shielding. New storage buildings have been built and waste packages will transferred in near future.

The following figures illustrate the key data of the environmental report 1994.

Fig. 3

Fig. 4

Fig. 5

FEDERATION OF CHEMICAL INDUSTRIES IN BELGIUM

COMPANY ENVIRONMENTAL AND SAFETY POLICY

- RESPONSIBLE CARE PRINCIPLES -

Belgoprocess, who signed the Responsible Care Act, applies the following principles:

1) To implement an integrated company policy on safety and environment and to apply the guidelines and codes of good practice to assure this policy.
2) To make sure that this policy and its objectives are well known to all employees and that this policy is applied.
3) To take actions to continuously improve the results of the company on safety, health and environmental protection.
4) To gather the necessary data to be able to evaluate the impact of processes, products and activities on the environment, and the health and safety of the public, in order to control possible effects.
5) To make health, safety and environmental considerations a priority in our planning for all new products and processes.
6) To report promptly to officials, employees, customers and the public, information of company-related health or environmental hazards. To oblige the employees to follow up adequate protective and safety measures and to oblige subcontractors to do the same.
7) To inform the public, directly or through the authorities, to allow the public to form its opinion on environmental, health and safety effects of the company activities. To recognize and to respond to the concerns of the public.
8) To counsel customers on safe use, transportation and disposal of our products.
9) To operate our plans and facilities in an manner that protects the environment and the health and safety of our employees and the public, and to assist the competent authorities to arrange corresponding external measures.
10) To extend knowledge by conducting or supporting research on the health, safety, and environmental effects of our products, processes and waste materials.
11) To participate with the government in creating responsible laws, regulations and standards to safeguard the community, workplace and environment.
12) To promote the principles and practices of responsible care by sharing, where possible, experiences with other companies and offering assistance to others.

30-6

ON-SITE LABORATORY SUPPORT FOR THE OAK RIDGE NATIONAL LABORATORY ENVIRONMENTAL RESTORATION PROGRAM

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ABSTRACT

A remedial investigation/feasibility study was undertaken at Oak Ridge National Laboratory (ORNL). Bechtel National, Inc. and partners CH2M Hill, Ogden Environmental and Energy Services, and PEER Consultants have been contracted to Lockheed Martin Energy Systems, performing this work for ORNL's Environmental Restoration (ER) Program. An on-site Close Support Laboratory (CSL) established at the ER Field Operations Facility has evolved into a laboratory where quality analytical screening results can be provided rapidly (e.g., within 24 hours of sampling). Management of the CSL has been transitioned to T N and Associates this fiscal year. CSL capabilities include three basic areas: radiochemistry, chromatography, and wet chemistry. Besides environmental samples, the CSL routinely screens health and safety and waste management samples. The cost savings of the CSL are both direct and indirect. Direct cost savings are estimated based on comparable off-site quick-turnaround analytical costs. Indirect cost savings are estimated based on: reduction of costs and liability associated with shipping for off-site analyses, preparation for sampling, assistance to Health & Safety staff, use of CSL results to focus further sampling efforts, and sampling crew downtime. Lessons learned are discussed.

INTRODUCTION

A remedial investigation/feasibility study (RI/FS) began at Oak Ridge National Laboratory (ORNL) in 1987 for ORNL's Environmental Restoration (ER) Program. Bechtel National, Inc. and partners CH2M Hill, Ogden Environmental and Energy Services, and PEER Consultants are the RI/FS subcontract team. In 1989 the project established the Close Support Laboratory (CSL) to provide rapid radiological (a/b/g) and volatile organics screens on samples to determine DOT classifications before shipment to the off-site CLP laboratory. The advent of the Observational Approach and SAFER led the RI/FS team to shift the main use of the CSL from preshipment screening to screening to help in technical decisions (e.g., delineating the extent of contamination). Basic wet chemistry techniques were added to assist in rapid and cost-effective sample characterization. CSL scope has recently changed further to support other groups performing environmental restoration activities for the Oak Ridge Reservation. The CSL is now managed and staffed by T N and Associates, an 8(a) small business subcontractor. Also, the CSL now has an interface to the Reservation-wide Analytical Services Organization (ASO), to provide integrated analytical service to its customers.

TECHNICAL SUPPORT

The CSL provides the quality, quick-turnaround data needed to support results-based field decision making. Analytical results can be provided to the data user overnight or within 24 hours as required by the customer (currently, a thirty day turnaround time is standard if there is no quick turnaround request). Under Bechtel's contract, CSL staff assisted RI/FS project geologists with planning, interpretation, and application of sampling and analysis plans and associated support documents. The staff have supported ER field efforts with analytical planning, cost estimating, and data interpretation for the past two years.CSL staff interact with various ER project staff to provide pre- and post-field-support activities. These including preparation of sampling kits, sample

screening for DOT transportation/packaging and radioactivity checks, analytical planning and coordination with either on- (ASO) or off-site confirmatory-level laboratories, receiving excess sample returned from these labs, and archiving or disposing of sample remnants (thus closing the chain-of-custody).

Immobile laboratory trailers at the ORNL ER Field Operations Facility (FOF) house the CSL. This location is convenient for sampling teams to pick up sample kits or to deliver samples since the FOF is the starting and stopping point for most ER field activities. We routinely screen environmental, health and safety, low-level decontamination and decommissioning and waste management samples. Our sample screening results are used by ASO and off-site labs to guard against instrument contamination and detector saturation.

ANALYTICAL TECHNIQUES

The analytical scope of the CSL covers basic radiological and volatile organics screening, and basic wet chemistry. Analyses can be performed rapidly, and results from complementary techniques are reviewed to provide a more complete technical understanding. Method detection limits are comparable to off-site confirmatory labs.

Primary radiological contaminants of concern are 90Sr, 137Cs, and 3H. Radiochemical analyses include gamma spectroscopy, tritium and carbon-14 screens using liquid scintillation analysis, and gross alpha and beta counting. Cerenkov counting and crown-ether-based separation are the two rapid methods used for determination of radiostrontium in water samples. Minimum detectable activity values for radiological samples may be adjusted by changing sample sizes and count times to meet the customer's needs.

Gamma spectroscopy is performed via an intrinsic germanium detector with a computer-based multichannel analyzer. Due to the lack of an autosampler and the long count times often required, the gamma detector system is a bottleneck in sample throughput. A second detector will soon be on-line to increase the CSL's capacity.

Liquid scintillation is used to perform 3H and screening 14C analyses. Samples are not distilled; instead, soil samples are DI water extracted (1:1 w/v) and instrumentation software corrects for quenching effects in all samples. Carbon-14 can be excluded based on negative screening results but cannot be confirmed based on positive results (other weak or quenched b particles may cause 'false' positives).

Gross a and b are measured using proportional counters. Low-activity samples are analyzed on a low-background gas-flow proportional counter. Higher-activity samples are analyzed on scalers because higher-activity samples might contaminate the low-background counter, and the ZnS solid scintillator probe is immune to the b->a cross-talk observed in the a signal from the gas-flow proportional counter.The CSL analyzes 90Sr in water samples using one of two methods. Strontium may be separated from unfiltered or filtered samples using SrSpec columns (EiChrom), then immediately counted for 90Sr as gross b before substantial 90Y ingrowth. Alternatively, after a two-week 90Y ingrowth, 90Sr Cerenkov counting may be performed on filtered samples using the liquid scintillation counter (and no scintillation cocktail). Strontium-90 Cerenkov counting also requires gamma spectroscopy to provide 137Cs/60Co correction to the Cerenkov-determined 90Sr activity.

The primary volatile organic contaminants of concern are fuel-based aromatics and solvent-based chlorinated hydrocarbons. Volatile organics

screens are performed by gas chromatography (GC) using photoionization (10.2 eV) and Hall electrolytic conductivity detectors and a CSL-specific method based on EPA methods 601 and 602. A sixteen-port purge-and-trap autosampler introduces samples onto the GC column.

Basic wet chemistry for environmental waters includes alkalinity, dissolved and suspended solids, ion chromatography (IC), and, (for various matrices) pH and resistivity. IC is used to analyze both cations and anions following a CSL-specific method based on EPA 300. Together, IC and alkalinity provide an ionic profile of water samples, and their sum can be compared to the total dissolved solids measurement as a performance check.

QUALITY ASSURANCE

The mission of the CSL is to provide rapid screening (EPA level II) for the ORNL ER program. The lab delivers these results, using lab-specific methods, without time-consuming deliverable requirements. Controlled CSL procedures and the laboratory quality assurance plan document quality requirements for each analysis and general laboratory practices. QA staff from ORNL Oak Ridge Reservation, DOE Oak Ridge, and, formerly, Bechtel, routinely audit the lab's procedural conformance and good lab management practices. The CSL has used independent performance evaluation (PE) samples to fine tune method accuracy. The PE samples were obtained from commercial sources. Recently, the lab has begun to take part in government-sponsored radiological (DOE MAPEP and EPA EMSL-LV) and chemical water pollutant (EPA EMSL-Cinci) PE studies. Participation in these studies confirms the CSL's accuracy and interlaboratory comparability.

COST EFFECTIVENESS/SAVINGS

The CSL is saving dollars both directly and indirectly. Direct cost savings are based on comparable off-site quick-turnaround analytical costs; premium charges for rapid response from off-site laboratories make the CSL especially cost-effective. The RI/FS team has documented CSL savings estimated to be greater than $1 million for fiscal years 1993 and 1994.

Indirect savings are difficult to quantify. They are based on reduction of costs and liability associated with shipping samples off-site for analysis, preparing for sampling and sample shipping, assisting Health and Safety (H&S) staff, and sampling crew downtime. CSL data provides analytical information for proper DOT classification of radioactive environmental samples. Sample container procurement, sample kit preparation, and sample chain of custody are all centralized through the CSL for most samples analyzed by the CSL. CSL staff also generally prepares and packages samples for shipment to off-site or ASO labs for further analysis. H&S staff uses the CSL to analyze monitoring samples to minimize personnel risk, and field sampling crews can be more productive because of the rapid turnaround of data from H&S and sampling based on results of previous sampling. The RI/FS team made extensive use of CSL data in the Remedial Investigation for Waste Area Group 5 at ORNL and other site characterization projects.

LESSONS LEARNED

Several lessons learned at the CSL may apply to similar screening laboratories.

- Participate in the initial scoping or DQO Process activities to identify data uses and opportunities to use CSL data.

Determine a general prioritization scheme for samples and analyses before competing deadlines or customers demand one. This planning should include holding time, data end-use, and lab staffing considerations. Lab customers should be aware of and agree with this scheme.

Establish appropriate sample selection guidelines to identify possible further analyses (e.g.,perform g spectroscopy only when b activity is greater than x) within the screening lab or at an off-site confirmation lab. Setting up a formalized analytical decision tree will save money by reducing unnecessary analyses and documentation requirements.

Invest in an expandable data handling system and integrate data handling into the appropriate project data management plan. Data quality can be undermined by a poor or 'make-do' handling system.

Stagger staffing hours. Varied schedules reduce overtime, improve morale, and serve both the first-of-the-day customers (generally technical staff) and end-of-the-day customers (generally field sampling staff).

FUTURE DIRECTIONS

The mission of the CSL continues under T N and Associates. With the recent appointment of an ASO technical interface, Energy Systems is taking a more active role in CSL activities. This interface is likely to strengthen as DOE analytical strategies become more coherent. Improvements in data handling will ensure seamless electronic data delivery to CSL customers and the Oak Ridge Environmental Information System. As quick-turnaround screening data are more broadly accepted, the analytical capability and sample capacity of the CSL will likely expand.

SUMMARY

The ORNL RI/FS team established the CSL to provide rapid radiological (a/b/g) and volatile organics screens for ER. Basic wet chemistry techniques were added to assist in rapid and cost-effective sample characterization. Today T N and Associates continue that mission. The CSL provides its customers with technical and analytical support, and lessons learned have potential application for similar sites or labs. ER is expanding the CSL's scope to support more environmental restoration/waste management activities at ORNL.

30-7

LOW-RESOLUTION SIM MODE GC/MS ANALYSES REQUIRING HIGH-RESOLUTION GC/MS TECHNIQUES FOR QUALITATIVE COMPOUND IDENTIFICATION

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ABSTRACT

The United States Air Force (USAF) is conducting a site investigation at Air Force Plant (AFP) PJKS near Waterton, Colorado. The compound n-nitrosodimethylamine (NDMA) was identified as a contaminant using gas chromatography/mass spectrometry (GC/MS) selected-ion monitoring (SIM) analysis of soil and groundwater samples. As part of the investigation, several background wells were sampled to determine environmental levels of specific target compounds in groundwater. NDMA is not naturally occurring, but was detected in one of the background wells. Upon reanalysis of the sample by another methodology, it was discovered that an organic acid was a direct interferent of NDMA in the SIM method. The interfering organic acid was also found to be present in some of the

project environmental samples with detected NDMA results. As a result, all positive detections determined by the SIM method were considered suspect. A new method of analysis was needed that maintained the analytical comparability with previous data collected (e.g., low detection limit). It was decided that high-resolution GC/MS would be used to confirm any detections produced by the SIM method. If a result was confirmed by the high-resolution method, then the NDMA result from the SIM analysis would be reported as usable.

INTRODUCTION

A supplemental remedial investigation/feasibility study (SRI/FS) was performed at AFP PJKS, near Waterton, Colorado as part of the Air Force's Installation Restoration Program (IRP) to assess past hazardous waste disposal and spill sites and develop remedial action strategies for cleanup. AFP PJKS activities included rocket fuels development, purification, and testing activities in support of the Titan missile. Hydrazine fuels (hydrazine, unsymmetrical dimethyl hydrazine, and monomethylhydrazine) were used in the testing and development activities. A degradation product of these hydrazines, NDMA, was detected in the soil and groundwater. The approved detection limit of US Environmental Protection Agency (EPA) method SW8270 (EPA,1987) for this compound (10 parts per billion) was considered inadequate in comparison to human health and ecological risk levels for soil and water samples. As a result an alternative method was needed. A modification of the EPA GC/MS method SW8270, for semivolatile organics, capillary column technique, was developed. This method, which involved the use of SIM GC/MS, was reviewed and/or approved by multiple regulatory and federal agencies during work plan development for the investigation. After several years of data analysis using this method, it was discovered that an organic acid present in the samples shares the primary ion with NDMA and acts as an interferent, which leaves much of the data unvalidatable (unconfirmed target compound identification) and therefore of limited use. It was determined that the modified method produced false positives for some sample results for NDMA analyses. A description of the method and resolution of the interference problem are discussed in the following paragraphs.

SELECTION OF ANALYTICAL METHOD

When determining the method for analysis of NDMA at AFP PJKS, detection limit requirements were of utmost importance. The concentrations of concern, as related to ecological and human health risk assessment (12.5 g/kg for soil samples and 0.000362 g/l for water samples), were compared to the detection limits for three analytical methods. Two of these methods, SW8070 and SW8270, are standard SW846 analyses. The third is a modification of SW8270 using the SIM mode of acquisition. Method SW8070 is a GC method that requires second-column confirmation and has estimated method detection limits (MDLs) of 0.15 g/l for water and 1.5 g/kg for soils. This analysis was rejected because second-column confirmation of NDMA detections increases the cost of analysis, and the detection limits were not as low as those provided by other methods under consideration. The detection limit for NDMA analysis by SW8270 also was insufficiently low to meet risk assessment quality objectives. An alternate method with a lower detection limit was required. It was determined that a modification of method SW8270 would produce the lowest detection limits. An explanation of why the SIM mode is a more sensitive modification of the method is given below.

There are two commonly used methods to acquire data from an analysis by GC/MS operating in electron impact (EI) mode. Scan mode, the most commonly used method, acquires information continuously from a signal originating from an electron multiplier over a wide range of ions. These ions have a specified mass, measured in atomic mass units (AMU), which is used to help identify the compound of interest. A scan of the range of ions (usually 35 to 500 AMU) occurs a minimum of one scan per second. SIM GC/MS functions in an identical manner with one exception. The designated scan range for SIM GC/MS is much narrower and is specific to the mass of the primary ion for the compound of interest. Therefore, when acquiring data in the SIM mode, a greater amount of data is collected for a specific ion mass. As a result of greater amount of acquired data over the same amount of time, greater instrument sensitivity is obtained in the SIM mode as opposed to the scan mode, resulting in a lower limits of detection for the target compound. Because the SIM mode produced lower detection limits that were more in keeping with the risk-based concentrations of concern, the SIM mode modification of the SW8270 method was selected as the investigative protocol.

ANALYSIS OF NDMA USING SIM GC/MS AND PROBLEMS WITH INTERFERENCE

Analysis of NDMA using SIM GC/MS involves the following procedure. A measured volume or weight of sample is extracted with methylene chloride at pH 11, and then extracted again at pH 2. The solvent extract is dried with sodium sulfate and concentrated to a final eluate volume of 1 ml. Surrogate spiking compounds are added during the extraction phase, and internal standards are added prior to sample analysis and after extraction. After chromatographic separation, qualitative identification is performed using peak retention time and the relative abundance of the primary NDMA ion, at 74 AMU. Quantitative analysis is performed using the internal standard technique relating the response of the internal standard to the response of the NDMA primary ion (74)(Southwest Laboratory of Oklahoma, 1992).

As part of the field effort for the AFP PJKS site investigation, several groundwater monitoring wells were chosen outside of suspected contaminated areas to measure the background levels of specific target compounds, including NDMA. NDMA was detected at a level of 39 g/l in one of the background wells. NDMA is not naturally occurring and should not have been detected in this well. The sample analysis was confirmed for NDMA when reanalyzed by method SW8270 SIM mode using the two secondary ions (42 and 43) in addition to the primary ion (74). The laboratory was asked to reanalyze the sample in the scan mode, where it was discovered that an unknown compound (an organic acid) coeluted with NDMA. The unknown compound contains, as part of it's spectrum, the same ion abundances of 74, 42, and 43 AMU that were used in the SIM mode to identify NDMA. It also produced a response at 73, 57, 56, and 45 AMU. Therefore the unknown organic acid (believed to be propanoic acid) produced a spectral signature in the SIM mode identical to that of NDMA, and was responsible for the false positive NDMA detection in the background well. Because of the shared ions and the similar retention time of both compounds, the positive detections which had been achieved so far in the investigation immediately became suspect. It could not be determined by the data available which compound had produced the positive results in the SIM mode. The samples that produced no detections were determined to be valid results, because neither NDMA or the unknown organic acid was present to produce a positive detection.

ANALYSIS OF NDMA USING THE SIM MODE GC/MS WITH CONFIRMATION BY HIGH-RESOLUTION GC/MS

The immediate problem was to find a reliable method for analysis of NDMA which would allow a low enough detection limit to be comparable to the concentrations of concern and also allow confirmation of the detection of NDMA. It was also important to maintain the comparability of the data so that the results of the samples that produced no positive detections could be used. It was determined that comparability could be maintained if the SIM mode GC/MS method was used with a confirmation method for positive detections. The confirmation method would distinguish between NDMA and the organic acid.

It was decided to confirm the positive detections with high-resolution MS in the SIM mode. The original SIM mode method measured the response of 74+/- 0.5 AMU. The high-resolution method measured the exact molecular ion at 74.048 AMU specific to NDMA only. When NDMA was detected in the low-resolution SIM mode analysis, a confirmation analysis using the high resolution technique was used. If NDMA was confirmed by the high-resolution analysis, then the original NDMA result from low-resolution SIM mode analysis was reported. If the confirmation method proved that the detected compound was the unknown organic acid, then NDMA would be reported was not detected.

CONCLUSIONS

GC/MS analysis techniques are commonly thought of as confirmational qualitative methods not requiring alternate analyses with dissimilar columns or detectors, as is typical when using GC methods without a MS detector. When utilizing SIM mode GC/MS methods, this may not always be an accurate assumption, as proven by the results for NDMA at AFP PJKS. The use of SIM methods requires knowledge of the project site operational and analytical history. If analytical history is unavailable, site screening of soil and water by full-scan GC/MS methods should be performed prior to the use of SIM methods in order to identify possible sample interferences. Without sample interferences, SIM mode analysis provides a qualitative and quantitative result in the primary sample analysis. Confirming results using high-resolution SIM techniques is an alternative to site screening with full-scan, low-resolution methods. Unique project considerations must be evaluated to determine the most cost-effective approach to this qualitative method-selection issue.

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30-9

STABILIZATION AND CONTAINMENT OF BURIED WASTE SITES USING JET GROUTING OF ACRYLIC POLYMERS*

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ABSTRACT

A full-scale field demonstration of an innovative subsurface stabilization technique was performed on simulated transuranic waste pits at the Idaho National Engineering Laboratory. The demonstration involved jet grouting simulated buried transuranic waste pits with an acrylic polymer. The jet grouting created a monolith out of the buried waste pits. The monolith can either be considered for in situ disposal of the buried waste with improved confinement or can actually enhance the contamination control of the transuranic material during a hot-spot retrieval effort by agglomerating the contaminants into courser less aerosolizable materials. The demonstration involved pit construction, two-component jet grouting for the polymer pit, destructive examinations of the polymer pit, and evaluation of the contamination-control aspects of using the polymer material to bind the contaminants.

INTRODUCTION

This project was sponsored by the Department of Energy Office of Technology Development Landfill Focus Area. It investigated two potential applications for jet grouting: 1) creating monoliths out of the soil/waste matrix for in situ disposal of buried waste, and 2) creating monoliths for interim storage of buried waste for eventual retrieval, treatment, and processing. Both of these scenarios are being considered by the Idaho National Engineering Laboratory (INEL) Environmental Restoration Program as part of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) process. Results of this investigation can be used to establish the Record of Decision for the 2 million cu ft of buried mixed transuranic waste commingled with 6-8 million cu ft of soil in shallow land burial at the INEL.

To accomplish these tasks, two simulated waste pits were jet grouted with two different formulations of an acrylic polymer grout provided by 3M Company, Inc. The grout is an acrylic chemical grout based on an acrylic comonomer blend. Jet grouted into one waste pit was a version of the polymer that produced a hard durable monolith out of the waste. This version of the polymer simultaneously stabilized the waste against subsidence and provided complete waste encapsulation. This monolith could be considered for disposal in situ of the buried waste. For the other pit, a slightly different formulation of the same basic polymer was jet grouted, which resulted in the same monolithic structure. However, the mixture of soil-polymer that resulted was eraser-like in texture and easily retrievable with enhanced contamination-control and dust-control attributes.

The experiment involved pit construction, grouting, retrieval of the soft eraser-like monolith with air sampling, and coring and destructive examination of the hard monolith.

PROCEDURE

Two identical pits (4.5 9 6 ft) were constructed with the same composition as the transuranic pits and trenches found in the INEL Subsurface Disposal Area. The pits contained 15 to 55-gal drums of simulated waste material including cloth, paper, metal, wood, asphalt, concrete, glass, and sludge. Loading of the simulated waste in the drums was typical of those from the Rocky Flats Plant, with a range of up to 50% voids for the metal-containing drums and 10% voids for the sludge

drums. A rare-earth tracer (dysprosium oxide) was placed in each container to simulate the plutonium/americium oxide present in the actual buried waste.

Past studies have shown that the rare-earth oxides move like plutonium under the INEL soil conditions. (1) The two pits were typically backfilled with INEL soil identical to the practice in the actual waste pits. The grouting phase involved jet grouting with the CASA GRANDE C6S drilling/jet-grouting system using a dual concentric annulus drill stem and two positive displacement pumps with balanced flows. This allowed mixing of the two parts of the polymer grout in the ground through a single nozzle at the bottom of the drill stem.

One component of the grout was injected at 6,000 psi using the CASA GRANDE JET5 pump, and the other component was injected at 1,000 psi using the Schwing pump. A typical grouting operation was to deliberately drive the 4-9/16 in. drill stem through the waste. When fully inserted, the procedure was to withdraw the drill stem while jet grouting. The test variables were withdrawal rate, drill stem rotation, pump pressure, and withdrawal step size. Based on field trials, the step size was set at 3 cm, with 2 revolutions per step, and 3 s on each step. This resulted in placing nominally 65 gal of combined polymer material into each of the holes. Also based on field trials, the holes where placed at 2-ft spacing on a closed-pack triangular pitch.

The retrieval phase of the soft eraser-like monolith involved placing a weather structure over the pits and using air samplers during retrieval to evaluate the contamination control by measuring collected dust and tracer. Retrieval was accomplished with a backhoe with thumb attachment. The hard monolith was evaluated by coring in two locations and by destructive examination (isolation and uncovering of the monolith). Laboratory durability studies were also performed involving the hard monolith for compressive strength, effect on compressive strength due to base attack, water immersion testing, and permeation of liquid phase volatile organics.

RESULTS

The demonstration involved jet grouting, retrieval (soft polymer pit), and destructive examination (hard polymer pit). The main objective was to prove the concept for this technology.

Grouting

Grouting was performed in 1 working day for both the soft polymer pit and the hard polymer pit. A total of 33 combined holes were grouted into the two pits (18 holes in the hard polymer pit and 15 holes in the soft polymer pit) as shown in Fig. 1. Grout returns (a mixture of soil and uncured polymer mix that flows up the drill stem during grouting) were considerably higher than returns previously experienced when jet grouting lower viscosity Portland cement/water mixtures. During a Fiscal Year 1995 demonstration called Innovative Grout/Retrieval, (2) jet grouting with Portland cement produced about 2gal of returns per 10-ft hole, and these returns were easily contained in a spoils collection hole around the drill stem.

Fig. 1

For the Fiscal Year 1995 polymer demonstration, 5 to 7 gal of returns were observed in each hole, and a special collection pit and canal system were employed to manage this secondary waste material. Figure 2 shows the jet-grouting operation, with the spoils collection pit and canal system to divert the spoils into the collection pit. Cure time in laboratory

mixtures of polymer and INEL soil was about 70 minutes. Therefore, the jet-grouting operation had to be sensitive to timing to avoid premature curing in the injection equipment. Premature curing of the polymer material was not observed during grouting, and the jet-grouting apparatus performed even better (less breakdown time) than with Portland cement, primarily because of the lower viscosity material. Since the jet-grouting operation used a dual concentric annulus drill stem, any mixing of the two components occurred outside the injection apparatus.

Fig. 2

Retrieval (Soft Polymer Pit)

Once grouted, a weather structure was placed over the pits and 10 high-volume (25 cfm) air samplers were oriented around the pits. With no operations inside the weather structure, dust and tracer concentration data were obtained (8.42E-7 g/L-air; less than 30 ppm dysprosium on the filter). Next, approximately 3 ft of overburden was removed while taking air-sampling data to establish a baseline digging scenario for comparison with digging in the soft polymer pit. Comparison of the dust concentration for digging of soil with no polymer verses excavating the pit with polymer gives an assessment of the degree of dust and therefore contamination spread data.

For the overburden removal case, the dust concentration was 1.21E-5 g/L-air and the dysprosium concentration averaged 20 ppm. A dust concentration of 1E-5 g/L-air is comparable to digging in a soil condition during the excavations discussed in the innovative grout/retrieval demonstration report.2 During digging of the soft polymer pit, the measured dust concentration was considerably lower than the baseline digging case. For the polymer pit, the dust concentration averaged 1.11E-6 g/L-air compared with 1.21 E-5 g/L-air for overburden removal. The dysprosium concentration averaged 3,397 ppm, or a two-order-of-magnitude increase in the concentration during overburden removal. Reduction in dust concentration for digging in the polymer pit compared with digging in just soil was 91%. To put this into a radiological perspective, if the source term were 450 nCi/g (estimated as an average concentration at the INEL buried

At the INEL, bubble-suited entry above about 10 nCi/g would require special waivers. The two-order-of-magnitude increase in dysprosium concentration for digging the polymer pit compared with either background or overburden removal is attributed to the presence of an ungrouted drum punctured during retrieval. The drum was a metal container interred on the edge of the pit. During backfilling, the drum moved outside the surveyed pit boundaries and was missed in the jet-grouting operation. In an actual pit retrieval, this event would not be a problem, because the area of grouting surrounding a hot spot or actual pit would extend up to 10 ft beyond the area excavated, which would avoid the ungrouted drum scenario. It is unlikely that with a 2-ft grid spacing, a drum in any orientation would go ungrouted.

Examination of the debris retrieved from the soft polymer pit showed that the polymer did cure and was easily retrievable. The pit came apart with standard backhoe digging as if the soil/waste matrix was merely wet clay. Examination of the debris showed that unlike the Portland cement pit described in past studies, (1) the individual waste forms were coated in a cured polymer material. Especially impressive was the paper material that had the appearance of having been first soaked with the polymer and then cured. The Portland cement would not have penetrated the paper.

Figure 3 shows a drum containing paper impregnated with the cured polymer.

Fig. 3

Examination (Hard Polymer Pit)

The hard polymer pit was first cored with 2-in. coring equipment and then isolated and finally destructively examined. Results of the two cores obtained show that the monolith was free of voids and that the polymer material did indeed cure while buried. For the destructive examination, the weather shield was removed and the monolith was isolated (9 4.5 6 ft). The structure maintained the monolithic shape. The southeast corner was accidently removed as one piece about 3 ft long by 4 ft wide during the isolation. The monolith was lifted from the pit as one unit with a 7-yard front-end loader but broke along the lines of the waste containers into three distinct pieces when placed on the ground. Based on visual observations, this hard polymer may have resulted in a considerable reduction in the spread of dust and contaminants during retrieval mainly because the waste can be removed in larger pieces.

Durability Tests

Representative samples were subjected to a set of scoping performance and durability tests to determine the adequacy of the polymer for in situ treatment of INEL buried waste. Testing included measuring hydraulic conductivity, resistance to immersion in water, resistance to immersion in a saturated aqueous solution of trichloroethylene (TCE), resistance to immersion in alkali, and resistance to wet-dry cycling. These resistance tests were designed to evaluate the effect of the test on compressive strength, which is an indicator of durability.

Hydraulic Conductivity

Hydraulic conductivities were measured using a flexible wall permeameter following American Society for Testing and Materials (ASTM) D-5084. The hydraulic conductivities of duplicate samples were found to be below the detection limit of the permeameter. Ignoring leakage and evaporation and using only the inflow volume results in a maximum hydraulic conductivity of 1.210-14 m/sec (1.2 10-12 cm/sec) for the brass blank and an average of 2.8 10-14 m/sec (2.8 10-12 cm/sec) for the polymer-soil grout specimens. This indicates a low leaching material with hydraulic properties superior to Portland cement soil grout.

Permeation of Volatile Organics

Original plans called for determining the effect of aqueous solutions of TCE on the hydraulic conductivity of the polymer-soil grout. It was postulated that the TCE might swell clay particles in the soil and further reduce permeability. Because of the low permeability of the specimens, it was impossible to perform this task.

Baseline Compressive Strength

Ten replicate samples of the soil grout were tested for unconfined compressive strength (ASTM C-39) to determine baseline strength. Average compressive strength was 20.7 0.9 MPa (3,000 130 psi).

Water Immersion

It is expected that stabilized buried waste may be exposed to aqueous environments from percolate water. The Nuclear Regulatory Commission (NRC) takes the position that all radioactive waste forms must be able to withstand at least a rudimentary water immersion test. Exposure of barrier materials to aqueous solutions can result in swelling, cracking, or dissolution. Also, soils (i.e., expansive clays) used as aggregate may interact with the water through swelling or dissolution of mineral

components. Aggregate interactions can cause internal stresses with resultant cracking and degradation of the waste form.

Following the testing procedure recommended by the NRC, five replicate samples of the acrylic/INEL soil grout were immersed in deionized water for 90 days. The samples were inspected visually for cracking and swelling periodically during the course of the test. The samples were weighed, measured, and destructively tested for compressive strength (ASTM C-39) at the end of the test period. No visual, dimensional, or structural changes were observed. The average compressive strength was 19.6 0.3 MPa (2,840 40 psi), which shows a very small change from the baseline compressive strength of 20.7 MPa.

Wet-Dry Cycling

The INEL is located in a semiarid region. The soil moisture content is low, ranging between 3 and 15%, but occasional precipitation percolates the ground. The waste form will be under saturated conditions during these times. Wet-dry cycling has a severe impact on construction materials such as hydraulic cement. Five replicate samples of the soil grout were subjected to wet-dry cycling adhering to ASTM D-4843. No dimensional changes were observed. The average compressive strength was 22.9 1.3 MPa (3,320 184 psi) compared with the baseline of 20.7MPa.

Base Attack

At the INEL buried waste site, waste forms will be expected to see alkali conditions. These conditions can arise from the pH of the soil (~8.4) due to the carbonate content, or the buried waste itself can be alkali (e.g., sodium hydroxide used to neutralize nitric acid from reprocessing). Five samples were immersed in an aqueous, sodium hydroxide solution at pH 12.5 for 90 days. The average compressive strength was 16.2 0.6 MPa (2,340 90 psi) for a change in compressive strength compared with the baseline of 20%.

Volatile Organic Attack

Five replicate samples of the acrylic polymer-soil grout were immersed in water saturated with TCE at room temperature for 90 days. TCE was chosen as the candidate medium for volatile organic compound solvent testing because it appears to be one of the predominate contaminants found throughout the DOE complex and it is expected in the INEL buried waste site. The average compressive strength was 19.6 0.3 MPa (2,850 40 psi), which compares favorably with the baseline of 20.7 MPa.

CONCLUSIONS

Based on the results, it was concluded that the two-component acrylic polymer grout can be jet grouted with the dual concentric annulus drill stem and jet-grouting equipment. The resulting monolith is a durable, stabilized matrix suitable for either long-term disposal of buried transuranic waste or interim storage and eventual retrieval. Indications from durability testing show that the polymer/waste/soil matrix is stable for a variety of conditions and is chemically compatible with the environment at the present time. If conditions do not drastically change, the monolith of soil/waste/polymer should remain stable.

During retrieval operations, for the soft polymer pit there is a 91% reduction in dust spread but a two-order-of-magnitude increase in the tracer concentration in the air compared with digging in soil only, suggesting that plutonium might have spread for this retrieval scenario. On a qualitative basis, however, it appears that the hard monolith material may prove to be a superior material compared with the soft polymer for the interim storage/retrieval idea, in that the monolith is

more cohesive. Nevertheless, further testing of that concept for top-down digging would be required to answer that question. In general, the concept of jet grouting a two-component acrylic polymer mixture was demonstrated, and there is a positive proof of concept both for leaving the waste for final encapsulation and for interim storage and eventual retrieval.

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30-10

DETERMINATION OF THE BUILD-UP CAPACITY OF SOME INDIGENOUS BRYOPHYTA IN QUEBEC

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ABSTRACT

An interest in using bryophyta as bioindicators has recently emerged considering their great capacity for accumulating pollutants, more specifically metallic ions from their environment. For example, Phacelia sericea has been used to detect the presence of gold in British Colombia (6). However, few studies on aquatic mosses have been undertaken, until now, in Quebec. The aim of this project is to assess the metallic ion content of natural bryophyta from different regions of Quebec. To realize this objective, -calcium, magnesium, copper, cadmium, manganese, iron, nickel, cobalt, and chromium- were analyzed in a number a species. The bioaccumulated load seems to differ depending on the type of metal analyzed, the species, and the part of the plant studied. The structure of most bryophyta is quite simple; no roots, a very simple vascular system and a single thickness cells in their leaves. These plant cells possess many negatively charged sites where the positive metal ions can attach, hence the rapid accumulation that is observed initially which appears to be centered on an ionic exchange (adsorption) process. Our results show that mosses coming from the Cach River near Quebec City are highly contaminated with cobalt. As for copper, this element is present in high concentrations at St-Alexis-des-Monts, Loretteville and around Quebec City. Fontinalis dalecarlica Schimp. ex B.S.G. and Platyhypnidium riparioides (Hedw) Dix. tended to accumulate more than the other species; thus indicating a species dependant selectivity. In comparing two mosses from Loretteville, P. riparioides contained 300 times more copper than Brachytecium rivulare. Except for calcium and magnesium, our results of the metal concentrations found in the stems and leaves correspond with those mentioned in the literature (9,4,8).

INTRODUCTION

Our aquatic environment is becoming more and more stressed by the various pollutants generated by modern industries. One of the main problems associated with the analysis of heavy metals is their very low concentration in the natural waters. In such cases it is often necessary to use bioindicators which concentrate the metals in order to monitor their evolution over time and space (1). To date a variety of bioindicators have been used and depending on the chemical substances, the efficiency will vary according to the type and complexity of the biocaptor chosen (2). Since 1970, the bryophyta are being studied for biomonitoring and assessment of water pollution. Up to now, most of the research is oriented toward the selectivity of the best indicators. Many scientists have been studying the physiological processes in plants for a better understanding of the metal ion interactions. Mosses are very simple plants that have no root structure. Their vascular system is not complex and suggests an easy accumulation process. The metal ionic exchanges reported are principally centered on the stems and leaves. According to some studies undertaken in Quebec, it seemed important to understand the metal accumulation and elimination mechanisms of these bioindicators. In order to use this kind of biomonitor in an assessment program, it is important to know how this organism lives and the capacity it has for concentrating heavy metals. Moreover, it is interesting to compare the metal accumulation by different plants as well as by their different parts which reflects the condition of the plants' natural environment. The capability to accumulate specific toxic substances is generally determined by the species. However, the selective sampling of mosses is not easy because many species often cohabitate on the same substrate. In addition, their identification is not simple since there are only a few specialists in Quebec who can do the identification.
The objectives of this study are 1) to find and to study an efficient and a resistant biomonitor for metal monitoring in aquatic environments 2) to verify the efficiency of the different parts of the plant for accumulating metals in the natural environment and 3) to characterize different regions of Quebec according to the metals thus bioaccumulated.

MATERIAL AND METHODS

Sampling site

The mosses were collected in the summer of 1993 from various rivers of the province. Five sites were found favorable for the sampling of the mosses: La Rserve Mastigouche near St-Alexis-des-Monts, La Rserve des Laurentides near Quebec city, Val-Jalbert near Roberval , Windsor and Loretteville. Many mosses live totally submerged and are attached to the bottom rocks by their rhizoides (a type of small hook on the end of a semi-root) while others are semi aquatic.

Table I

Here, only B.rivulare are considered to be a semi aquatic bryophyta. This moss lives in or outside of the aquatic environment.
Once the mosses were found, they were collected by cutting the top two thirds of the plant rather than tearing them from their anchorage so as to preserve the survival of this species. A serie of successive washings helped remove the unwanted debris (invertebrates, sand, stones, etc.). Then the samples were placed on ice and transported to the laboratory for the acclimatization period. It is well known that mosses harvested during the summer season will have less difficulty adapting to laboratory temperatures (18-20C) than those collected in the fall (3).

The physico-chemical parameters (temperature, conductivity, dissolved oxygen, pH, and oxydo-reduction potential) were measured with a Hydro-Lab Surveyor II (SVR 2 model).

The conservation of the muscinal biomass necessary for the experimental stage was done in rectangular basins made of fiber glass (1 x 2.5 meters). The water temperature was controlled by a glycol filled serpentine which cooled the water in an adjacent container. This temperature adjusted water was circulated by the continuous pumping action of pressurized air in the return pipe. A Fluorescent "Grow Lite" furnished sufficient light to maintain the aquatic mosses in good health (7) .

Chemical Analysis

Each sample was subdivided in five replicates. Each subsample was oven-dried at 110C for 12 hours. The mass of each subsample was approximately 0.3 grams dry weight. After weighing, the samples were digested in concentrated nitric acid (HNO3) for 7-8 hours until the elimination of the brown fumes; the reaction was driven to completion with 30% hydrogen peroxide (H2O2) (2). When the reaction was finished, the reaction tubes were cooled and maintained at 4C. The digestions were diluted to exactly 50 mL each with distilled water. They were then manually stirred to assure a completely homogeneous solution. Before proceeding to the spectroscopic analyses, it was very important to allow the solutions time to attain room temperature, otherwise a physical interference was evident in the subsequent analysis. The digested mosses were analyzed by atomic absorption spectrophotometry (Varian model AA1275) with air/acetylene flame. Certified standards for water (Standard Reference Material 1643 C) and mosses (Certified Reference Material BCR No 61) were used in the quality control scheme. The digested mosses were analyzed for their Ca, Mg, Cu, Cd, Mn, Fe, Ni, Co and Cr content.

RESULTS

Fig. 1

In Fig. 1, the cobalt concentration in F. dalecarlica (stem) is 138 gg-1 when coming from Quebec area (Cach River) while that at St-Alexis des Monts is only 74 gg-1. For the other metals, the concentration varying from 0 to 20 gg-1. No significant concentration was found for chromium.

Fig. 2

Examining the histogram of the abundant metals, the copper concentrations was elevated in seven of the cases. The highest concentrations was found in F. dalecarlica and P. riparioides . For the latter species, the value is around 24 000 g-1 while the other cases varied from 7 400 to 15 000 gg-1 (Fig. 2). The magnesium and calcium in the five sites varied according to the species and the site, but all concentrations are below 3500 gg-1.

Table II

The cobalt concentration for all the mosses was the highest for F. dalecarlica from the Cach River near Quebec City (81 gg-1) and from St-Alexis-des-Monts (58 gg-1) while the copper concentration was the highest in P. riparioides at St-Alexis (23 589 gg-1) and Loretteville (24 331 gg-1). For other species studied, the copper values presented a irregular dispersion pattern from 44 to 12 261 gg-1 (Table II).

Table III

Generally, the concentration of the metals in the stem or in the leaves attained a higher value than that found in the whole plant depending on the type of metal. For example, F. dalecarlica from St-Alexis-des-Monts

had the highest values in the leaves for calcium (3 248 gg-1), copper (17 205 gg-1) and magnesium (23.75 gg-1) while it appears that the stem had the highest values for the cadmium (3.62 gg-1), cobalt (74.61gg-1), iron (23.36 gg-1), manganese (19.29 gg-1) and nickel (6.80 gg-1) concentrations. A similar pattern was founded for F. dalecarlica from the Cach River near Quebec City.

DISCUSSION

The aquatic mosses are good bioaccumulators of metals and as such can indicate their regional distributions. The results obtained from this study illustrates the capacity of mosses to accumulate well copper ions. In addition, the comparison of the accumulation results shows that Fontinalis is a better accumulator of cobalt than the other aquatic species. Mosses coming from the Cach River near Quebec City are contaminated by cobalt. Oddly, the Cach River is located in a provincial park (Le Parc des Laurentides, Qubec, Canada); one must presume a natural mineral source of cobalt exists upstream.

The bioaccumulation capacity of the different parts of the plant is an important factor if they are to become official bioindicators. For example, the metal ions are more concentrated in the stems rather than in the leaves except for calcium and magnesium. Our results are consistent with other reported studies except for calcium and magnesium (8). It will be important to consider the part of the plant in order to apply this method in different countries. Except for calcium and magnesium, Brachythecium is only slightly contaminated by environmental metals ions. This species is not as useful for monitoring the aquatic environment. We presume this low accumulation potential is due to its semi aquatic nature. The "aquatic" mosses appear to be better accumulators of metals than the "semi aquatic" ones. This result is due to the ionic exchange process between the organism and the water.

Aquatic mosses can store a considerable amount of metallic pollution. Generally, they accumulate a large amount of pollutants on their cell walls, whereas only a small portion of these non-essential metal ions are being absorbed into their cytoplasm. One exception appears to be the chromium ions which are effectively excluded from the inside of bryophyta.

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DESIGN, DEVELOPMENT, INTEGRATION AND TESTING OF CONTROL SYSTEMS FOR A SELF-GUIDED TRANSFER VEHICLE AND REMOTE EXCAVATOR FOR COOPERATIVE BURIED WASTE RETRIEVAL INTEGRATED DEMONSTRATIONS

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ABSTRACT

This paper describes the design, development and implementation of control systems for a Self Guided Transfer Vehicle (SGTV) and a Remote Excavator (REMEX) used for cooperative buried waste integrated demonstrations with a remotely controlled gantry system. Demonstrations were carried out at the Idaho National Engineering Laboratory (INEL) cold test pit site in August 1995. The INEL test site provided simulated Low Level Transuranic Wastes (LLTW) and facilities for demonstrating remote operations from workstations in a cooperative manner for command and control of the remotely operated systems. These systems were developed in support of the integrated approach to retrieval and were part of a contract with INEL to further develop the technologies from the equipment developed, integrated and demonstrated in 1994.

The retrieval of Transuranic waste from landfill type pits/trenches using remotely controlled systems is being considered as part of the Department of Energy (DOE) plans to remediate sites such as INEL's Radioactive Waste Management Complex (RWMC). Over 65,000m3 of TRU wastes are stored in the RWMC in the form of barrels, boxes, large objects and organics.

The purpose of these demonstrations has been to expand on the successful FY94 demonstrations using the REMEX with an Innovative End Effector (IEE) to remotely excavate simulated LLTW and to perform cooperative exchanges with an Integrated Transfer Module (ITM) to the SGTV for remote conveyance of retrieved waste. To accomplish this the Telerobotic Transfer Vehicle (TTV) control system was converted into a self guided vehicle by incorporating a navigation control system combined with GPS and compass. In addition a path planner and high level controller were added at the remote work station to implement self guidance from the operator station.

INTRODUCTION

To upgrade the TTV (2) to an SGTV (Fig. 1) required significant technology enhancements to the TTV control system by the addition of more

processor boards, a fully integrated dead reckoning sensor suite, new collision avoidance sensor suite, GPS interface, compass, new software and a task/path planner control system in a remote workstation.
Fig. 1
For the REMEX (Fig. 5) a change in the innovative end effector size and reconfiguration front shovel to back hoe required retuning of the control system.
The purpose of the integrated demonstrations at the INEL through August and September, 1995, was to verify performance of the SGTV self guidance control system in representative cooperative retrieval demonstrations with the REMEX and as appropriate with a Gantry crane system.

PROJECT DESCRIPTION AND OBJECTIVES

The design, development, integration, testing and evaluation of the control system for the SGTV was funded by the DOE's Office of Technology (OTD) through the Buried Waste Integrated Demonstration program managed by the INEL. The program was contracted by the INEL through RAHCO International to Spar Aerospace in conjunction with RSI Research Ltd. The control system for the REMEX was developed jointly by Spar and RSI in conjunction with the University of British Columbia from 1992 through 1993. Enhancements to the control system to meet the needs of this program were undertaken by Spar Aerospace with funding by the OTD for end effector modifications and the addition of extra cameras.
The objectives of this program were to demonstrate self guided control of the SGTV, the development of a mission planner with user friendly Graphical User Interface (GUI) to permit creation of SGTV paths, initialization and alignment with GPS derivatives and to create a field of safe operational zones for high level control of the SGTV. These objectives were to be demonstrated in conjunction with the REMEX in representative cooperative retrieval of simulated LLTW. The FY94 demonstration program had successfully demonstrated the concept of remote excavation and dust free dumping between the REMEX and TTV.

SELF-GUIDED TRANSFER VEHICLE SYSTEM

Control System Development

The SGTV control system (Fig. 2) was developed from the TTV controller which was designed with the "Hooks" and "Scars" for upgrade to self guidance in place. The SGTV system thereby retains the telerobotic TTV (manual) control mode as the first failure level recovery of the vehicle.
Fig. 2
The control system for the TTV was developed by Spar Aerospace and RSI Research Ltd in the first instance at a higher level to derive the most cost effective solution that met schedule and technical requirements for a self guided vehicle that was ultimately to operate within a containment building.
The earliest concepts by RAHCO International for remote conveyancing evolved very rapidly from a static loader container transfer station and straddle type crawler to a skid mounted container towed behind the excavator. Robotic transfer was assumed to be achieved by having the SPAR/RSI team adapt a front end loader with a self guidance control system for the purposes of the demonstrations.
The SPAR/RSI team proposed a reconfiguration to a dedicated vehicle using an available track set and building these from the skid concept together with an available diesel power unit for propulsion. Transmission and actuation of the TTV functions were developed from standard hydraulic system components typically used in mobile machinery. Hence the basic

vehicle represented low risk established technology to which it was required to add a telerobotic control system.

Functionally it was required that the system be designed to provide the following performance criteria: Control and maneuverability of vehicle within +/- 1 ft of the desired track path; ability of the vehicle to turn on its own centre and within a 20 ft wide roadway; variable speed control from zero to a maximum of 300 ft/min; capable of detecting obstacles in the path of the vehicle and providing a warning indication to the operator as well as slow down or stop the vehicle; operation and status of the Transfer Container lid; sensors status and displays at the operator's workstation.

Proposed Vision Tracking System

To implement self guided control the original TTV control system had been designed to accommodate growth to self guidance using a vision tracking system. The proposed vision tracking offered a completely external wrap around controller with no changes to the original TTV control system. This therefore would provide a completely separate system that was independent of the vehicle and required no inputs, calibrations, initializations, resets or loss/interference from EMI and/or satellite constellations. Vision tracking determines position, rate, collision detection and visual oversite of vehicle at all times. Later it was determined that the demonstrations would be conducted outside with many obstacles and that use of Differential GPS (Global Positioning System) was preferred.

Self-Guided Control System

Self-guided control is divided into two stages: Mission Planning (MP) and Vehicle Control and Command Generation (VCCG). MP is an off-line operation which allows the operation to plan a path. A path (Fig. 3) consists a start point, pause points, flyby points and an end point. Each point is assigned with two circles, which are either used for smooth acceleration or deceleration. The speed limit for each segment of the path can also be defined. VCCG is executed in real time to control the vehicle to follow a planned path based on the feedback of the sensed position and heading of the vehicle provided by GPS and dead reckoning. VCCG generates linear rate command to move the vehicle towards a target position and angular rate command to correct both the off-track error and heading error. VCCG also has a simulation capability which allows the operator to simulate the vehicle tracking prior to an actual run. To allow the operator interacting with the tracking operation, a Graphic User Interface (GUI) was implemented. Through the GUI, the operator can pause, resume, reverse or stop the vehicle at any time.

Fig. 3

In the new SGTV control system, the telerobotic mode and Direct Track mode are still retained from TTV developed in FY94. Both control modes use a two-axis joystick to generate vehicle commands. The difference is that the Direct Track mode is an open loop control with the sensory feedback for the track servos disconnected. Also retained is the lid/latch open and close control which can only be used when the vehicle is stationary.

The control system architecture is shown in Fig. 2. The sensory units consist of a Trimble 4000SSI differential GPS, a Precision Navigation TCM2 electronic compass, a Systron Donner Rate Gyro, dual track encoders and 18 acoustic range sensors (9 at each end of the vehicle). These sensors were connected to the onboard computer system. The onboard

computer system comprises of five RSI Research Ltd. microcontroller units. Each microcontroller is dedicated to the following functions: Network between microcontrollers, servo controllers of the vehicle, camera control, obstacle detection and avoidance, and navigation algorithms such as dead reckoning. Two color cameras, with weather-proof enclosures and pan and tilt units, are mounted separately at each end of the vehicle. A control pendant is available for local operation.

The remote control room has one workstation for Self-guided operation and one for Teleoperation. The Self-guided workstation consists of Pentium Personal Computer system. Whereas the Teleoperation workstation consists of a microcontroller based computer, a control panel and a telemetry radio. The control panel consists of a two-axis joystick, switches for mode selection, camera control, lid open/close and latch/unlatch, emergency stop, keypad and LCD display. A color monitor displays the vehicle's camera's view.

A spread spectrum radio modem provides the telemetry communication between the onboard computer and control station computer. A redundant, dedicated radio controls the emergency stop of the vehicle.

A dead reckoning system is implemented to solve the 1.5 seconds latency in the GPS feedback. Sensor data from rate gyro, compass and track shaft encoders are used to estimate the incremental position. A final best estimate of the vehicle position is calculated by combining weighted GPS data and dead reckoning estimate. Dead reckoning also estimates the vehicle heading using the above data.

An Obstacle Avoidance System (OAS) was implemented in a dedicated microcontroller. OAS has nine acoustic range sensors mounted on each end of the vehicle. The sensors can detect objects within about 35 feet. The obstacle ranges are divided into three zones: detection zone, collision zone and E-stop zone. In the detection zone, the operator is alerted but no action is taken. In the collision zone, the vehicle speed is reduced gradually as it approaches the obstacle. In E-stop zone, the vehicle is stopped automatically.

System health monitor was also implemented with sensors installed on the vehicle to keep the operator advised of the control status and to monitor important system health parameters. System health monitoring includes: engine oil pressure, engine temperature, engine speed, battery level, hydraulic oil temperature, onboard electronics temperature, base latch status, container lid status, GPS status, electronic compass status, data network status and telemetry status.

Test Results

Factory acceptance testing of SGTV was held at the Idaho Falls National Engineering Laboratory (INEL) in September 1995. Three paths, from simple to complex, were used for testing the performance of the Self-guided mode. Each of the path was repeated ten times to determine the repeatability as well as the off-track errors. The vehicle was on average able to track the planned paths within +/- 1 foot error in steady state. The final docking always achieved a position error of less than 1 foot and the heading was locked on the target orientation. The errors might increase when the control system encountered GPS error, telemetry error, compass error or network error. Once the errors were cleared, the vehicle would return the planned path gradually. There were several causes for errors encountered in operation: vibration of the vehicle, electromagnetic interference from other systems operating in close

proximity and GPS loss/float. The maximum commanded rate achieved under self-guided mode was 150 feet/min.

Normally, the telerobotic mode could be operated at higher speeds, up to 300 feet/min, than the self-guide mode, mainly because the telerobotic loop has shorter time delay. As expected, it had much higher off-track errors, which could exceed 5 feet easily, and required cognitive attention by the operator to generate rate commands to navigate and monitor the tracking.

Due to the high sensitivity and resolution of key sensors set up and calibration was found to be one of the most difficult tasks. For example the compass was sensitive to local magnetics, vehicle and vibration during operation. Due to metallic body of the vehicle, the compass's bias varied with the heading and therefore required non-linear calibration. When the vehicle was moving, the heading was determined by the weighted compass output, GPS output and dead reckoning estimate from gyro and rate sensors. The OAS was able to detect obstacle within the specified zones. The vehicle would slow down or stop according to the obstacle range. With a sharp corner of an object facing the vehicle, OAS was not able to stop the vehicle before hitting the object because acoustic signals were reflected away by the sharp corner.

The SGTV GUI and an example of the test runs are shown in Fig. 4.

Fig. 4

REMOTE EXCAVATOR SYSTEM

The control system (Fig. 4) of the REMEX was designed for installation on a conventional excavator and in this instance the Hitachi EX200LC (Fig. 5) was selected as this had already been developed by Spar Aerospace and RSI. The control system implementation described is generic and therefore applicable to all existing excavators as these use similar systems.

The existing manual controls for the Hitachi comprise a right hand side joystick for boom and bucket and a left hand side joystick for the arm and swing. Travel is derived from right and left pedal controls. These manual controls provide outputs to pilot valves that provide signals to control valves operating the hydraulic motors and cylinders.

The modifications to the excavator were designed to demonstrate remote, or in the cab, coordinated motion control of the bucket, stick , boom and swing functions. The travel function was not included with these but was provided separately with manual remote controls.

Remote coordinate motion control of the bucket is implemented using the following items:

4-Degree of Freedom (DOF) handcontroller; control console and VME CPU (VM30 running 68EC030 Mp); ruggedized Vehicle mounted VME CPU (as above); Bucket, Stick, Boom & Swing sensors; vehicle cabling systems and telemetry cable for remote operations.

Inputs from the handcontroller are processed in the workstation CPU according to mode selected and then fed to the vehicle CPU to perform the inverse kinematics for coordinate control of the bucket. Joint sensors data are monitored and fed to the vehicle CPU to derive joint angle and rate for closed loop control of each joint. Joint and engine sensors data are also fed back to the remote workstation for status and health monitoring. These functions are described in more detail as follows:

REMEX Operator's Workstation

The remote workstation console provides system power up, enable and safety controls. Track control levers are provided for travel under remote control. System status is provided by LED's for remote power and

enable, engine oil pressure, alternator and temperature. A mode switch is provided to select the following modes: Off-hydraulics disabled; coord-vehicle operates in coordinate motion; joint-vehicle operates in joint control; manual-vehicle operated joints open loop; remote-C, J&M modes operated remotely via telemetry cable.

The handcontroller provides operator inputs for control of the excavator in any of the modes and can be located remotely at the workstation or in the cab so that the operator always uses the same control input at either location.

Fig. 5

Handcontroller inputs are processed by the workstation CPU to derive command outputs to the vehicle CPU via the telemetry cable in a similar manner to a master/slave manipulator. The design is rugged and had been field proven for control of machinery in the forestry industry.

Modes of Control

Coordinated Motion Control

In coordinated motion control the operator uses the 4DOF handcontroller to control all of the functions of the excavator. The handcontroller provides an intuitive correspondence between the motion of the handcontroller and the bucket at the designated point of resolution (POR).

Joint Control

In joint motion control the deflections of each degree of freedom of the handcontroller are reproduced by each of the joints of the excavator. The rates of each joint are scaled in direct proportion to the deflection of each DOF of the handcontroller. In this mode of control the joint velocities are independent of load and kinematic configuration as hydraulic pressure is modulated in each actuator.

Manual Control

In manual control, as in joint control, the deflections of the joystick correspond to motion of the corresponding joints. However in this instance the deflections are fed directly without feedback from the joint sensors.

The system provides diagnostics features via a port into the vehicle CPU. The range of diagnostic functions offered are as follows:

1) Calibrate joint sensors
2) Calibrate handcontroller
3) Edit faults
4) Test Outputs (digital & binary)
5) Display mode
6) Download
7) Test Response

REMEX Modifications for Innovative End Effector (IEE)

To fit the IEE to the REMEX in either a front shovel or back hoe configuration with a Balderson thumb required an additional actuator, modified four bar linkages, joint sensor ,hydraulic valves and circuitry. To provide thumb control a special unloading circuit was conceived with a simple switch operation. This provided the operator with control of the thumb and the ability to grapple an object between the thumb and bucket and to retain this without the operator maintaining control of the thumb switch.

To operate the ITM lock and unlock functions another single control switch was provided at the operator's workstation together with LED's to indicate status. These arrangements were typical of an R&D configuration

but were nevertheless found to function quite satisfactorily to demonstrate the proof of concept and operator functionality. For fully remotized operations a basic video system was installed on the REMEX to provide overview of excavation , general positioning and interoperation with the SGTV and Gantry.

CONCLUSION

A SGTV control system and modified REMEX was developed and tested in an very aggressive schedule of four months. The system was fully functional and tested successfully at INEL. The precision specification in self-guided control was to track a path within +/- 1 foot and the test data showed that the tracking accuracy varied from an average of 0.58 for 0.972 ft. The Obstacle Avoidance System was able, except when facing a sharp corner, to reduce or halt the vehicle motion prior to reaching an obstacle. Both the Mission Planner and Vehicle Command and Control Generator exceeded the specifications. The coordinated operations of the SGTV, REMEX and Cooperative Telerobotic Retrieval Gantry Crane was demonstrated in performing retrieval and conveyance of simulated buried waste. The equipment used for this demonstration was design to investigate concepts for remote conveyance of buried waste. Prototype hardware were used to facilitate an inexpensive, short schedule and proof-of-concept demonstration. Based on the successful test results, it will not be difficult to upgrade the hardware to operate in a transuranic buried waste retrieval environment.

ACKNOWLEDGEMENTS

The authors would like to acknowledge the funding from the Office of Technology Development, Department of Energy and the support from Philip Rice of Lockheed Martin Idaho Technology and Tom Crocker, Ray Daigh and Randy Peterson of RAHCO International.

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RETRIEVAL SYSTEM FOR THE INEL PIT 9 REMEDIATION

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ABSTRACT

The world's largest remotely operated mobile work system is being built to excavate landfills. This 1,500 ton, self propelled machine is 260 feet long, 100 feet wide, and 60 feet high, and can span landfills up to 120 foot wide. It is designed to separate waste from the soil, package them separately for transport to processing, and backfill the pit while maintaining double containment and negative internal pressure. An inert atmosphere can be maintained inside the system if needed to prevent ignition of pyroforic materials.

The first application of this system is to be excavation of Pit 9, a 379 foot long by 120 foot wide waste burial site at the United States Department of Energy's Radioactive Waste Management Complex at the Idaho National Engineering Laboratory. Pit 9 was used as a waste disposal site between November 1967 and June 1969 and is being remediated in accordance with a Record of Decision signed by the United States Department of Energy, the United States Environmental Protection Agency, and state of Idaho Department of Health and Welfare.

The mobile work system is operated entirely by remote control. Seven remotely operated tool systems operate within the enclosure. Two use neutron radiation detection instruments to look into the dig face for radiation sources strong enough to require special handling. Two soil retrieval tools use brushes to dig out the soil and expose the waste. Two grapple tools gather and size reduce the waste and box it for transport to a processing area. The seventh tool system is designed to do routine maintenance on the other tools, resolve contingency situations, take samples, and excavate items which require special handling.

The telerobotic tools are monitored by software which prevents the tools from colliding and allows digging only in areas which have passed survey by the criticality control survey instruments. A thousand foot fiber optic umbilical connects the mobile work system to consoles located in a control room well away from the excavation area. Advanced telemetry systems relay commands from the consoles and provide the operators with video and feedback from an extensive array of instruments which monitor the status and activities of the equipment.

The remotely operated work systems separate the excavated soil and waste, package them separately in steel boxes, check the boxes to ensure that radiation levels are safe for downstream handing, and load the boxes on special trucks. Specially designed, remotely operated equipment keeps the outside of each box clean as the boxes are filled, closed, and air-locked into trucks for transport to a processing area.

HISTORY OF THE TECHNOLOGY

The technology upon which this system is based has evolved from mobile remotely operated equipment used for sophisticated work tasks in unstructured environments in the offshore oil fields.

In the early days of the offshore oil industry, designers placed as much equipment as possible atop steel platforms which rested on the sea floor and rose higher than storm waves. Divers did the construction and maintenance work on the platforms and pipelines, and on other equipment could not be placed above the surface. By the mid 1970's these platforms were being installed in water too deep for divers to reach the sea floor. This forced development of Remotely Operated Vehicles, or ROV's, capable of sophisticated work tasks in the unstructured environment of construction sites deep in the ocean.

By the mid 1980's, 100 horsepower ROV's with multiple onboard computers, highly dexterous manipulators, 1,000 pound "draw bar" pull at any vector, and an extensive suite of tool sleds and manipulator tools could perform almost any task that a 20 man saturation diving team could handle.

By the early 1990's these vehicle systems were routinely doing nondestructive testing of welds and were capable of repairing pipelines and replacing four inch valves in well head control units. The capability of these remote work systems made it practical to construct and maintain oil production equipment installed directly on the sea floor, thereby avoiding the need for giant steel platforms which can cost hundreds of

millions of dollars. In 1995 the reliability of these remotely operated work systems allowed extended sea floor work missions of a month or more without recovery to the surface for maintenance or repair.

The Kerr Hollow Quarry site on the Department of Energy's Oak Ridge reservation saw the first application of this remote work systems technology to remediation of a radioactive waste site. Remotely operated work systems were used exclusively to retrieve over 50 tons of mixed radioactive, toxic, and energetic waste and process it for long term storage as low level radioactive waste. The work was done entirely by remotely operated work vehicles, and no personnel entered the contaminated zone or the processing area at any time.

This remote systems technology is now being applied to excavation of Pit 9, a landfill site at the United States Department of Energy's Radioactive Waste Management Complex at the Idaho National Engineering Laboratory.

PIT 9 SITE

Pit 9 is a 379 foot long by 120 foot wide landfill used as a waste disposal site between November 1967 and June 1969. It is being remediated in accordance with a Record of Decision signed by the United States Department of Energy, the United States Environmental Protection Agency, and state of Idaho Department of Health and Welfare.

Approximately 110,000 cubic feet of waste generated at Department of Energy's Rocky Flats Plant and 40,000 cubic feet of waste generated at Department of Energy's Idaho National Engineering Laboratory was placed in Pit 9. A nominally 8 foot thick waste zone is situated on top of three and a half feet of soil and covered by six feet of overburden soil. The pit is to be excavated down to the underlying basaltic rock formation which is nominally 17 to 20 feet below grade.

EXCAVATION SYSTEM

An artist's concept of the remotely operated excavation system is shown in Fig. 1. This self propelled machine is 260 feet long, 100 feet wide, and 60 feet high, and weighs over 1,500 tons. It is designed to span and excavate landfills up to 120 foot wide. It moves along the length of the land fill, excavates the waste and the interstitial soil in separate waste streams, packages them for transport to processing, and backfills the pit. The system is designed to maintain double containment of all exposed contaminants under environmental extremes including 84 mile per hour winds, seismic events, and temperatures down to -40 degrees Fahrenheit. Negative internal pressure helps prevent the escape of airborne contaminants. Extensive measures are included to help prevent ignition of the waste mass in the event that pyroforic materials are exposed during excavation.

Fig. 1

All excavation tools are deployed by bridge cranes. Several design considerations led to this configuration. Since it is supported from overhead, the retrieval equipment can not collapse into any voids in the waste mass. The hoists on the bridge cranes can be readily controlled to ensure that the tools do not dig beyond the depth which has been surveyed and released for excavation. All power and control umbilicals run overhead and are therefore not exposed to damage or entanglement in the waste. Perhaps most important, the bridge crane deployment approach ensures that the retrieval tools can be removed from the excavation area, isolated, and washed down prior to any manual intervention for maintenance or repair.

REMOTELY OPERATED TOOLS

The mobile excavation system is operated entirely by remote control. Seven remotely operated tool systems work inside the enclosure. Two of the seven tools deploy passive neutron detection instruments to examine localized volumes of the dig face before it is disturbed. Special excavation and handling procedures are implemented if these instruments detect a radiation source which approaches the established limits for which criticality control measures are required.

Two remotely operated soil removal tools use rotary brushes to remove interstitial soil and expose the waste. Figure 2 shows one of the soil brush tools during factory acceptance tests. The X, Y, Z position of this 150 horsepower precision excavation machine is controlled by a bridge crane. This deployment method provides precise, reliable control of the location and depth of each cut. Six independently mounted rotating brush heads sweep an eight foot wide path, digging down only a few inches on each pass.

Fig. 2

The brush heads remove interstitial soil without disturbing large waste items such as drums or gas cylinders. Figure 3 shows drums which were buried in a test site and then exposed by the soil brush tool. The six brush segments move up and down independently to remove soil from between waste forms such as these drums. Video observation of the freshly exposed surface can then identify waste items which may require special handling. Shielding materials which may have blocked the view of the neutron sensing instruments can also be identified when exposed in this manner.

Fig. 3

The brush heads sweep soil and loose material onto elevating conveyors which dump onto a vibrating screen where fines are separated from debris. These two waste streams are accumulated in separate compartments on the tool. When approximately a ton of fines and debris have been accumulated the bridge crane positions the tool to dump the fines onto the cross pit conveyor and dump the debris into a waste transfer box.

The brush heads are made from materials similar to commercially available street sweeper brushes. Tests were conducted to determine the most effective range of bristle stiffness, size, spacing, and configuration for use with various soil conditions. Alternating wire and elastomeric bristles were found to be effective in both damp and dusty soils. This same bristle configuration has proven to effectively cut hard, dry, highly consolidated soil and exposes buried waste containers without significantly degrading them.

A cross pit conveyor transports soil from the soil brush tools to a hopper which meters out 64 cubic foot batches of the soil into soil transport boxes. The soil box lids are then closed by a remotely operated double lid transfer system.

After waste items have been exposed by the soil brushes, those items which do not require special handling are retrieved by one of the two grapple tools and placed in 4' by 4' by 8' steel waste transport boxes. Items too large to fit in a box can be cut down by a remotely operated shear. When a waste box is filled it moves to the end of the enclosure on a mine car trolley system and is removed from the contaminated area by a remotely operated double lid transfer system which keeps the outside of each box from becoming contaminated. Remotely operated sensors check radiation levels to ensure that each box meets the requirements for downstream handing.

Soil and waste boxes are individually air locked from the retrieval system into specially designed truck bodies so that double containment is maintained at all times. A fleet of four trucks transports the 10,000 pound boxes of soil and waste to the process building for assay and treatment.

The seventh tool system is designed to do routine maintenance on the other tools, resolve contingency situations, take samples, and excavate items which require special handling. This tool can also be used to wash down the enclosure after excavation is finished.

All seven telerobotic tools are monitored by computer software which prevents the tools from colliding and allows them to dig only in areas where the survey instruments have shown that there are no criticality concerns. A thousand foot long fiber optic umbilical connects the mobile work system to consoles safely located in a control room well away from the excavation area.

SAFETY AND ALARA CONSIDERATIONS

The system is designed to complete the excavation exclusively by remote operation, with no need for personnel to cross the pit boundary once excavation of waste commences. The tools are designed to deal with any waste items which may credibly be encountered. Electrical utility equipment, air filtration systems, and other service equipment are located in exterior rooms which are not exposed to the retrieval area.

One of the seven bridge crane tools, called the Robug, is designed to perform most of the routine maintenance and repair tasks required for the other bridge crane systems. This tool and two others are equipped with fire suppression systems as an additional safety feature.

Occasionally a bridge crane system will require more extensive maintenance than can be accomplished by the Robug. To accommodate this need, any of the bridge cranes can be moved to a maintenance area which is beyond the pit boundaries but still in containment. If a bridge crane should fail to move under its own power it can be moved by one of the other bridge cranes. The maintenance area can be isolated from the retrieval area and washed down before personnel enter.

This approach ensures that personnel are never exposed directly to the waste, and need never be exposed to a retrieval tool until after it has been washed to remove gross contamination. Radiation dose is thereby kept as low as is reasonably achievable, and the level of personnel exposure to toxic or energetic materials is likewise minimized.

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RAPID SITE CHARACTERIZATION AND MONITORING SYSTEMS USING SURFACE ACOUSTIC WAVE/ GAS CHROMATOGRAPHY

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ABSTRACT

This paper describes the development of a surface acoustic wave/ (SAW) technology for the rapid characterization of waste site radiation emissions. Developed under an EM-50 Department of Energy contract, the system was designed to rapidly assess contamination in field conditions.

The new technology allows capture, identification and analysis of Volatile Organic Compounds (VOCs), Non-VOCs, Dioxins, PCBs, and Energetics. This is done within water and soil matrices, within real time, on-site with parts per billion accuracy. The system has been modification capability to detect mixed waste containment, in radioactive environments. The ability to detect to picogram or ppb levels of compounds using fast chromatograph columns has been demonstrated for a wide range of materials. The application of SAW/GC technology will lead to the manufacturing of low cost, field portable products for waste site characterization and monitoring.

The design theory and development of the technology will be presented as well as the test results from the Savannah River Facility field demonstration. The SAW sensor has demonstrated high specificity and sensitivity for trace analysis screening at DOE remediation sites. Also, the results demonstrate significant cost savings when compared to reference laboratory analysis. The related deployment strategies for the various mixed waste sites and product maintainability issues will also be noted.

INTRODUCTION

This paper describes research on a fast GC vapor analysis system which uses a new type of Surface Acoustic Wave detector technology for characterizing organic contamination in soil and groundwater. The technology was originally developed through a Small Business Innovative Research (SBIR) Contract awarded by the United States Department of Transportation for the rapid field detection of contraband drugs of abuse. A proof of concept prototype was built and demonstrated for this project. Follow-on awards resulted in a portable fast gas chromatograph that detects contraband drugs of abuse. The system was again modified to detect compounds of pollution related to environmental site characterization and monitoring. The effort in developing this capability was sponsored by the United States Department of Energy (DOE).

PROJECT OBJECTIVES

The DOE research objectives were to demonstrate detectability and specificity of a Surface Acoustic Wave Gas Chromatograph (SAW/GC) for a representative number of Volatile Organic Compounds (VOC) materials. Once the VOC compounds were identified, data gathering and field demonstrations of the new analyzer were executed at a DOE site. Field testing of the SAW/GC was performed at the DOE Savannah River Facility, Augusta, Georgia. Matrix testing was performed with water, soil and gas samples. The performance of the SAW/GC analyzer was validated by comparing results taken with an on-site HP chromatograph. Through these tests, the system, based upon surface acoustic wave/ gas chromatography, demonstrated the ability to identify and quantify the presence of VOCs.

DESCRIPTION OF THE TECHNOLOGY

The SAW/GC instrument is comprised of a head assembly that contains the capillary column and the SAW detector and the support chassis, containing the helium carrier gas, laptop computer and the related micro-processors. There are two modes of operation: Sample and Inject. The system utilizes a two position, 6 port GC valve to switch between sampling and injection modes. In the sample position, environmental air containing the suspect compounds, is passed through an inlet preconcentrator or water trap and then through a sample loop trap. The function of the loop trap is to concentrate VOC materials when in the sample position. During the sample mode, helium carrier gas flows down a capillary column and impinges onto

the surface of a temperature controlled SAW resonator crystal shown in Fig. 1.

Fig. 1

Switching the valve to the inject position causes helium carrier gas to flow backward through the loop trap and onto the column. After the valve is switched into the inject position the loop trap is rapidly heated to 200C causing the trapped VOC materials to be released into the GC column. The temperature of the GC column is linearly raised to approximately 125C over a 5-10 second time and this causes the VOC materials to travel down the column and exit at a time characteristic of the VOC material.

The SAW resonator is a unique type of GC detector that, incorporated with the theory of gas chromatography, allows for the analysis of suspect compounds to new ranges of specificity and concentration. VOC materials as they exit the GC column are trapped on the surface of the resonator and cause a change in the characteristic frequency of the crystal. The adsorption efficiency of each VOC material is a function of the crystal temperature and by operating the crystal at different temperatures the crystal can be made specific to materials based upon that materials vapor pressure. Also, since the crystal acts as a micro-balance it integrates the total amount of material present. To obtain a conventional chromatogram plot of retention time, the derivative of frequency versus time is calculated. This is in contrast to a conventional GC detector which detects the flux and where peak integral calculations are required to obtain the amount of each material present.

RESULTS AND ACCOMPLISHMENTS

To demonstrate the technology a portable laboratory scale instrument was constructed and tested with the representative VOC materials listed in Table I. Each material was tested with a calibrated vapor source either purchased as calibrated bottled gas or created by injection into a known volume (tedlar bag). Calibration results based upon a 10 second sample are listed in Table II. In general the sensitivity of the instrument for all materials was 1 ppm or better. For materials with lower vapor pressure, such as Toluene and tetrachloroethylene, sensitivity extends well into the ppb range. To achieve ppt sensitivity it is only necessary to extend sample time. However, the advantage of a short sample times is near real time operation.

Table I

Table II

FIELD TEST RESULTS

The prototype instrument was shipped to the DOE Savannah River Facility, Aiken, SC, where it was used to obtain real time measurements of well head gases. Place in a small van, the system was transported to numerous well heads for sample analysis. The instrument was designed for battery power, however, it was discovered that it could operate for long periods of time by drawing power from the transport vehicle. Power was supplied utilizing a 110 volt DC-AC invertor connected to the automobile battery. Samples were taken under 3 modes: 1) the instrument was stationary and engineers brought tedlar bags to a central point for well-head analysis; 2) the system was mounted in the van and driven to various well head locations for sample analyte analysis using tedlar bags; 3) the system was driven next to a well head and with the detectable head unit, connected to a sampling port for direct injection of sample analytes. To verify the accuracy of the instrument, calibrated tedlar bag samples were used to calibrate the SAW/GC. A typical output screen for one such

bag containing approximately 100 ppm TCE and PCE is shown in Fig. 2. The user interface shows two chromatograms, one is the derivative of SAW frequency and the other is SAW frequency vs. time. The duration of the chromatogram is 10 seconds and retention times for TCE and PCE is 3.54 and 5.54 respectively. The operator can display quantitative information as ppm/ppb, in mass units of picograms or nanograms, or alternately in SAW units of frequency.

Fig. 2

Many different measurements were taken and compared with an on-site HP GC as shown in Fig. 3. The results of this relative comparison indicate that the SAW/GC and the HP GC agree within approximately 20%. Much of the variation is attributed to variations in sampling and preconcentration within each instrument.

Fig. 3

APPLICATION , BENEFITS AND ISSUES

Those involved in environmental characterization and monitoring are demanding innovative technologies that are significantly lower in cost while providing rapid methodologies for the collection and analysis of soil, water, and air samples in the field. The SAW/GC instrument, if properly integrated into the sampling/analytical plan, represents a significant savings in cost and time. The instrument can be effective if the 2 major criteria are met affirmatively: 1) Is there a need for measurement or monitoring decision making data "in situ"? and 2) Is there a critical need to make decisions in the field in real time, e.g. 8 hours or less.

On-site Monitoring (OM) is useful for the following situations:

- Define emergency response actions
- Assess impacts to potable water
- Monitor purge water
- Screen ground water during exploratory drilling
- Screen monitor wells
- Define the extent of soil contamination
- Determine migration pathways
- Estimate the amount of contaminated soils
- Obtain ground water samples without monitoring wells
- Determine sources of contamination

Rapid on-site screening and analysis has been identified as a potential solution to four major issues:

Operator Safety- Is the exposure level too high for site personnel? A significant problem exists regarding the safety of personnel involved in characterization and remediation. Several solutions exist using amino-assay kits, however, specificity and real time analysis is limited. The SAW/GC analyzer represents a solution for identifying suspect compound specificity and concentration in near real time (10 seconds).

Real-Time Analysis- How can we identify contaminated areas by reducing the total number of samples taken for analysis? Past site characterization techniques follow a pattern of sending samples at regular intervals to a reference laboratory for validation of suspected areas of contamination in both specificity and concentration.. Fixed laboratory costs range from $600 to 700 per sample and average 6 days from sample submittal to reporting of anlyte results. Mobile field laboratories range from $250 to $300 per sample and can process between 30 to 40 samples a day. A portable SAW/GC instrument can be utilized effectively by defining the three to four (instead of 40) compounds in

the suspect anlyte you are searching for at the site. This planning will allow for site screening for mixed waste by identifying commonly related compounds.

Screening- Where will we most likely find contamination? Reducing the turn-around time associated with analyzing compounds will save money. Field personnel can make decisions in hours instead of waiting days for the return and analysis of data. The use of a portable SAW/GC allows an operator to rapidly screen for suspect compounds and in turn make cost effective decisions.. Compounds found in rapid screening are confirmed by a reference lab for compliance risk assessment. This reduces the cost of site assessment and remediaton.

Primary Remediation Site - Can we characterize the primary location to remediate by identifying the "hot spots" of contamination? Historically, field portable instruments were limited because they lacked specificity and sensitivity. The advantages of the SAW/GC are portability, accuracy, and speed. The new SAW sensor demonstrated sufficient specificity and sensitivity to be used as a fast trace analyzer or screening tool at DOE remediation sites. Using the SAW/GC analyzer as a field screening tool, cost savings over current techniques, which require expensive laboratory testing, are estimated to be more than $50,000 per month. The cost of the SAW/GC screening instrument will be recovered within less than two months of operation.

SAW/GC technology provides both advantages and disadvantages to current analytical methods. A brief comparison is as follows:

Advantages

- Easy to learn the Instrument
- Easy to Use
- Inexpensive, and portable
- Capable of correlation with laboratory analysis
- Verifiable and reproducible
- Capable of calibration to contamination matrix

Disadvantages

- On Site Personnel must be mechanically competent
- Field personnel must have the authority to make decisions
- Cost effective if a large number of samples are collected
- Contaminant must be known prior to site activities
- Heterogeneous conditions may lead to inconsistent data sets
- Field screening methods may not be applicable to all site conditions

FUTURE PLANS

There are many related applications for SAW/GC technology. While at Savannah River the instrument was also used to measure catalytic converter performance, DNAPL probe experiments, and to characterize VOC break through in carbon scrubbers. Future plans are to use the instrument in field conditions to measure performance against existing methods. Based upon the current results the goals are to begin development of SAW/GC screening instruments for use at DOE remediation sites. The commercialization effort is being carried out by Electronic Sensor Technology, Inc., a limited partnership company managed by Amerasia Technology and tasked with the development of SAW/GC instruments. The commercialization effort is being aided by a partnership between Amerasia Technology, Inc., and the U. S. Department Of Energy Morgantown Energy Technology Center. This new program will involve continued Field testing at DOE sites, EPA certification and verification, and the development of new SAW/GC instruments to detect and quantify Dioxins,

Furans, and PCBs at DOE sites. The new units are smaller in size and contain improvements noted from the Savannah River Field work. These units will be available for field demonstrations and data gathering in the First Quarter of 1996. Designated the Model 4100, it has been designed for environmental needs. A photograph of the Model 4100 is shown in Fig. 4.

Fig. 4

ACKNOWLEDGMENTS

We wish to thank the following participants in the field methods and data gathering sessions necessary in generating the results and conclusions of this paper.

Eddie Christy, Contracting Officer, METC
U. S. Dept. of Energy, EM-50
Dr. Joe Rosabbi
Westinghouse Savannah River Site
John Decker, Manager
Lockheed-Martin Corporation

This effort was supported by the Department of energy's Morgantown Energy Technology Center, under contract DE-AR21-94MC31177 Covered from June 1994 to March 1995.

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SUCCESS IN HORIZONTAL BARRIER DEVELOPMENTS

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ABSTRACT

A successful proof of concept demonstration has been conducted of operational methods and tooling for the in situ construction of underground horizontal barriers for the control and containment of groundwater and contamination. The method involves jet grouting with specially adapted tools guided between twin, parallel directional wells for the placement of a grout "slurry floor" beneath a waste site. This paper describes progress by principal subcontractor Brown and Root Environmental on the Fernald Environmental Restoration Management Corporation (FERMCO) In Situ Land Containment Project sponsored by the U.S. Department of Energy (DOE) Office of Science and Technology (OST) through DOE's Fernald Environmental Management Project (FEMP). Two types of special horizontal grouting tools that operate on different principles have been designed and tested through significant field trials. Experiences with the tools and results of field tests are presented.

INTRODUCTION

Brown and Root Environmental has undertaken a project, under contract to FERMCO and DOE, to develop advanced technology for the construction of underground, horizontal, "slurry floor" barriers for application to

environmental restoration of contaminated soil. Horizontal barrier construction capability is desired within the remediation industry and by DOE to enable containment of environmental contamination where inadequate natural barriers exist to mitigate the spread of the contaminants.

The objective of the work is to develop reliable methods of constructing extensive, competent horizontal barriers underneath waste sites without excavating or penetrating the waste during the process. The principal technical objectives of the work include in situ constructability, the ability to form a barrier of sufficient width and length to be of practical value, and the ability to operate in the various types of soil encountered on a significant number of waste sites.

This project stems from an approach developed by Halliburton Services, a sister company of Brown and Root Environmental which provides commercial environmental services. The approach involves an adaptation of a construction method known as jet grouting. The new method is a process for depositing grout(or any pumpable fluid) underground in the horizontal plane using special jetting tools. In late 1992 a proof of concept demonstration was conducted which resulted in the successful construction of a prototype horizontal barrier of limited width. Although successful in placing a horizontal barrier, the demonstration revealed several technical hurdles relating to grouting tool design and operation that need to be overcome to enable practical (economical) application of the method. The following sections describe the progress toward overcoming the hurdles.

HOW THE PROCESS WORKS

To apply the process, parallel, directionally drilled pipes (guide tubes) are installed under the waste site, as indicated in Fig. 1. A special horizontal jet grouting tool is attached one end of two of adjacent pipes so that the tool spans between the pair. The other end of the pair of pipes is attached to a controlled pulling device. High pressure grout is supplied through one of the pipes to the special tool that has jet orifices through which the grout exits with high kinetic energy. The grout jets fluidize the soil into a slurry, and the tool is then advanced along the trajectory between the pair of pipes by withdrawing the pipes using the pulling device. As the tool moves through the soil it continues to fluidize soil in front of it and leaves behind a panel of amended soil. The panel or section of barrier formed by a single pass of the jetting tool is the unit "building block" for construction of a large horizontal barrier formed by joining several panels side-by-side. Figure 2 shows a single panel constructed during the proof of concept field demonstration.

Fig. 1

Fig. 2

The spacing of the drill pipes, and the degree to which they must be parallel depends on the design of the grouting tool. For example, rigid tools (described below) require placement of the drill pipes within close tolerances, say + six inches of design centerline, while the flexible tools have a high tolerance (+ several feet) for being non parallel.

Forming competent (waterproof) joints between adjacent panels is essential for successful construction of large barriers. The technique devised by Brown and Root for assuring that the next panel formed mates up with the previous panel is to have the jetting tool draw in the next drill pipe along one edge of the panel as it is being formed. This drawn-in pipe is used as one of the pair needed to place the next adjoining

panel of the barrier. The grout planned for use in construction of horizontal barriers is cement-bentonite mixtures of the type commonly used in the construction of vertical slurry walls, although any pumpable grout can be used in the process. Note that the grout must be able to withstand high pressure, shear and impact.

EVOLUTION OF HORIZONTAL JET GROUTING TOOLS

Four different concepts have been investigated for the design of the special tool required for in situ construction of horizontal barriers. The approach planned for use of each design was the same; that is, the tool spans a pair of drill pipes and is advanced by pulling the pipes. The four initial concepts were as follows:

1. A rigid member (rigid bar tool) with a line of jets facing the direction of advancement of the tool. This tool might resemble a comb, with the teeth of the comb representing the jets. This design was intended to achieve a simple, no moving parts approach but with jets all along the advancing face of the tool.
2. A rigid tool (Rotary Tool) that rotates and operates in a manner similar to a rototiller with blades, as well as jets that cut through the soil. This design was intended to provide mechanical assist to the jets, form a thicker panel, and allow the rotating jets to compensate for others that may become plugged.
3. A shuttle tool with a rigid frame and a mechanism with forward facing jets on the leading edge of the tool that shuttles back and forth across the face of the tool. This design was intended to minimize the number of jets required for formation of a single horizontal panel, since the higher the number of jets, the more energy is required to conduct the operation.
4. A flexible tool (FlexTool) that operates in a manner resembling that of a cable saw. This design was intended to minimize the need for precisely parallel drill pipes, as well as to minimize the number of jets.

The four candidate tool designs were subjected to a series of field trials with the objective of selecting the tool(s) with the highest prospect for successful application. Previous reports (1,2) detail testing and results that eliminated the rigid bar tool (made barriers of variable and insufficient thickness, and failed due to jet plugging), and the shuttle tool which failed from mechanical complications related to subsurface reciprocating parts. The tools selected for further adaptation were the RotaryTool and the FlexTool, pictured respectively in Fig. 3 and Fig. 4 in their latest stage of evolution.

Fig. 3

Fig. 4

The Rotary Tool is powered from the surface by a shaft passing down through one of the drill pipes to transmit torque to a universal gear which rotates the tool. The Rotary Tool is constructed of a 10-ft (width of horizontal tool) section of 6-in pipe with blades affixed to mechanically cut a minimum barrier thickness of 12 inches. Each end is equipped with outboard cutting blades and jets intended to cut into neighboring barrier panels to facilitate seaming.

The FlexTool shown in Fig. 4 was constructed of 100 feet of sealed bearing chain resembling the track on heavy earthmoving equipment. This equipment was found to have sufficient strength to withstand the tensile forces placed on it by the pullers, yet have sufficient flexibility to operate during barrier formation between drill pipes at reasonable and

approximately parallel spacing. The "treads" of the FlexTool consist of 6" x 12" bearing plates intended to slide horizontally along the subsurface cutting face. Jet substructures (subs) are located at three positions along the tool: one at the center, and at each end of the tool. The center sub has an 8" x 18" port through it to allow passage of small rocks. There are twelve jets in each of the subs.

This report focuses on the advances and results of field tests with the FlexTool, as testing of the Rotary Tool is less advanced, and is intended to be the subject of future detailed reports.

FLEXTOOL TEST RESULTS

Field trials of the FlexTool were conducted in October 1995 at a testing facility in Duncan, Oklahoma operated by Halliburton Services, a provider of high pressure pumping services. The soil at the site is dense, red Oklahoma clay with occasional strata of hardpan, and is considered to provide stringent test conditions since the clay is very dense, with standard penetrometer test results exceeding 100 blow counts in some locations of the site. The purpose of the testing was to check out operating characteristics and mechanical features of the tool, and to practice the type of operations to be performed with the tool and support equipment during construction of horizontal barriers.

Two parallel, 2 3/8-in drill pipes for advancing the FlexTool were installed on 30-ft centers 6 feet below grade. The pipes were installed by cut and cover: trenching, laying the pipe in the trench, and then backfilling and compacting. The pipes formed a gentle arc to the surface at each end, and the horizontal portion of each pipe was 100 feet in length. The cut and cover pipe placement method was used, instead of installation by drilling, simply as an economy measure to avoid the cost of directional drilling. The cut and cover method is considered to result in a more severe test condition, since friction on the pipes and resistance to advancement is expected to be greater in the compacted backfill than for drilled-in pipes installed using lubricating drilling fluids.

The jetting fluid used in the field trial was water at a pressure of 4,000 psi. With the FlexTool at the surface, jetting was started and the tool pulled into the soil and advanced along the path between the drill pipes. The drill pipes were pulled with a D-6 class bulldozer attached to each pipe. The method of operation was to advance one bulldozer approximately 20 feet, stop it, and then advance the other one a similar distance. This operation was conducted for 60 minutes, advancing the FlexTool a distance of 117 feet and forming 3,510 square feet of subsurface "barrier". The operation progressed smoothly with no interruptions, despite encountering a buried trash disposal pocket containing debris including plastic construction barrier fence and hose. Some fluid breakthrough to the soil surface occurred at the location above the debris, otherwise the surface between the buried drill pipes subsided uniformly by about 1 foot as the tool passed beneath. The subsidence resulted from the use of water as jetting fluid. The water cut through the soil in front of the tool, forming a fluid soil/water slurry which washed out behind the tool through the pathway the tool had traversed. When grout is used as the jetting fluid, the increased density and viscosity of the resulting slurry is expected to minimize washout and reduce subsidence.

CONCLUSIONS

The work produced several advances in the technology of barrier construction and provided the following conclusions relating to equipment and method of operation.

A viable method exists for in situ construction of horizontal, sub surface soil barriers using the FlexTool. Barriers of at least 30 feet width are constructable with present technology.

The FlexTool is operable for barrier formation in clay and other cohesive soil without cobble and large rock.

State-of-the-art precision in placement of directional drilled pipe is adequate for construction of horizontal barriers using the FlexTool.

Horizontal barrier can be formed at a rate of 3,500 sq.ft. per operating hour at an approximate cost of $50 per square foot.

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30-17

MODELING OF SUBSURFACE BARRIER PERFORMANCE

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ABSTRACT

Perfluorocarbon tracer technology is being tested at the Hanford Geotechnical test facility as a means of verifying subsurface barrier integrity. To support the experimental design of this project modeling of the transport of the tracer from its injection point to monitoring wells placed outside the barrier has been conducted. The conceptual model considered diffusion and advection due to barometric pumping as the transport processes. Simulations were performed for intact barriers as well as barriers with small (1 - 10 cm) imperfections in the barrier. The role of advection in determining the distribution of a contaminant in the subsurface was also studied.

INTRODUCTION

Subsurface barriers are a promising remediation option. Some uses of subsurface barriers include surrounding and/or containing buried waste, as secondary confinement of underground storage tanks, to direct or contain subsurface contaminant plumes and to restrict remediation methods, such as vacuum extraction, to a limited area. The performance of these barriers will be most effective if the barrier is continuous around the wastes. If a breach forms, that is, the barrier is noncontinuous around the wastes, performance will be degraded. It is recognized that current methods to verify the barriers integrity do not exist (1). Perfluorocarbon tracer (PFT) technology is being tested at the Hanford Geotechnical test facility as a means for verifying subsurface barrier integrity.

The physical system under study is displayed in Fig. 1. The approximate dimensions of the system are 25 feet deep and 40 feet in diameter. The barrier walls slope at an angle of 45 degrees. Eight monitoring wells are

uniformly spaced parallel to the barrier wall at a distance of approximately one meter from the wall. The final subsurface barrier will consist of a close-coupled system of a polymer inner layer and a soil/neat cement outer layer. To better understand the performance of the entire system, testing will be conducted in two phases. In the first phase, only the soil/neat cement barrier will be in place. PFT's will be injected into the soil inside the barrier. Detection wells placed approximately one meter outside the barrier will be used to follow the time evolution PFT concentration. In the second phase, the polymer grout layer will be emplaced and the testing repeated.

Fig. 1

To support the PFT testing procedure, modeling of the subsurface movement of the PFT gas has been conducted for the soil/neat cement barrier. The modeling will be used to estimate the rate at which the tracer will travel through the barrier and reach the monitoring wells, the effects of small holes (1 - 10 cm) on concentration at the well, and the effects of the waste tank on movement of the gas. This information will be used to assist in the determination of injection rates for PFT's into the subsurface system, the frequency and duration of sampling, and the time to flush the system prior to testing the close-coupled system.

After the tests are completed, the data will be analyzed and the modeling will be repeated. In this stage, modeling will be used to attempt to determine the size and location of breaches (if any), and estimate the range of possible values for transport parameters (i.e., diffusion coefficients of the PFT's in the barrier and soil).

This paper will discuss the modeling work performed to support the experimental design of the subsurface barrier verification testing program. Currently, the data collected from the in-situ experiment is being analyzed and modeling to match the data will be attempted and reported later.

CONCEPTUAL MODEL

The problem involves transport of an injected tracer through an engineered barrier (soil/neat cement) to a monitoring well. To model this requires knowledge of the rate of tracer injection, location of injection, geometry of the system, location of the monitoring well, and transport properties of the PFT through the soil and soil/neat cement barrier.

The inverted pyramidal structure is displayed in Fig. 1. The subsurface barrier consists of a cone of injected soil/neat cement with a subtended angle of 90 degrees, approximately 1250 cm in diameter and 630 cm in height. The upper, open end, of the cone is covered with approximately 60 cm of soil. The design basis for the effective thickness of the barrier is 100 cm (this is the thickness of the barrier in the plane that is parallel to the barrier). Monitoring wells are placed at a distance of approximately 100 cm from the outside edge of the barrier and parallel to the barrier. To simulate this geometry, a two-dimensional slice through the center of the cone was taken. Cylindrical geometry was used to represent the subsurface system. The tank in Fig. 1 was represented as a no flow boundary. The original plans were to inject the PFT tracers inside the pyramid at the centerline near the bottom. Therefore, symmetry was assumed along the centerline of the inverted pyramid.

To determine the effects of a small breach in the barrier, simulations with a completely intact barrier will be performed as a baseline. Then the effects of having small imperfections (caused by imperfect grouting)

will be studied. In this analysis, the imperfections are represented as a hole through the entire wall. The range of hole sizes that were modeled was between one and ten cm.

The dominant transport process for air in soil systems is believed to be diffusion (2). Advection resulting from barometric pressure changes can facilitate the release of the tracers to the atmosphere and is also considered. Site specific data on appropriate transport parameters are not available. Therefore, literature values have been used. For simplicity and because of the lack of site-specific data, it is assumed that there are two distinct materials in the subsurface system, the soil and the soil/neat cement barrier.

For the base case, the diffusion coefficient of the PFT in the soil has been selected as 10-2 cm2/s. This value is similar to that for radon gas in dry soils (3). The diffusion coefficient through the soil/neat cement was selected as 10-4 cm2/s. This is on the low end of the range of diffusion coefficients for radon gas through residential concretes (4). The low value was selected in an attempt to provide a lower estimate of release to the monitoring wells and to insure that sampling would be able to detect the PFT's.

Advection in the subsurface system as a result of barometric pressure changes was modeled assuming a sinusoidal change in velocity within the soil which corresponded to changes in the barometric pressure. The amplitude of the velocity was 3.4 10-4 cm/s. This corresponds to the velocity expected for a pressure variation of 1/30 of an atmosphere and permeability consistent with the range of values found for sandy soils (5). The period was five days, 2.5 days of which the system exhaled air from the contaminated region and 2.5 days of which essentially tracer free air from the atmosphere entered the subsurface. It was assumed that the response to the pressure variations was instantaneous within the entire modeled domain. Ancillary calculations were performed to support this as a reasonable first approximation for the parameter values selected in this simulation. The permeability of the soil/neat cement is expected to be a few orders of magnitude lower than in the soil. Therefore, it was assumed that the advection velocity in the barrier was zero.

COMPUTATIONAL MODEL

The subsurface barrier system is modeled in cylindrical geometry using a two-dimensional finite-element transport code, BLT (6). This problem has two size scales. The first scale is that of the system itself. The height from the bottom of the subsurface barrier to the ground surface is 6.6 m. The radius of the barrier was also approximately 6.6 m and a total distance of 12.8 m was simulated in the direction parallel to the ground surface. The second scale is that of the size of the potential breach which is on the order of a few centimeters. It would require nearly one million computational points to model the entire system on the scale of one centimeter. This was not computationally feasible. To account for the two scale sizes, variable mesh spacing was used. A fine mesh (order of one centimeter) was used in the region of the hypothetical breach. The mesh was increased in size as the distance away from the hypothetical breach increased. The slanting soil/neat cement barrier was modeled through definition of the finite elements used to represent the barrier to also slant at a 45-degree angle. These two details caused a complicated mesh with three thousand computational points to be used. The finite element grid is displayed in Fig. 2.

Fig. 2
Initially, the system is tracer free and, therefore, the initial condition is zero concentration at all locations. The boundary condition assumed zero flux at the centerline due to the assumed symmetry. Zero concentration boundary conditions were used at the top boundary, bottom boundary, and right-hand boundary defined in Fig. 2. The right-hand boundary is located at a large enough distance such that essentially no tracer reaches the boundary of the simulation period of 0.1 years. The top boundary was selected to have zero concentration to represent PFT concentrations in the atmosphere which are assumed to be zero.
The source was treated as a point source being injected at the centerline at an elevation of 97 cm above the bottom of the facility. This is 40 cm above the top of the soil/neat cement layer in Fig. 2. Two injection scenarios were modeled: a 3.7 day pulse injection and continuous injection over the entire simulation period of 37 days. The air injection rate was assumed to be 30 cm3/min at a unit tracer concentration. This problem exhibits a linear response to the injection concentration. This information was used to normalize all of the simulation results to the injection concentration.
PFT tracers are non-reactive in soil systems and can be detected at levels of one part in 1015. Typically, injection concentrations are on the order of one part per million. Therefore, the detection limit will be approximately 10-9 of the incoming concentration. One objective of the modeling work was to define the time at which the PFT's will first be detected at the monitoring wells and the time evolution of concentration at the monitoring wells. For the purposes of defining the experimental protocol, the minimum detection limit was multiplied by a factor of 100 for the design objective. This provides a design goal for the normalized concentration in the monitoring wells of 10-7
The advective-dispersive transport equation for a non-zero velocity in the z-direction only is used to solve for the movement of the tracer from the injection location throughout the modeled domain. The equation is:
Eq. (1)
where:

C(x,z,t) is the tracer concentration
D is the diffusion/dispersion coefficient (cm2/s)
V(t) is the volumetric flow rate per unit area of the gas in the subsurface system,
Q is the volumetric air content of the porous medium, and
Q(x, z, t) is the rate of contaminant injection into the system.

The diffusion/dispersion coefficient consists of two terms, one term represents molecular diffusion while the other represents dispersion due to advective flow in a porous medium. Mathematically it is expressed as:
Eq. (2)
where:

Dm is the molecular diffusion coefficient, (cm2/s),
Dt is the transverse dispersion coefficient (cm),
Dl is the longitudinal dispersion coefficient (cm),
Vx is the x component of the velocity, (zero in this study),
Vz is the z component of the velocity,
|V| is the magnitude of the velocity, and
dij is the Kronecker delta = 1 if I=j and 0 otherwise.

For the problem under study, molecular diffusion is expected to dominate over the dispersion terms.

In the above expression, the air phase Darcy velocity is expressed as a function of time. It is assumed that the Darcy velocity responds instantly to barometric pressure changes at all locations in the modeled domain. As stated before, the velocity through the wall is assumed to be zero. Elsewhere, it is assumed to flow only in the direction perpendicular to the ground surface. With these assumptions, the flow occurs only in the z-direction in the soil and is:

Eq. (3)

where:

V0 is the amplitude of the velocity,
w is the frequency of the pressure oscillation, and
t is time.

MODEL RESULTS

The computer code BLT (6) was modified to simulate the time-dependent velocity as expressed in the preceding equation. The modified BLT code was then used to solve the above equations for the contaminant plume due to injection of the tracer. A wide range of cases was considered to assist in gaining an understanding of the system behavior. The objective of these simulations was to estimate the time evolution of tracer concentration at the monitoring well.

The time evolution of the contaminant plume was followed for 0.1 years (36.5 days) using the base case parameters in Table I. In the base case it is assumed that the subsurface barrier wall is intact and no substantial breach occurs. The results of this simulation 14.6 days after the start of the experiment are presented in Fig. 3. The contour plot color key is presented in Fig. 4. All projected concentrations are normalized to an injection concentration of unity.

Table I

Fig. 3

Fig. 4

In Fig. 3, it is seen that for the base case parameters, the soil/neat cement wall provides an effective barrier to migration of the PFT's. Concentrations at the well 14.6 days since the beginning of tracer injection are more than eight orders of magnitude (which is the design objective of the experiments and the lowest value represented on the contour plots) less than the injection concentration. Inspection of the output files indicates that the projected baseline concentrations are nine orders of magnitude less than the injection concentration at this time.

To determine the effect of the barrier diffusion coefficient on release, the base case was modified by increasing the barrier diffusion coefficient by a factor of 10 to 10-3 cm2/s. This value is toward the high end of measured radon diffusion coefficients through residential concretes (4). In this case, predicted concentrations at the monitoring well at 14.6 days reached a maximum normalized concentration of 7 10-6 and averaged more than 10-6. This exceeds the design basis concentration by an order of magnitude.

To determine the effect of a small breach in the barrier a 5-cm hole was simulated as having the same properties as the soil, diffusion coefficient of 10-2 cm2/s. This hypothetical hole in the barrier was located at an elevation of 1.8 m from the bottom of the modeled domain, 0.8 m higher than the source. The total distance from the source to the edge of the hypothetical hole in the barrier is 1.8 m. The results of this analysis at 14.6 days after the start of the injection, Fig. 5,

indicate that the breach has a pronounced effect on the contaminant plume. Streaming through this 5-cm breach is clearly evident. The peak normalized concentration at the well is 4 10-6. Average concentrations along the lower section of the well are above 10 -6 a three order of magnitude increase over the projected concentration for the intact wall. In fact, the projected concentrations at the monitoring well for the 5-cm hole simulation was of the same order of magnitude as the case with the barrier diffusion coefficient increased an order of magnitude over the base case value.

Fig. 5

In all three cases, the contaminant plume within the region bounded by the subsurface barrier is almost identical. Average concentrations in this region are approximately 10-3, four orders of magnitude larger than at the monitoring well location for the case with a barrier breach. This indicates that only a small fraction of the tracer reaches the monitoring wells under the conditions simulated.

The hole size was varied from 1 - 10 cm and the results were similar. Even a 1 cm. hole would permit the concentration of PFT tracer that reaches monitoring well to exceed the base case (an intact barrier) value by 2 - 3 orders of magnitude.

The role of advection was studied by comparing the case with the 5-cm hole in the barrier to an identical simulation except that the amplitude of the advection velocity was set to zero. In this case, diffusion is the only transport mechanism. Predicted concentrations at the monitoring well increased by as much as an order of magnitude in the absence of advection. Advection tended to smear the plume and shift the plume slightly toward the surface on average. Therefore, it did increase the release of the tracer through the ground surface to the atmosphere. Following the time evolution of the plume, it was observed that the plume did shift up and down with the changes of advection velocity. Although, the velocity variations were cyclic, when the flow was directed out of the subsurface, it moved tracers from a region of higher concentration upwards to regions of lower concentrations. When the flow was into the system, it moved tracers in regions of lower concentrations deeper into the subsurface toward regions of higher concentrations. The net effect was a slight shift in the plume toward the surface as compared to the simulations that did not model advection. The net upward shift is due to the fact that the source is injected at the bottom of the domain. It may not be true for a source located in the middle of the modeled domain.

COMPARISON OF EXPERIMENTAL RESULTS WITH MODEL PREDICTIONS

PFT tracers were continually injected for three days into the area contained by the soil/neat cement barrier, Fig. 1, just beneath the empty tank at the center of the region bounded by the cone. Seven monitoring wells were located parallel and approximately 1 meter outside of the barrier. The wells were approximately 1 meter from the barrier. Measurements for PFT tracers were taken from each well for 18 days after the start of injection. PFT concentrations within the region bounded by the barrier were measured for the three-day injection period

The data showed a net drift toward one side of the barrier. The interior monitoring well designated as N, for North, had measured concentrations approximately one order of magnitude greater than the interior well designated S, for south. This indicates that advection is occurring. The cause of this net drift is not known, however, it has been postulated

that it is due to the injection velocity (12 - 15 cm3/min). This and other possible explanations are under investigation.

The drop in concentration between the inner and outer monitoring wells (i.e., across the barrier) was approximately four orders of magnitude. There was no evidence of a substantial breach in any region as the drop in concentration across the barrier was consistent at all monitoring wells.

To estimate diffusion coefficients in the soil and the barrier, prospective model evaluations were performed. The computational model is similar to the one described to examine the influence of a breach in the barrier, Fig. 2, with the exception that the dimensions were changed to match the as-built dimensions exactly and the source location was changed to reflect the experimental conditions.

The base case diffusion coefficient values, Dsoil = 10-2 cm2/s and Dwall = 10-4 cm2/s, provided concentration estimates that were far lower than the measured value. This was expected because the base case values were chosen with the intent of under predicting the amount that would reach the wells to insure that detection would be possible.

A range of different values of the diffusion coefficients was simulated. The results have been compared to the measured average value of the seven monitoring wells and are displayed in Fig. 6. From these evaluations, the soil diffusion coefficient for the PFT has been determined to lie between 1 - 5 10-2 cm2/s under the test conditions. The diffusion coefficient for the soil/neat cement barrier has been determined to lie between 1 - 5 10-3 cm2/s. The best fit was obtained using a soil diffusion coefficient of 2 10-2 cm2/s and a barrier diffusion coefficient of 2 10-3 cm2/s. Further evaluations to improve the best fit will be taken.

Fig. 6

In Fig. 6 the effects of barometric pumping can be seen from the two simulations with Dsoil = 10-2; Dwall = 10-3 cm2/s. In one case, barometric pumping was not simulated. In the other case, barometric pumping was simulated and the fluctuation in concentration at the well is clearly in evidence. The fluctuations had only a minor impact on the concentration and were not in evidence in the experimental data. Therefore, barometric pumping was not simulated in other evaluations.

CONCLUSIONS

Modeling of transport of PFT tracers in a subsurface system consisting of soil and a soil/neat cement barrier has been conducted. The results support the feasibility of detecting tracers outside of the barrier on the time frame of a few weeks. For the base case, a two order of magnitude difference in the PFT diffusion coefficient in the soil and barrier, small holes (on the order of cm) should be easily detectable. As the difference in diffusion coefficients of the soil and barrier decreases, the ability to detect small holes also decreases.

Site-specific data on transport parameters were not available. Therefore, the model evaluations were compared to the experimental data and used to estimate the diffusion coefficient for the PFT through the soil and barrier. The best fit to the data indicates that the soil diffusion coefficient is approximately 2 10-2 cm2/s and the barrier diffusion coefficient of 2 10-3 cm2/s. It was determined that barometric pumping did not contribute significantly to transport in this experiment.

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30-20

MERCURY RETORTING OF CALCINE WASTE, CONTAMINATED SOILS, AND RAILROAD BALLAST AT THE IDAHO NATIONAL ENGINEERING LABORATORY

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ABSTRACT

The Idaho National Engineering Laboratory (INEL) has been involved in nuclear reactor research and development for over 40 years. One of the first major projects involved the development of the nuclear powered aircraft engine. This project used mercury as a shielding medium and over several years a significant amount of mercury was spilled along the railroad tracks where the test engines were transported and stored. In addition, experiments with volume reduction of waste through a calcine process resulted in mercury contaminated calcine waste. Mercury was used as a catalyst in this process.

These wastes have been identified in a Removal Action Work Plan to be retorted and thereby separating the mercury from the various contaminated media. Lockheed Idaho Technologies Company awarded the Mercury Retort contract to ETAS Corporation and assigned Parsons Engineering Science, Inc. to manage the Removal Action field activities.

The mercury retort process is a mobile unit which consists of several trailer-mounted units, requiring electricity, propane, and a water supply. The conclusions presented in this paper will describe the performance of the retort unit for the various waste streams and show how the mobile unit is an effective means to retort waste and generate minimal secondary waste.

SITE BACKGROUND

The Idaho National Engineering Laboratory (INEL), formerly the National Reactor Testing Station (NRTS), encompasses 2,305 km2 (890 mi2), and is located approximately 32 km (20 mi) west of Idaho Falls, Idaho (Fig. 1). In 1949, the United States Atomic Energy Commission, now the Department of Energy (DOE), established the NRTS as a site for building and testing a variety of nuclear facilities. Since 1952, the INEL has also functioned as the storage facility of transuranic radionuclides and low-level radioactive waste. At present, the INEL supports engineering and

operations efforts of DOE and other Federal agencies in areas of nuclear safety research, reactor development, reactor operations and training, nuclear defense materials production, waste management technology development, and energy technology and conservation programs. The DOE Idaho Field Office (DOE-ID), having responsibility for the INEL, designates authority to operate the INEL to government contractors. The primary contractor for DOE-ID at the INEL is Lockheed Idaho Technologies Company (LITCO), which provides managing and operating services to the majority of INEL facilities. The remedial design/remedial action contractor for LITCO at the INEL is Parsons Engineering Science, Inc., which provides complete remedial design and remedial action (RD/RA) services.

Fig. 1

ORIGIN OF MERCURY CONTAMINATED MATERIAL

Calcine Material

In the late 1950's and early 1960's, research teams at the INEL conducted developmental work on the recovery of uranium from spent nuclear fuels and the subsequent treatment of the resulting wastes. Solutions from which the uranium had been extracted were calcined to reduce their volume and make them easier to handle and dispose. In various pilot studies, carried out at the Central Facilities Area (CFA) Chemical Engineering Laboratory at the INEL site, simulated aluminum clad fuels were treated, with mercury added as a catalyst in the calcining process. The calcined material and elemental mercury was disposed in a depression near the test facility and eventually covered with gravel. Through different mechanisms, the calcine waste was partially exposed to the elements. Subsequent investigations determined that the calcined material exhibited low-level radioactivity (from radioactive tracers, Cesium-137, Strontium-90, Ruthenium-106, and unidentified Uranium isotopes used during the calcine experiments) and mercury concentrations above 260 mg/kg. The waste calcine and surrounding contaminated soil were retrieved and containerized for future treatment. In a fluidized-bed calciner, the calcined material was formed at temperatures between 400 and 500 degrees Celsius. Held at this temperature only long enough to drive off the water coming in with the feed, the material left a large percentage of the feed mercury in the calcine when the calcine cooled. Though the calcine is porous, the mercury had insufficient time to vaporize and migrate out of the calcined material. A significant portion of this mercury is still present in the calcine in the form of mercury oxide.

The selected remedial action will meet final concentration limits of less than 0.2 mg/L mercury as determined by the Toxicity Characteristic Leaching Procedure, Environmental Protection Agency (EPA) analysis Method 1311, and less than 80 mg/kg total mercury as determined by EPA Method 7470/7471 (SW846).

Test Area North Material

In addition to the calcine waste, a second waste stream will be treated. This waste stream consists of mercury contaminated soil and railroad ballast contaminated by a large spill of elemental mercury at the INEL's Test Area North (TAN) in 1958. This mercury contaminated material was excavated and containerized in 1994. Sample results indicated that the material exhibited low level radioactivity (Cesium-137 and Strontium-90) and mercury concentrations above 450 mg/kg. The radioactive isotopes , Cesium-137 and Strontium-90, are the result of radioactive storage in the immediate vicinity.

The selected remedial activity will meet final concentration limits of less than 0.2 mg/L mercury as determined by the Toxicity Characteristic Leaching Procedure, Environmental Protection Agency (EPA) analysis Method 1311, and less than 4 mg/kg total mercury as determined by EPA Method 7470/7471 (SW846).

MERCURY RETORT SYSTEM

ETAS Corporation is not only the operator of the mercury retort system, but ETAS developed, designed, and fabricated the mercury retort system as well.

System Components

The mercury retort process entails a mobile unit which consists of trailer-mounted sub-systems with multiple components or units in each subsystem (Fig. 2). The four subsystems are:

- Feed System
- Retort Main System
- Vapor Recovery System
- Material Discharge System

Fig. 2

Feed System

Vacuum Unit

A vacuum unit was initially used to transfer material from the drums and boxes into the retort main system. The vacuum unit, an off-the-shelf manufactured system, is equipped with integral dust controls, including a multi-stage cyclone and a HEPA filter. A slide gate between the vacuum unit and the retort provided the necessary seal between the units.

The vacuum system was designed to minimize blowing dust during transfer of dry material from the waste containers. As treatment progressed, monitoring showed that almost all the material was at least damp and much was very wet, producing severe caking in the vacuum unit. Subsequently, an auger feed was tried, and finally a conveyor belt system installed. After the material was shoveled from waste containers into a hopper at ground level, a fully enclosed conveyor belt dumped the material onto a slide gate. The slide gate opened periodically to feed the material into the retort. The conveyor enclosure minimizes blowing material. The slide gate seals the feed end of the retort and provides for circulation of gases through the retort and into the vapor recover system.

Retort System

The Retort Main System consists of the following components/units:

- The Retort Chamber
- Burners and Mechanical Controls
- Gas Supply Piping Components
- Burner Control Cabinet
- Power and Motor Control Cabinet

Retort Chamber

The Retort Chamber is a heated assembly, which consists of a rotating process tube mounted axially inside a refractory lined shell. The rotating drum is lined with curved vanes, which both propel material forward and mix the material as the drum rotates. This not only conveys the material, but assures that the material is heated quickly and evenly. The unit rotates from 4 to 25 rpm, which allows a material transfer rate of 0.54 to 4.51 m3 (0.7 to 5.9 yd3) per hour. Constructed entirely of 304 steel, the rotating drum sits in the refractory lined outer tube chamber, which houses the burners. The curved shape of this chamber directs the burner flames around the tube for efficient heating.

Burners & Mechanical Controls

The burners essentially fire horizontally under the process tube for minimal flame impingement on the tube. The radiant heat of the flame and the convective heat of the hot combustion products heat the tube, which transfers heat to the product traveling through the tube.

Although the burners can be individually started and stopped, the burner firing rates are controlled by three process tube zone temperature controllers, with each controller driving three burners. ETAS has included several automatic features to provide remote and dual temperature controller set points, and will provide "cutback" logic in case stack temperatures become too high.

Rated for 1.0 MBTU/H firing propane, each burner consists of an integral fan housing directly coupled to the fan motor, gas burner and blast tube, gas pilot/igniter assembly, flame detector port, and viewport. The burners utilize modulating motors to control firing rates and fuel/air ratios. The system is equipped with temperature controllers for each of the three process tube zones; each controller will provide a control signal to regulate the modulating motors on all three burners for that zone.

Gas Supply Piping Components

Propane to the main fuel and pilot on each burner is turned on and off by the burner safety shutoff valves, with the fuel rates being controlled by the regulators and the main fuel modulating control valve. All safety shutoff valves are of the automatic type. The automatic shutoff valves provide for automatic shutdown of the fuel feed during certain conditions, and further provide for a single button shutdown during any emergency situations.

Burner Control Cabinet

The burner control cabinet is an air purged, free standing cabinet containing all burner system controls, indicators, and alarms.

The entire system is designed for maximum burner operating flexibility and maximum user friendliness. The system utilizes individual flame programmers for each burner, each with its own status indicator visible through a window on the cabinet door. ETAS included one enhanced display module designed to automatically switch to display the operating burner, providing enhanced sequence and alarm information for that particular burner. In addition, the system includes extensive alarm logic and indications for virtually all operating conditions, as well as main system purge controls and indicators.

Power and Motor Control Cabinet

This cabinet is an air purged, free standing cabinet, containing circuit breakers and motor starters for all nine unit motors, all nine burner motors, and control power circuit breakers for the burner control cabinet. Main 480 volt 60 cycle 3 phase (480/60/3) power and all nine unit motors can be controlled by the respective start/stop push buttons, and are indicated by the respective lights mounted on the cabinet door. The primary side of these transformers are protected by two-pole circuit breakers. The main logic section and each of the nine burner sections are protected by individual single-pole circuit breakers.

Vapor Recovery System

The vapor recovery system (VRS), having its own control cabinet, is a stand-alone system designed specifically for the removal of mercury. The entire Vapor Recovery process is performed under a vacuum created by a blower, which pulls the vapor stream through the VRS and pushes the final

vapor stream out through the carbon canisters. This vacuum assures that no pressure can build in the vapor recovery system. Pressure buildups have been shown to cause leaks in retort operations without vacuum assisted vapor recovery systems.

The VRS consists of three main removal sections, followed by carbon canisters. These sections are the Spray Scrubber, the Sieve Tray Scrubber, and the Vapor Separator.

Spray Scrubber

Material first passes through the spray scrubber, a direct contact heat exchanger which condenses the bulk of the vapors. This scrubber is powered by process water, which in turn is cooled by an indirect heat exchanger linked to a cooling tower.

Sieve Tray Scrubber

The Sieve Tray Scrubber condenses any remaining condensable vapors which escape condensation in the Spray Scrubber. The Sieve Tray Scrubber is also a direct contact heat exchanger, linked to the same cooling tower and indirect heat exchanger as the Spray Scrubber. Condensed vapors from both scrubbers are collected in a common surge tank, where baffles at the bottom of the tank capture mercury and particulate matter from the process water stream.

Vapor Separator

The vapor separator, the last piece in the system, provides a mist eliminator and a low vapor velocity zone to collect mist and return it to the surge tank.

Product Cooler System

An airlock provides a vapor seal between the retort and the material discharge system.

Airlock

The airlock is a rotary eight-vane unit, which provides a high-quality seal against air leaks. As the airlock rotates (at approximately 40 RPM), material passes via the rotating vanes from one end of the airlock to the other. The unit has a 1.5 HP motor and uses sizable gear reducers to achieve the 40 RPM working speed.

Product Loading System

The material discharge system cools the treated material while moving the material from the retort into storage boxes. The cooling and transfer is accomplished by three totally enclosed screw augers, two of which have coolant passing through jackets. This system shares a cooling tower and refrigeration with the Vapor Recovery System.

SYSTEM PERFORMANCE

Production Rates

The retort system has proven effective at meeting the treatment standards. Production rates have not reached the maximum potential of the system's capacity due to site specific issues. Feed material high moisture content in conjunction with freezing temperatures, size gradation of the railroad ballast relative to the material handling system design, and foreign objects, such as railroad spikes and construction debris, have all been contributing factors to the production rates contained in the feed material.

Treatment Results

Based on more than 20 samples taken and analyzed for total mercury and/or TCLP, the retort system is demonstrably effective in removing mercury from various waste streams, as summarized in Table I.

Table I

CONCLUSIONS

Material Handling Considerations

Subcontractors must evaluate the feed stock and determine the appropriate material handling equipment needed to handle the feed material and all expected worst case conditions. A proactive design that addresses the above concerns may require a greater initial capital cost, but consistently proves less expensive in the long run when compared to the costs of frequent mini-design changes occurring as situations arise.

Cold Weather Operations

Manual labor operations and work activities in unprotected areas in the winter may result in hypothermia or exhaustion caused by working in heavy clothing and respirators. Equipment, particularly wetted systems, must be protected from freezing, especially during non-operating periods. Employee well-being is directly proportional to production and every effort must be made to account for the expected climate during the project execution.

DOE Contracting

Subcontracts let at DOE facilities in the future will rely more heavily on fixed price lump sums or unit rates for fixed quantity contracts. These contracts will require subcontractors to incur more initial financial risk: the bulk of their payment will be received only after their services have been both completed and certified as meeting all performance specifications.

This requires bidders to ask for more clarification so that they are fully aware of the site conditions, as change orders and overruns are being tolerated less and being funded even less.

If conditions are accepted by the subcontractor, then it is assumed by the contractor that the subcontractor is comfortable with the site conditions and has bid the project accordingly.

30-21

CHARACTERIZATION STUDY OF RADIOACTIVELY CONTAMINATED SOILS FROM THE TONAWANDA FUSRAP SITE

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ABSTRACT

The National Air and Radiation Environmental Laboratory (NAREL) received eight whole soil samples from four individual sites collectively referred to as the Tonawanda FUSRAP Site. The four sites are Ashland I, Ashland II, Linde, and Seaway. The primary radioactive contaminants in the soils are uranium-235, uranium-238, radium-226, thorium-230, and their decay products. Two samples were received from each site, one sample collected to represent the maximum levels of radioactivity (Bias) and one to represent the average levels (Systematic). The samples were designated as

Ash I Bias, Ash I Systematic, Ash II Bias, Ash II Systematic, Linde Bias, Linde Systematic, Seaway Bias, and Seaway Systematic.

The primary objectives of this initial characterization of the samples were to: 1) determine if particle-size separation techniques could be effective in reducing the volume of contaminated soil, 2) identify any radiominerals present, 3) determine the presence and magnitude of contamination associated with particles of high density and ferromagnetism, and 4) determine if chemical scout extraction tests are effective in reducing the volume of contaminated soil in these samples. Particle-size separation by wet sieving and hydroclassification was performed after the samples were subjected to particle liberation by vigorous washing. Fractions were analyzed for radionuclides, and select fractions were subjected to petrographic analysis after density separation at 2.89 g/cc. Magnetic separations, using a hand-held magnet, were also performed on the heavy-mineral fractions to separate the ferromagnetic particles. The heavy-mineral nonmagnetic and magnetic fractions were analyzed for select radionuclides. Scout chemical extractions were performed with hot nitric acid on the samples that were judged to be amenable to contaminant reduction by the technology: four Systematic Tonawanda samples, the +50 and -50-mesh portions of the Linde Bias sample, and the +50-mesh fractions of the Linde Systematic sample. Analysis of data from the eight Tonawanda samples examined in this study demonstrates that particle-size separation alone will not provide a successful remediation solution for any of the sites. Although significant concentrations of the radionuclides were identified in the heavy-mineral fractions of materials from the eight samples, there were insufficient quantities to affect the overall radionuclide concentrations by their removal. Ferromagnetic materials separated from the heavy-mineral fractions also contained elevated concentrations of radionuclides; but these were, likewise, insufficient in quantity to affect the remediation potential by their removal. The results of the 3M nitric acid extraction indicate that the Linde +50-mesh material, particularly the Bias soil, shows promise for meeting cleanup criteria with additional chemical extraction.

INTRODUCTION

The National Air and Radiation Environmental Laboratory (NAREL) received eight whole soil samples from four individual sites collectively referred to as the Tonawanda FUSRAP Site. The four sites are Ashland I, Ashland II, Linde, and Seaway. Two samples were received from each site, one sample collected to represent the maximum levels of radioactivity (Bias) and the other to represent the average levels (Systematic). The samples were designated as Ash I Bias, Ash I Systematic, etc.The primary objectives of this initial characterization of the samples were to: 1) determine if particle-size separation techniques could be effective in reducing the volume of contaminated soil, 2) identify any radiominerals present, 3) determine the presence and magnitude of contamination associated with particles of high density and ferromagnetism, and 4) determine if chemical scout extraction tests are effective in reducing the volume of contaminated soil in these samples.

The radionuclide cleanup criteria for the Tonawanda FUSRAP site soils (in pCi/g) are: radium-226, 5-15; thorium-230, 5-15; thorium-232, 5-15; uranium-238, 28.4, and total uranium, 60.

EXPERIMENTAL

Gamma Spectrometry

Whole soil samples, particle-size fractions, and wash waters were analyzed for gamma-emitting radionuclides using high-purity germanium detectors (1). The wash water samples were counted for 1000 min, and the soil samples were counted for 100 or 1000 min, depending on the levels of radioactivity. With the relatively large uranium concentrations in the samples, erroneous Ra-226 values were obtained from the gamma spectrometry analysis program. Values cited for Ra-226 are based, therefore, on the Pb-214 values reported by the program and are more indicative of the actual Ra-226 concentrations.

Alpha Spectrometry

Analyses for uranium and thorium radionuclides were performed by alpha spectrometry (2). Uranium was extracted, coprecipitated with lanthanum fluoride carrier, and analyzed by alpha spectrometry. Thorium was separated by ion-exchange chromatography, coprecipitated with lanthanum fluoride carrier, and analyzed by alpha spectrometry.

Sample Preparation

The eight whole soil samples were received at the NAREL and initially screened for gross beta/gamma activity using a Geiger-Mueller portable survey instrument. After initial screening, each sample was thoroughly mixed and split into 400-mL aliquots using a riffler. The aliquots were weighed, dried at 60C for 48 hr, and reweighed. Four aliquots from each whole soil were analyzed by gamma spectrometry. The whole soil samples taken from Ashland I and Linde (Ash I Bias, Ash I Systematic, Linde Bias, and Linde Systematic) were also analyzed for uranium and thorium by alpha spectrometry, after they were ashed at 565C for 72 hr.

Vigorous Wash

The vigorous washing process liberates small particles from large particles without generating excessive fines. After sample preparation, each aliquot from every soil was vigorously washed in water for 30 minute in a Nalgene container at a rotational velocity of 350 rpm with a liquid-to-solid ratio of 4 mL/1g (3).

Wet Sieving

After vigorous washing, an aliquot of each soil sample was fractionated by wet sieving with American Standard Testing of Materials (ASTM) standard sieves (4). The aliquots were separated into particle-size fractions at 4 mesh (4.75mm), 8 mesh (2.38mm), 16mesh (1.19mm), 30 mesh (0.590mm), 50 mesh (0.297mm), 100 mesh (0.149mm), 200mesh (0.074mm), and 400 mesh (0.037mm). The fractions were dried at 60C for 48 hr, weighed, and analyzed for radionuclide content by gamma spectrometry. Analysis by alpha spectrometry was performed on only fractions from two of the four sites, Ashland I and Linde.

Process Water

Water from the vigorous wash and separation procedures for each sample was collected and a Percol 788N flocculant was added to settle suspended material. The water was then filtered through a 0.025-mm pore filter paper to separate suspended solids from the wash water. The material retained on the filter was analyzed by gammaspectrometry and, if called for, by alpha spectrometry and reported as the -400 activity. An aliquot of filtered wash water from each of the eight whole soil samples was analyzed by gamma spectrometry.

Petrographic Study

Petrographic examination was performed on the eight radioactive soil samples to determine the minerals or materials comprising the radioactive particles as well as the physical properties and composition of the host

materials. The petrographic tests were conducted according to the EPA protocol procedure for radioactive soils (5).

Soil-size fractions from vigorous washing and wet sieving were examined for mineral identification and physical properties by an optical petrographic microscope and a binocular microscope. In order to provide an assessment of the heavy-mineral fractions that frequently contain a major portion of the radioactive materials, density separations using the sink-float method with sodium polytungstate were performed on the -50/+100, -100/+200, and -200/+400-mesh fractions to remove heavy minerals (specific gravity greater than 2.89) and to facilitate mineral identification and preparation of photomicrographs. The gravel material on the 4-mesh sieve and materials on the 8, 16, and 30-mesh-size sieves was examined under the binocular microscope. A minimum of 100 particles were counted; all particles were counted if 100 were not present. Particle-size material from each fraction was divided into rock, mineral, slag, and other-mineral categories to illustrate the make-up of the soil sample. The materials on the 50, 100, 200, and 400-mesh sieves were examined under both the binocular and optical petrographic microscope. The material examined under the petrographic microscope was immersed in an index oil of 1.544 refractive index. Photomicrographs of significant features were made in both transmitted and reflected light.

Magnetic separations, using a hand-held magnet, were also performed on the heavy-mineral fractions to separate the ferromagnetic particles. The heavy-mineral nonmagnetic and magnetic fractions were analyzed by alpha spectrometry.

Scout Chemical Extractions

Scout chemical extractions using hot nitric acid (HNO3) were performed on Systematic whole soil samples collected at each of the four sites. In addition, nitric acid extractions were conducted on the combined +50-mesh fraction and -50-mesh fraction of Linde Bias samples and +50-mesh fraction of the Linde Systematic samples.

The nitric acid extractions were performed on aliquots (approximately 50 mL) from each Systematic soil prepared by proportionally recombining each of the wet-sieved fractions from the particle-size study described above. The resultant recombined fractions represented the whole soil. The aliquots used for extraction of the Linde +50 and -50-mesh fractions were similarly prepared from the previously sieved material.

For the chemical extraction, a spherical reaction kettle was closed with a 4-neck cover and fitted with a reflux condenser, centigrade thermometer, stirrer with paddle, and thermocouple assembly. The stirrer was rotated by a high-torque electric motor. Stirring speed was monitored every five minutes by an optical tachometer. The thermocouple assembly consisted of a iron-constantan thermocouple sealed in a glass tube to protect it from acid corrosion and packed with glass wool to aid the transfer of heat from the glass tubing to a thermocouple. Heat was provided by a mantle whose temperature was monitored and controlled by a Glas-Col Digitrol connected to the thermocouple and the variable autotransformer. Visual monitoring of temperature was also performed by observation of the centigrade thermometer.

Each aliquot was extracted with 3M nitric acid (5 mL/g of soil) heated to 90C while stirring at a nominal 350rpm. The sample was introduced to the reaction flask with a spatula over a five-minute period to assure complete contact between the reagent and the sample. The sample was extracted for 60 min, then filtered through a Buchner funnel, and rinsed

using hot 3M HNO3 followed by hot water. The resulting filtrate was again filtered over a membrane filter using a Millipore filtration apparatus to collect any fine particulates. The whole soil, extracted soil, membrane filter, and filtrate were analyzed for radium by gamma spectroscopy and for uranium and thorium by alpha spectrometry.

RESULTS AND DISCUSSION

Particle-Size Distributions

The particle-size distributions for Ash I, Ash II, and Seaway are similar with at least 49 percent -400-mesh material, as illustrated in Fig. 1. The Ash I Systematic soil contains notably more gravel and coarse sand-sized material than the Ash I Bias and both Ash II and Seaway soils. The Ash I, Ash II, and Seaway soils all contain from 50 to 60 percent -400-mesh (silt and clay) material. The Linde Bias and Systematic soils are similar in their distributions with 41 and 37 percent -400-mesh material, respectively. These soils are notably different from the other soils in that they contain 45 percent +50-mesh material.

Fig. 1

Radionuclide Distributions

The primary radionuclides above background levels identified in the Tonawanda samples are members of the U-238 and U-235 decay series. Radionuclides associated with the Th-232 series are present at quantities within the range of natural background. The nuclides of primary concern are U-234, -235, and -238; Th-234 and -230; and Ra-226. Gamma spectrometry data were collected for all whole soils and soil fractions, and those from alpha spectrometry (U and Th) were collected for the AshI and Linde samples. Uranium and thorium analyses were not performed on the Ash II or Seaway soils.

The concentrations of the primary radionuclides of the U-238 decay series in the eight whole samples are shown in Fig. 2. With the possible exceptions of the Linde samples, the uranium series radionuclides do not appear to be in equilibrium. (While the radium-226 values are about one-half those of the Th-230 and U-234/238, the great inaccuracy in the Ra-226 values, discussed earlier, does not preclude secular equilibrium.) With the exception of the Ash II Systematic soil samples, the Th-230 concentration of the remaining sample is higher, by up to 18 times that of the U-234/238 values. The Ra-226 concentrations are approximately one-half those of U-234/238 in the Linde samples, approximately equal in the Seaway samples, and lower, by up to a factor of 13, in the Ash I and Ash II samples.

Fig. 2

The relative relationships between uranium-series radionuclide concentrations in the whole soil remain stable for the separated fractions in the Ash I soils. The concentrations of the radionuclides are relatively constant throughout the particle sizes. There is a general inverse relationship between particle size and radionuclide concentrations for all the radionuclides in the Linde samples. However, the Th-230 is more highly concentrated in the smaller size fractions than are Ra-226 or the uranium nuclides.

Ash I Samples

The radionuclide with the highest concentrations in the Ash I samples is Th-230. The average concentrations of Th-230 are 2,100pCi/g and 1,526pCi/g in the whole soils of AshI Bias and Ash I Systematic, respectively. The values in the sieved fractions range from 394 to 2,585pCi/g (coarse to fine) for Ash I Bias and from 168 to 3,723pCi/g for

Ash I Systematic. Th-234, U-234, and U-238 are present in approximately equal concentrations throughout all size fractions in both samples. The values range from approximately 20 to 300pCi/g (coarse to fine) for Ash I Bias and from approximately 20 to 200pCi/g in Ash I Systematic. The Ra-226 concentrations are 71pCi/g in the Ash I Bias and 65 in the Ash I Systematic whole soils. Concentrations in the size fractions range from 17 to 125 and from 9 to 68 (coarse to fine) in the size fractions of the Bias and Systematic samples, respectively. Because of the magnitude of the radionuclide concentrations throughout the size fractions, for both Ash I samples, particle-size separation alone will not achieve the remediation levels required to meet site cleanup criteria.

Ash II Samples

Only gamma spectrometry analyses were performed on the Ash II sample fractions, thus no data are available for Th-230 as a function of particle size. However, a Th-230 concentration of 1,545 pCi/g in one aliquot of the Bias whole soil sample and its particle-size distribution suggests that it is similar to Ash I Bias. The Th-230 concentration in the Ash II Systematic whole soil is 520 pCi/g.

The Th-234 values may be used to estimate the U-238 concentrations since these two nuclides should be in equilibrium. The average ratio of Th-234 to U-238, as determined on the samples for which alpha spectrometry was performed (Ash I Bias, Ash I Systematic, Linde Bias, and Linde Systematic) is 0.88; therefore, the uranium concentration can be estimated from the Th-234 values derived from gamma analysis. There is no valid way to estimate the Th-230 concentrations in each particle size. The Ra-226 concentrations in the whole soil samples are 51pCi/g for Ash II Bias and 38pCi/g in the Systematic sample. Based on the Th-234 gamma values, the U-234 and U-238 concentrations are approximately 120 and 70pCi/g in the AshII Bias and Systematic samples, respectively. The Ra-226 concentrations in the size fractions range from 10 to 96pCi/g (coarse to fine) in the Bias sample and from 7 to 61pCi/g in the Systematic sample; the Th-234 ranges from 31 to 248 pCi/g in the Bias sample and from 15 to 125 in the Systematic sample.

Since only the +4-mesh particle-size fractions, representing less than six percent of the Bias and Systematic samples, contain Ra-226 concentrations at or below 15pCi/g; particle-size separation alone will not serve as a volume reduction technique to achieve site cleanup criteria.

Linde Samples

The Linde Bias whole soil sample contains concentrations of U-234 and U-238 of approximately 300pCi/g, Ra-226 of 120pCi/g, and Th-230 of 245pCi/g (Fig. 2). The +50 material, which represents 45 percent of the sample, has concentrations of U-234 and U-238, Ra-226, and Th-230 of 105, 20, and 31pCi/g, respectively. The uranium is virtually evenly distributed in the fractions with particle sizes greater than 100mesh, with U-234 and U-238 above 200 pCi/g in each size fraction.

The Ra-226 and Th-230 concentrations are greater than 15pCi/g, and the uranium is above 60pCi/g in all size fractions with significant volumes. Therefore, particle-size separation techniques alone will not achieve results consistent with cleanup criteria. However, with the concentrations of +50-mesh fraction only slightly higher than the clean-up criteria, the results suggest a potential for volume reduction by attrition scrubbing of the +50 material in a solution that will extract the uranium. Attrition has the potential to reduce the radium and thorium

concentrations by removing contaminated surface material from the coarse fractions. If the uranium can be removed so that it is primarily concentrated in the -50 material, volume reduction of up to 50 percent may be possible.

The Linde Systematic whole soil sample contains concentrations of U-234 and U-238 of approximately 150pCi/g, Ra-226 of 81pCi/g, and Th-230 of 140pCi/g. The +100-mesh material, which represents 50 percent of the sample, has concentrations of U-234 and U-238, Ra-226, and Th-230 of 102 and 103, 26, and 37pCi/g, respectively. The uranium and Ra-226 are approximately evenly distributed within fractions having particle sizes greater than 200mesh. Results from the Linde Systematic samples suggest that attrition scrubbing might reduce the radioactivity in the +100-mesh material, and chemical extraction might reduce the uranium.

Seaway Samples

The Ra-226 and Th-234 concentrations in the Seaway Bias whole soil sample are 74 and 17pCi/g, respectively. The U-238 and U-234 are expected to be below 60pCi/g, based on the Th-234 values as determined by gamma spectrometry. Only the +4 mesh particle-size fraction has a Ra-226 concentration of 15pCi/g or less (8 percent of the material). No thorium alpha spectrometry was performed on the particle-size fractions. However, based on the fact that the whole soil sample contained 724 pCi/g of Th-230 like other samples in this study, no size fraction is expected to be at or below 15 pCi/g.

The Ra-226 and Th-234 concentrations in the Seaway Systematic whole soil sample are 33 and 13pCi/g, respectively. The U-238 and U-234 are expected to be below 60pCi/g, based on the Th-234 values determined by gamma spectrometry. The +50-mesh material has a Ra-226 concentration of 14pCi/g. However, the +50-mesh material represents less than 17 percent of the total soil.

In summary, analysis of data from the eight Tonawanda samples examined in this study demonstrates that particle-size separation alone will not provide a successful remediation solution for any of the sites. Particle-size separation alone is not capable of achieving significant volume reduction because a number of factors including: the relatively high concentrations of radionuclides; their uniform distribution, in most cases, throughout the range of particle sizes; and the high quantity of fines in most of the tested soils. Attrition studies on the +50 or +100-mesh material in combination with chemical extraction might, however, remediate up to 50 percent of the Linde soils.

Petrographic Study

The bulk composition of the soil samples from the Ashland sites is generally similar to the host constituents but differs from the host materials of the Linde and Seaway sites. The radioactive constituents are but minor constituents of the host material. However, because of the variance in the host constituents, the Ashland, Seaway, and Linde sample results are described separately in the following paragraphs.

Ashland Soil Samples

The Ashland Bias and Systematic soil samples contain mostly natural rock and mineral particles, significant amounts of slag materials, and minor amounts of other man-made materials and plant debris: 1) limestone, dolomiteite, and dolomite minerals constitute 40 to 60 percent of the gravel-sized (+4.76 mm) materials and three to 45 percent of sand-sized materials; this material is natural rock aggregate native to the area and essentially free of radioactivity above background levels; 2) calcareous

siltstone and minor claystone comprise from 20 to 33 percent of the gravel-size and approximately 50 percent of the sand-size material; these particles are essentially free of radioactivity above background levels; 3) quartz occurs in the sand-size fractions with greater abundance with decreasing grain size; in the finer sand fractions, the quartz generally averages about 20 percent; minor occurrences of feldspar are included in this category; this is natural material free of radioactivity above background levels; 4) man-made slag materials comprise from three to 40 percent of the gravel-sized particles with greatest abundance in the Ash II Bias and least abundance in the Ash I Systematic samples; the slag particles in sand-size material generally occur in lesser amounts; the solid crystalline slag has its highest concentration in the Bias samples; this material is anthropogenic material that has been identified in previous studies as containing radioactivity above background levels; 6) the black, glassy slag averages six percent of Ashland II Bias sample but generally less than two percent in the other Ashland samples; this material has also been related with radioactivity counts above background levels (6).

The heavy minerals separated by sink-float from the fine sand and coarse silt-size fractions comprise between two and nine percent of the fractions separated on the Ashland samples. To facilitate identification, the heavy minerals were further separated into a magnetic and nonmagnetic fraction. The magnetic fraction comprised 10 to 30 percent of the heavy-mineral fraction.

The magnetic fraction consists of black angular magnetite, red oxidized magnetite particles, and round metal balls. The distribution of metal balls in the three size fractions for the Ashland samples is listed in Table I:

Table I

The metal balls vary in color and luster suggesting variable properties and composition. Some resemble ion-exchange beads that collect uranium, and this specific material may possibly have radioactivity above background levels.

The nonmagnetic, heavy-mineral fraction comprises 70 to 90 percent of the heavy-mineral fraction. The opaque minerals comprise approximately one-third of the nonmagnetic heavy-mineral suite and, in order of decreasing abundance, include red hematite, ilmenite/leucoxene, metal balls, and black, angular uraninite. The latter is radioactive but is very minor in occurrence; it should be further examined by specific tests for opaque minerals for verification.

The transparent, nonmagnetic, heavy-mineral fraction is comprised of predominantly hornblende and garnet, minor epidote, zircon, hypersthene, and monazite, and trace amounts of rutile, chlorite, apatite, staurolite, sillimanite, and tourmaline. In addition, the Ash I Bias and Systematic samples contain appreciable melitite-rich slag particles.

The radioactive minerals present in the transparent fraction are zircon and monazite, which occur most abundantly in the Ash II Bias and Systematic samples. Zircon in the Bias samples ranges from trace amounts to five percent and in the systematic fractions from three to eight percent. Monazite in these samples ranges, respectively, from trace to two percent and from zero to one percent. In zircon, radioactive thorium or uranium is capable of as much as four percent substitution of the zirconium in the minerals structure. In monazite, the chief ore mineral

of thorium, thorium oxide, ranges between three and 10 percent, while uranium is generally present in lower percentage distribution (7).

The alpha spectrometry results of heavy-mineral, nonmagnetic material from the Ashland sample indicate that, in general, the activity is concentrated in the heavy-mineral magnetic fractions. However, the heavy-mineral magnetic fractions represent a small percentage of the whole soil; thus, the total radioactivity is minimal. By removing the heavy-mineral or heavy-mineral magnetic material, the specific activity in the whole soil either remains unchanged or increases slightly. This is true for both the U-238 and Th-230.

The organic materials in the Ashland soil samples occur in all size fractions in generally trace amounts. These materials, in order of abundance, include woody plant fragments, seed spores, snail shell, and diatoms. Other materials in the Ashland samples range from trace amounts to as much as four percent. They are comprised of cement, ceramic materials, glass beads, amorphous putty, and miscellaneous concrete aggregate particles.

Seaway Samples

In the Seaway Bias and Systematic soil samples, the most abundant materials are limestone and dolomite followed by sandstone and siltstone. The latter differ from the material found in the Ashland soil samples. Three slag varieties are generally similar to those observed in the Ashland samples. The quartz is also generally similar, but the heavy minerals are more abundant than the average Ashland sample: 1) limestone and dolomite average 19 and 17 percent of the Bias and Systematic sample, respectively; this is natural rock aggregate similar in physical properties to the Ashland samples; this rock is free of radioactivity above background levels; clastic sandstone and siltstone comprise respectively 21 and 12 percent of the Bias and Systematic samples, respectively; this material is generally free of radioactivity above background levels; 2) quartz grains average 32 and 40 percent of the Seaway Bias and Systematic soil samples, respectively; this fraction is free of radioactivity above background levels; 3) anthropogenic slag materials average 26 percent of the Seaway Bias and 20 percent of Seaway Systematic soil samples; the categories of slag are similar to the Ashland soil samples but occur in different proportions; it is essentially free of radioactivity above background and constitutes, respectively, 11 and five percent of the Seaway Bias and Systematic samples; the dense crystalline slag and glassy slag have radioactivity levels above background(6); the dense crystalline slag comprises 20 and 10 percent of the gravel-size particles of Seaway Bias and Systematic fractions, respectively, and lesser amounts in order of decreasing grain size.

The heavy minerals comprise eight, nine, and 11 percent of the -50/+100, -100/+200, and -200/+400 mesh-size fraction of the Seaway Bias, respectively, and nine, seven, and eight percent of the corresponding Seaway Systematic soil samples, respectively. The magnetic fraction of the heavy-mineral fraction ranges from 16 to 36 percent of the heavy minerals.

The magnetic fraction is comprised of black magnetite, reddish, oxidized magnetite particles and round metal balls. The distribution of metal balls in the three size fractions for the Seaway Bias and Systematic sample locations is similar to Ash II Systematic Samples described above.

The nonmagnetic fraction averages about three-fourths of the heavy-mineral fraction or about seven percent of the fine-sieve fractions. The mineral species are generally similar in abundance to the Ashland samples except for the radioactive minerals, which are several times more abundant. The relatively high zircon and monazite composition is, respectively, 18 and two percent of the Seaway Bias -100/+200-mesh fraction and 11 and two percent of the same size of Seaway Systematic sample. Of all the Tonawanda locations, the Seaway samples have the highest concentration of zircon and monazite, and this is reflected in the concentrations of Th-232 observed.

Alpha spectrometry results reveal that the activity is concentrated in the heavy-mineral magnetic fractions. However, as with the Ashland soils, the heavy-mineral magnetic fractions represent a small percentage of the whole soil; thus, the total concentration represented is minimal. By removing the heavy-mineral or heavy-mineral magnetic material,the specific activity for both the U-238 and Th-230 in the whole soil either remains unchanged or increases slightly.

The organic materials comprise from trace amounts to two percent of the Seaway samples; materials are plant fibers, and minor spores and snail shells. Brick, concrete, and cement constitute the bulk of other materials, which comprise from trace amounts to five percent of some sieve fractions.

Linde Samples

In the Linde Bias and Systematic soil samples, the predominant host material is a carbonate rock that is unique to these samples. These samples are also void of some of the siltstone/claystone or sandstone/siltstone found as abundant host material in the other Tonawanda samples: 1) white to light-gray carbonate rock comprises approximately 40 percent of the Linde soil samples; this rock resembles lime rock from Florida but not material native to the Tonawanda area, such as the limestone and dolomite sedimentary rock; it is probably carbonate rock produced or modified by an industrial process; 2) dark-gray to light-gray cherty limestone and dolomite particles comprise 21 percent of the Linde Bias and six percent of the Linde soil analyzed on the sieved material; in general, this is sedimentary rock native to the area and similar to that reported from the other Tonawanda soil samples; this material is essentially free of radioactivity above background; 3) dark-gray dense chert particles comprise from 10 to 14 percent of the gravel-sized material and not more than two percent of the finer-sieve fractions; the chert is derived from the limestone and dolomite rock and is free of radioactivity above background levels; 4) quartz on the average comprises 20 to 30 percent of the material retained on the sieve fractions of the Linde soil samples.; this fraction also contains very subordinate amounts of feldspar; it is free of radioactivity above background levels; 5) slag material consists of three to four percent cinder slag of similar composition as that from the other Tonawanda soil samples; this material is known to contain some radioactivity above background levels (6).

The heavy minerals (greater than 2.89 specific gravity) comprise 33, 23, and five percent, respectively, of the -50/+100, -100/+200, -200/+400-mesh size Linde Bias fractions and 31, 33, and five percent, respectively, of the Linde Systematic soil sample. The magnetic fraction of these samples ranges from 12 to 22 percent.

The magnetic, heavy-mineral fraction is comprised of similar materials as the other Tonawanda soil samples. The distribution of the metal balls in the three size fractions is similar to the Ash I Systematic sample except that the Linde samples contain a higher percentage of metal balls in the -100/+200 fraction than does the Ash I Systematic sample.

The nonmagnetic fraction comprises about 85 percent of the heavy-mineral fraction and contains appreciable dense, angular, slag particles. Some of the slag particles contain amorphous coatings. The other heavy minerals are generally in similar proportion as the Ashland soil samples.

Alpha spectrometry results indicate that, in general, the activity is concentrated in the heavy-mineral magnetic fraction, except for the Linde Systematic -50/+100 fraction. In this fraction the Th-230 activity, 17 pCi/g, is concentrated in the heavy-mineral nonmagnetic fraction. The specific activity reduction resulting from removal of the heavy minerals is from 60.5to 22.6 pCi/g in the total fraction. Like the Ashland and Seaway soils, the heavy-mineral and heavy-mineral magnetic fractions from the remaining Linde fractions represent a small percentage of the whole soil. Thus the total concentration represented is minimal.

Organic materials comprise trace amounts of woody plant material. The other materials, comprising up to five percent of gravel-sized fractions but generally less than two percent of finer fractions, consists of concrete, concrete aggregate (some with cement adhering to the particle surface) and red brick fragments. This material is free of radioactivity above background levels.

In summary, significant concentrations of the radionuclides were identified in the heavy-mineral fractions separated by sink-float methodology at 2.89 g/cc in samples from each of the eight locations. However, there were insufficient quantities to affect the overall radionuclide concentrations by their removal. Ferromagnetic materials separated from the heavy-mineral fractions also contained elevated concentrations of radionuclides; but they were, likewise, insufficient in quantity to affect the remediation potential by their removal.

The presence of slag materials identified by SEM (6) to contain uranium compounds warrants further study into the potential of froth floatation as a possible removal technology.

Chemical Extraction

The extraction results are summarized in Tables II and III. In these tables, the results are tabulated for the four critical radionuclides for the Tonawanda Site: Ra-226, U-234 and U-238, and Th-230.

Table II

Table III

Significant amounts of the soils were solubilized during the extractions. The solubilized amounts varied from 44% for the Linde Bias +50 mesh to 17% for the Linde Systematic sample. The reasons for this degree of solubilization require further study.

The "fractional activity recovered" column of Tables II and III provides a materials balance indicator by comparing the total radioactivity in the head sample (whole soil) to that contained in the end products (recovered soil, membrane filter, and filtrate). The values given are the ratio of the total radioactivity of the extraction products to the radioactivity of the head sample for the individual radionuclides. The values range from 0.6 to 1.6 and indicate that, within experimental error, no significant material was unaccounted for in the final measured products.

The Ra-226 removal percentages for the whole Systematic soils range from 79 percent (Ash I) to 86 percent (Seaway). The concentration in the extracted Seaway sample was reduced from 25 to 3.4 pCi/g. The uranium (U-234 and U-238) removal percentages for the systematic whole soils range from 32 percent (Seaway) to 73 percent (Linde). The low values for the U-234 removals on the Ash II and Seaway soils are considered to be an artifact of the inordinately high values in the recovered product, as compared to the U-238, reported for these samples. The combined uranium concentration for the Seaway sample is 25 pCi/g (U-234 + U-238). The Th-230 removal percentages for the systematic whole soils range from 65% (Ash I) to 84% (Seaway). The lowest Th-230 concentration for a recovered soil is 42.2 pCi/g for the Linde soil.

The hot nitric acid extractions on the Linde Bias +50-mesh material produced a product with Ra-226 and Th-230 less than 15 pCi/g (7.4 and 8.5, respectively) and combined U-234 and U-238 of 63 pCi/g. Additional extractions may have further reduced the radionuclide concentrations. Comparable results were achieved for the Linde Systematic +50-mesh material with the exception of Ra-226. The lack of radium removal in this sample is difficult to understand and may be the result of a laboratory error in counting the sample. Of note is the excellent removal percentages achieved for all radionuclides in the extraction on the Linde Bias -50-mesh fraction. It begs speculation on the results of a second extraction and on how the other seven soils would react to an extraction performed on only their -50-mesh fractions.

In summary, the hot 3M nitric acid extractions performed on the four Systematic soils did not remove sufficient quantities of the radionuclides from any sample to produce products that would meet the cleanup criteria for all radionuclides. However, there was no optimization of the extraction conditions, and sequential extractions were not performed. The removal percentages for all the radionuclides in all the samples, on the other hand are significant; for the Ash I and II soil samples, the Th-230 was not removed below a level that makes chemical extraction a promising process for meeting the cleanup criteria. The nitric acid extraction of the Linde and Seaway soils produced extracted products with radionuclide concentrations sufficiently proximate to site cleanup levels to warrant continuation of the scout chemical extraction studies as originally proposed.

The results of the 3M nitric acid extraction indicate that the Linde +50-mesh material, particularly the Bias soil, shows great promise for meeting cleanup criteria with additional chemical extraction. This process should be linked to the attrition study previously recommended to achieve the optimum removal.

Because of the exceptional results achieved on the chemical extraction of the -50-mesh material for the Linde Bias sample, a comparable experiment should be conducted on the Linde Systematic -50-mesh fractions.

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ADVANCES IN SOIL DECONTAMINATION METHODS FOR ROCKY FLATS SOILS

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ABSTRACT

Improved separation methods are needed to remove contaminants selectively and efficiently from soils at the Rocky Flats Environmental Technology Site (RFETS). The methods must meet appropriate state and federal regulations for residual contaminant concentrations, and to minimize waste generation, the radioactive contaminants need to be incorporated in as small a waste volume as possible. To meet these needs, advances have been made for decontaminating soils at the RFETS.

Based on recent decontamination studies with Rocky Flats (RF) soils, the americium and plutonium in the >0.5-millimeter soil fraction (>90 wt.%) can be lowered to <5 picocuries per gram of soil via wet sieving/size separation and rotary scrubbing/soil dispersion. The benefit of utilizing established wet screening technology is high volume soil decontamination. The contaminated fine fraction (clays) can be treated to further reduce the volume of contaminated soil.

INTRODUCTION

Soil decontamination studies have been conducted at Rocky Flats in the past (1-8). These studies evaluated wet and dry sieving, attrition scrubbing, rotary scrubbing, ultrasonic scrubbing, vibratory grinding, calcination, vitrification, flotation, acid leaching, hydraulic classification and mineral jigging. Various dispersants, surfactants, complexing agents and adsorbents were incorporated into these evaluations. The most promising technologies tested were wet sieving, attrition scrubbing, vibratory grinding, hydraulic classifying, acid leaching and the use of dispersants. Previous results showed that wet screening was effective in removing plutonium and americium from the larger soil particles, thereby concentrating the activity in the fine clay and silt fraction (6-8). Vibratory grinding was more effective at physically decontaminating RF soil than rotary scrubbing, and sodium hydroxide solution at pH 12.5 was found to be the best washing solution to lower actinide concentrations in RF soils (3). Recent work with wet sieving, scrubbing and a gravimetric separator (mineral jig) showed that sieving removed >98% of the activity in the <50 to 4.0-mm soil fraction, representing >50 wt.% of the soil, and attrition scrubbing was better

than wet sieving and rotary scrubbing in removing activity from the <4.0 to 0.42-mm fraction; the mineral jig was effective in removing actinide-containing clays from the <0.42-mm fraction (4).

As a part of the process to screen and select remedial options at the RFETS, a treatability study was recently performed to evaluate a series of soil decontamination techniques. The study assessed the technological capability of high-potential physical and chemical separation processes to decontaminate the soil media of its Pu and Am constituents. The results of this study are described below.

EXPERIMENTAL

Testing was conducted using soils collected from a location on the east side of the 903 Pad (9). Each wet sieve test was performed with a Tyler Model RX-24 sieve shaker and U.S. Standard sieves (20-cm diameter). A sieve lid (with spray head) and bottom pan (with drain tube) was used in conjunction with the sieves (see Fig. 1). In addition, a Masterflex peristaltic pump as well as a supply reservoir and receiving vessel (4-liter containers) was part of the wet sieving system. Several wet sieve tests were conducted in duplicate, each using 500 grams of soil with 2.5 liters of wash solution, to evaluate various types of wash solutions. The soil was wet sieved into five fractions (<50.0- to 12.5-mm, <12.5- to 4.0-mm, <4.0- to 0.5-mm, <0.5- to 0.063-mm, and <0.063-mm). Each of the soil fractions separated was dried at ambient temperature and weighed. Samples of the soil fractions were collected and sent to a certified laboratory for complete soil digestion and analysis of Pu-239, Pu-240 and Am-241 by specific alpha and gamma counting.

Fig. 1

The spent wash solutions from the wet sieve operations were filtered through coarse (Whatman No. 41 or 20-25 m) and fine (Whatman No. 42 or 2.5 m) paper. A portion of these wash solutions was syringe filtered through a 0.2 m membrane, and these samples were sent to the laboratory for specific Pu-239, Pu-240 and Am-241 analysis.

A subsequent wet sieve test was performed in duplicate using 500 grams of soil (from the recombined fractions of the sieving processes which identified the most effective wash solution) with an additional 2.5 liters of the fresh wash solution. The second sieving operation separated the soil into five particle size fractions (<50.0- to 12.5-mm, <12.5- to 4.0-mm, <4.0- to 0.5-mm, <0.5- to 0.063-mm, and <0.063-mm). Each of these fractions was dried and weighed. Samples of the soil fractions and wash solutions were collected and treated as described above prior to analysis. Two rotary scrub tests were performed. Two samples of 500 grams of soil were initially wet sieved into five fractions (<50.0- to 12.5-mm, <12.5- to 4.0-mm, <4.0- to 0.5-mm, <0.5- to 0.063-mm and <0.063-mm) using 2.5 liters of 0.1 M sodium hexametaphosphate (HMP) wash solution. Next, the separated soil fractions were recombined (except the <0.063 mm fraction) and placed into an 8-liter wide-mouth polyethylene bottle with 2.5-liters of fresh wash solution. The soil slurry in the bottle was then rotated on a rotary jar mill at 80 rpm: one batch for 8 hours and the other for 18 hours. Each of the rotary scrubbed soil slurries was then wet sieved into five fractions (<50.0- to 12.5-mm, <12.5- to 4.0-mm, <4.0- to 0.5-mm, <0.5- to 0.063-mm, and <0.063-mm) with 2.5 liters of fresh wash solution. The soil fractions and wash solutions generated were filtered and sampled as described above.

One attrition scrub test was performed with soil (<4.0- to 0.5-mm) which was previously wet sieved and rotary scrubbed. The soil was attrition

scrubbed with 0.1 M HMP wash solution at a soil to solution mass ratio of 1:5 for 10 minutes at 1,000 rpm using a laboratory attrition scrubber. This scrubbing device consisted of two three-bladed, stainless steel opposed-pitch propellers (7-cm diameter) mounted in tandem (4-cm apart) with one propeller at the end of a stainless steel drive shaft. The propeller assembly (lowered into a 1-liter stainless steel beaker containing the soil slurry) performed the scrubbing action. The attrition scrubbed soil was then wet sieved into three fractions (<4.0- to 0.5-mm, <0.5- to 0.063-mm, and <0.063-mm) with fresh wash solution. Each of these fractions was dried, weighed and sampled for analysis. The wash solutions generated during the experiment were filtered and sampled as described above.

A chemical leaching test was performed using the <4.0-mm soil fractions from the attrition scrub experiment and the <0.5-mm soil fractions from the rotary scrub experiments. The chemical extractant solution was 0.1 M sodium citrate with 0.1 M sodium dithionite. The soil and leaching solutions (a soil to solution mass ratio of 1:10) were placed into an 8-liter wide-mouth polyethylene bottle and rotated on a rotary jar mill at 290 rpm for 18 hours. The soil slurries were then wet sieved into three fractions (<4.0- to 0.5-mm, <0.5- to 0.063-mm, and <0.063mm) with 5 liters of water. The resultant soil and water were treated as described above.

RESULTS AND DISCUSSION

Different wash solutions tried made little difference in the weight percent of soil reporting to each of the size fractions (9). As expected, there was no significant soil dissolution or dispersion between the wash solutions. However, greater dispersion of the smaller particles from the larger particles could have been attributed to the characteristics of some of the reagents. Essentially 33-34 wt.% of the soil reported to the <50 to 12.5-mm fraction; 9-11% to the <12.5 to 4.0-mm fraction; 24-25% to the <4.0 to 0.5-mm fraction; 23-27% to the <0.5 to 0.063-mm fraction; and 5-8% to the 0.063-mm fraction. These data (all are averages of duplicate runs) are in good agreement with previous studies of RF soils (6).

Generally the actinide concentrations remaining in all the fractions above 4.0-mm were within the same range (1-3 pCi/g Am and 2-11 pCi/g Pu) except water washed soil (9); thus hot water works about as well as wash solutions containing phosphate and/or hydroxide ions. However, for the next smaller size soil fraction (<4 to 0.5-mm), a 0.1 M solution of HMP at pH 11 and hot water gave the lowest actinide results. The actinide concentration only varied between 1 and 4 pCi/g. The <0.5 to 0.06-mm fraction contained high concentrations of Am and Pu with the lowest amount after HMP washing. The HMP solution also decontaminated the <0.063-mm fraction and transferred some of the contamination to the liquid phase. All of these results as well as the data below are an average of duplicate runs.

An additional wet sieving operation with fresh HMP wash solution was performed during the second soil decontamination test to gauge if improvement in the effectiveness of the separation resulted from the supplemental washing and mechanical sieving action contact time. Data from this double wet sieving treatment process were used for comparison with the rotary and attrition scrubbing soil treatment processes, as they involved a secondary wet sieving procedure. Additionally, the second wet sieving operation separated the recombined fractions from the initial wet sieving operation into fractions of even finer particle size

discrimination. Measurements of the physically separated soil fractions were repeated to determine the mass of soil distributing into each of these fractions as well as the partitioning of the Pu and Am contaminants into each mass fraction. The results are shown in Figs. 2 and 3, and are described in detail elsewhere (9). The double wet sieving showed slightly improved decontamination compared to a single wet sieving operation.

Fig. 2

Fig 3

After an initial wet sieving wash with HMP, the >0.063-mm soil fractions were recombined and rotary-scrubbed for 8 and 18 hours with fresh HMP wash solution. The purpose of recombining the soil fractions was to provide enhanced scrubbing action of the fine soil particles by the coarse soil particles. The 18 hour time interval was selected to encourage maximum dispersion of the clay particles in the soil. Each soil slurry was wet sieved after rotary scrubbing using fresh HMP wash solution. Soil fractions were then analyzed for Pu and Am content, and the results are shown in Figs. 2 and 3. The rotary scrub promoted further decontamination of the larger fractions compared to wet screening, and the eight hour period provided approximately the same result as 18 hours, showing that 8 hours or less is a sufficient rotational time.

Recombined soil fractions between 0.5-mm and 4.0-mm, previously wet sieved and rotary scrubbed,were attrition scrubbed with fresh HMP wash solution for 10 minutes at 1,000 rpm. This soil slurry was then wet sieved using fresh wash solution, and the soil fractions and solutions analyzed for Pu and Am content. The results are shown in Figs. 2 and 3; although additional activity was removed from the <0.5 to 0.063-mm fraction with attrition scrubbing, the small decrease in decontamination would probably not warrant the use of attrition scrubbing in a large-scale process.

A solution of sodium dithionite with sodium citrate was tested with rotary scrubbing because of its ability to desorb Pu and Am from the fine soil fractions by leaching the transuranics into the aqueous phase. This solution was chosen because of favorable results from another experimental investigation (10). The <4.0-mm soil fractions from the attrition scrubbing experiments and the <0.5-mm soil fractions from the rotary scrubbing experiments were utilized in this evaluation, which provided the best conditions for decontamination of soils >0.063-mm. The dithionite-citrate mixture leached more actinide from the clay-silt fraction than HMP (see Figs. 2 and 3).

CONCLUSIONS

Different types of wash solutions made little difference in the quantity of soil reporting to each of the size fractions: 34, 10, 24, 25, and 6 wt.% reported to the <50 - 12.5, <12.5 - 4.0, <4.0 - 0.5, <0.5 - 0.063 and <0.063-mm fractions, respectively. Actinide concentrations remaining in all the fractions above 4.0-mm (except room temperature water washing) were within the same range (1-3 pCi/g Am and 2-11 pCi/g Pu). Thus, heated (near boiling) water works about as well to decontaminate the >4.0-mm soil size fractions as wash solutions containing phosphate and/or hydroxide ions. However, for the smaller size fractions, a 0.1 M solution of sodium hexametaphosphate at pH 6.5 gave the lowest actinide results as well as yielding the highest results for the wash solution, showing that actinide was transferred from the soil to the solution phase.

The most favorable decontamination conditions found in this study were a preliminary wet sieving with a sodium citrate/sodium dithionite solution

followed by a five hour rotary scrub and a second wet screening using citrate/dithionite wash solution; these conditions decontaminated about 87 wt.% of the soil to <2.4 pCi/g Pu activity and outperformed a double HMP wet sieving and a single wet sieving followed by either attrition or rotary scrubbing and a second wet sieving using HMP. To simplify wash solution recycle operations and reduce waste generation, hot water rotary scrubbing followed by wet screening with hot water may be sufficient to decontaminate the >0.5-mm soil particles; further decontamination of a smaller soil fraction may require chemical processing with HMP or citrate/dithionite.

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APPLICATION OF EPA QUALITY ASSURANCE PROCEDURES TO A SOIL CHARACTERIZATION STUDY AT THE DOE NEVADA TEST SITE

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ABSTRACT
The transfer, modification, and application of well formulated and tested quality assurance (QA) procedures from one project to another deserves consideration. The use of a proven QA program design could result in cost savings and the collection of data with a greater degree of confidence. To test this thesis, a QA program, originally developed for large nationwide Environmental Protection Agency (EPA) programs, was adapted and implemented in a site characterization study at the Department of Energy (DOE) Nevada Test Site to ensure that laboratory data satisfied pre-determined measurement quality objectives (MQOs). The QA procedures were adapted from EPA programs such as the National Acid Precipitation Assessment Program and the Environmental Monitoring and Assessment Program, and to a lesser degree, from programs operating under the requirements of the Comprehensive Environmental Recovery, Compensation and Liability Act (CERCLA) and the Resource Conservation and Recovery Act (RCRA). The QA design adopted the batch or lot concept, in which samples are organized into groups of samples for laboratory analysis. Approximately 10 percent of the total samples were measurement quality samples (non-blinds, blinds, and double-blinds), which were included in each batch to evaluate and control measurement uncertainty. Detectability was assessed using instrument detection limits and precision data for low-concentration samples. Precision was assessed using data from reference samples under a two-tiered system based on concentration ranges. Accuracy was investigated in terms of bias with respect to reference values. The results showed that QA concepts developed for previous nationwide EPA programs were successfully adapted for the site-specific DOE project.
INTRODUCTION
Protocols from well formulated and tested quality assurance (QA) and quality control (QC) programs are potentially applicable to what may appear, initially, as programmatically unrelated projects. This is not an endorsement of the unwavering application of one QA/QC program to different projects without an evaluation regarding applicability. However, the evaluation and transfer of a functional and efficient QA/QC program for its application to other projects certainly has merit. The use of a proven QA program results in obvious benefits, such as the reduction in time needed to develop a new design, and less obvious benefits, such as those resulting from the collection of data with a greater degree of confidence.
During the 1980s, the U.S. Environmental Protection Agency (EPA) brought together scientists and managers with expertise to integrate QA and QC into nationwide programs such as the National Acid Precipitation Assessment Program (NAPAP) and the Environmental Monitoring and Assessment Program (EMAP) for assessing the status and health of our nation's ecological resources (1,2). One consequence of their efforts was the development of protocols for environmental sampling and analysis of soil (3).
The subject of this paper is the transfer of the protocols developed for these nationwide EPA programs as well as programs operating under the

requirements of the Comprehensive Environmental Recovery, Compensation, and Liability Act (CERCLA) and amendments under the Superfund Amendments and Reauthorization Act (SARA), and the Resource Conservation and Recovery Act (RCRA), to a U.S. Department of Energy (DOE) site-specific project. Specifically, this paper focuses on a laboratory QA program implemented for a soil characterization study conducted at the DOE Area 5 Radioactive Waste Management Site (4) on the Nevada Test Site (NTS).

LABORATORY QUALITY ASSURANCE PROGRAM

The laboratory QA program for this soil characterization study was designed to increase the likelihood that the data satisfied or exceeded the data quality objectives (DQOs). DQOs, which are statements of the levels of uncertainty that data users are willing to accept in the data, are quantitative and qualitative goals established by the data users prior to the initiation of an investigation (5).

The sample measurement system employed by this soil characterization study consisted of sample collection, preparation, and analysis phases. Measurement uncertainty is a small part of the allowable overall data uncertainty (including uncertainty from possible spatial or data aggregation sources) defined by the data users because the quality of data can be quantified in relation to the DQOs, thereby allowing the data user to evaluate the hypotheses with a known level of confidence. Uncertainty that exceeds the DQOs renders the data less reliable in the judgement of the data users. The DQOs encompass all components of uncertainty resulting from sample measurement, operational activities, and population-based uncertainties (e.g., spatial variability). Measurement quality objectives (MQOs) are specific goals that clearly describe in a quantitative manner the data quality that is sought for each measurement phase being monitored.

The QA program for this soil characterization study included the following components: batch sample design, measurement quality samples, and data quality attributes. These aspects of the QA program are described below.

Batch Sample Design

The design of the QA program incorporated the concept of batch sample analysis, which is an adaptation of lot analysis familiar to manufacturing QA/QC programs. For this DOE soil characterization study, soil samples were combined by the QA staff into batches containing approximately 40 samples each. The batch sample design was based on the following assumptions and considerations:

- The primary sources of measurement uncertainty to be identified, controlled, and assessed are produced during the sample preparation and analysis phases;
- Each of the primary sources of measurement uncertainty can be considered a combination of several smaller component sources of uncertainty (e.g., sample analysis can include within-laboratory, within-batch, and among-batch uncertainties);
- Measurement uncertainty at any given phase (e.g., analytical within-batch precision) can be confidently evaluated using data from a minimum of 20 measurement quality samples of a given type (6). Because a large number of soil batches were expected to be analyzed, a sufficient number of samples would be present to make reliable among-batch statistical estimates;
- The quantitative within-batch MQOs can be attained if each batch satisfies predetermined acceptance criteria; and

Approximately 10 percent of the overall soil analysis effort is applied to the analysis of soil measurement quality samples for independent assessment purposes. Based on previous studies of this nature (7,8,9), this level of QA sample analysis is a reasonable proportion to achieve QA optimization. Also, batches of approximately 40 samples are sufficient for an effective soil QA program based on organizational, operational, and cost considerations.

Measurement Quality Samples

The specific QA acceptance criteria provided a balance between time and cost constraints and the data quality necessary to achieve the study objectives. To assess the quantitatively defined MQOs, a series of measurement quality samples was analyzed with the routine samples in a manner that was statistically relevant and that allowed conclusions to be drawn concerning measurement quality.

Measurement quality samples used in this study, although originally developed for the EPA's NAPAP and EMAP programs, were similar, or in some cases, equivalent to the samples used in EPA's CERCLA Contract Laboratory Program and in RCRA projects (Table I). This similarity occurs even though the QA/QC program for this study was primarily modeled after earlier EPA programs developed for tracking non-point acidic deposition and detecting long-term environmental change rather than CERCLA and RCRA projects that emphasized point-source contaminants. Sample terminology was changed slightly in order to adequately identify the different types of samples being used and the phases in which these samples entered the sample measurement stream. For instance, the analytical laboratory calibration check (ALCC) sample was introduced during the Analytical Laboratory phase of the measurement system and was used as a Calibration Check sample. The use of this terminology allows the data user to uniquely identify the status and utility of a sample in the measurement quality assessment scheme; it also accommodates the insertion of other types of samples into an ongoing QA/QC program (e.g., as necessary to assess field sampling error or overall measurement uncertainty, or to calculate a measurement system detection limit).

Table I

Measurement quality samples were of three types: non-blind, blind, and double-blind. By definition, a non-blind sample has both a concentration and a location within the batch that are known to the analyst. A blind sample has a concentration that is unknown to the analyst. A double-blind sample cannot be distinguished from a routine sample and has a concentration that is unknown to the analyst (10). For this soil characterization study, each measurement quality sample had a specific purpose in data evaluation and was used in a distinct manner to assess variability or adherence to the sample preparation and analysis protocols (Table II).

Table II

Two distinct groups of measurement quality samples, contractual compliance samples and laboratory QC samples, were randomly inserted into each batch to evaluate and contractually control the various laboratory components of measurement uncertainty. The contractual compliance samples, which were double-blind soil samples, allowed an independent assessment of measurement quality by the QA staff. The laboratory QC samples, which were non-blind and blind (both soil and liquid matrices), enabled the laboratory to control its measurement error (random and

systematic) in anticipation of satisfying contractual quality requirements. The QA staff also had access to all laboratory QC data.

Contractual Compliance Samples. Two varieties of contractual compliance samples were used in this study:

Analytical Laboratory Measurement Reference (ALMR) Sample - The ALMR sample is a median-concentration soil reference sample that was randomly assigned by the QA staff in duplicate or triplicate within each batch sent to an analytical laboratory. The ALMR sample material was collected in bulk at the NTS during routine sampling, and its matrix was representative of the routine samples analyzed in this study. The bulk sample was air-dried, disaggregated, sieved, homogenized, and subsampled by the QA staff. Aliquots of the subsamples underwent replicate analysis at a referee laboratory, and the resulting data were used to establish confidence intervals and tentative acceptance criteria for precision and accuracy. Known quantities of rock fragments were added to batched sets of ALMR samples to help ensure their suitability for independent double-blind evaluation of data quality by the QA staff. The ALMR sample data was used by the QA staff to estimate analytical within- and among-batch imprecision, bias and its sources (e.g., contamination or method error), and laboratory trends (e.g., interlaboratory and intralaboratory differences among batches). With the exception of the added rock fragments and its usage for precision assessments, the ALMR sample is roughly equivalent to the regional blind performance evaluation (PE) sample used commonly in CERCLA and RCRA.

Preparation Laboratory Homogenization Duplicate (PLHD) Sample - The PLHD sample was homogenized and split-subsampled at the sample preparation laboratory. One replicate remained at the laboratory as the first sample in the batch, and the other replicate was returned directly to the QA staff who reconstituted the sample with a known quantity of rock fragments and placed it randomly as a double-blind measurement quality sample in the next batch assembled. Confounded PLHD sample error (i.e., due to statistical confounding encompassing the sample preparation and analysis phases) can be distinguished by comparing variability of the different PLHD-and-routine sample pairs with the variability observed in the corresponding ALMR sample pairs. The PLHD was primarily used to estimate the sample preparation (homogenization and subsampling) error component of measurement uncertainty. In addition, the PLHD-and-routine sample pairs spanned multiple batches, which allowed the QA staff to use PLHD data to make assessments of among-batch precision at the analytical laboratory. With the exception of its double-blind character in this study, the PLHD sample is comparable to the internal laboratory duplicate (split) sample used occasionally in CERCLA and RCRA.

Laboratory QC Samples. Six types of laboratory QC samples were used in this soil characterization study:

Analytical Laboratory Control Reference (ALCR) Sample -- The ALCR sample is a soil reference sample of known concentration that the QA staff provides directly to the analytical laboratories along with corresponding reference values. The ALCR is used as an internal QC sample to control laboratory bias and to minimize the among-batch component of measurement uncertainty. The ALCR sample is roughly equivalent to a sample of certified reference material used commonly in CERCLA and RCRA.

Analytical Laboratory Control Duplicate (ALCD) Sample-- A duplicate subsample of the 12th soil sample in each batch was selected as the ALCD sample at the analytical laboratory. It is used as an internal check by

the QA and laboratory staff to ensure that predefined within-batch precision MQOs are being satisfied. The ALCD sample is equivalent to an internal laboratory duplicate (split) sample used occasionally in CERCLA and RCRA.

Analytical Laboratory Calibration Check (ALCC) Sample-- The ALCC sample is a liquid (or occasionally a solid) sample containing the analyte of interest at a concentration in the mid-calibration range. It is used to verify the calibration curve or mid-range of an instrument reading. The ALCC sample is equivalent to a calibration verification sample used commonly in CERCLA or a quality control check sample (QCCS) used in RCRA.

Analytical Laboratory Detection Check (ALDC) Sample-- The ALDC sample is a low-concentration liquid (or occasionally a solid) sample that contains the analyte of interest at a concentration specified by the QA staff. Typically, the concentration is two to three times above the established instrument detection limit. The primary uses of the ALDC sample are for the estimation of laboratory detection limits and as a check for significant baseline drift. The ALDC sample is roughly equivalent to a low-level calibration verification sample used occasionally in CERCLA or a low-level QCCS used in RCRA.

Analytical Laboratory Calibration Blank (ALCB) Sample-- The ALCB is a test for baseline drift. The ALCB sample is defined as a zero mg/L standard and contains only the matrix of the calibration standards (i.e., equivalent to the first standard in the set of standards). The ALCB sample is equivalent to a calibration blank sample used commonly in CERCLA and RCRA.

Analytical Laboratory Reagent Blank (ALRB) Sample-- The ALRB sample is a liquid (or occasionally a solid) sample composed of all the reagents in the same quantities used in actual sample analysis and which undergoes the same digestion and extraction procedures as an actual test sample. The ALRB reflects any analyte contamination from the sample matrix and apparatus used in the analytical procedure. The ALRB sample is roughly equivalent to a reagent blank or method blank sample used commonly in CERCLA and RCRA.

Data Quality Attributes

The attributes of data quality are qualitative and quantitative characteristics which provide an overall assessment of quality for the various measurement phases of the data collection activity. For the purposes of this report, measurement quality was defined in terms of the following four quality attributes:

Detectability is the determination of the low-range critical value of a soil characteristic that a method-specific procedure can reliably discern;

Precision is the level of agreement among multiple measurements of the same soil characteristic;

Accuracy is the level of agreement between an observed value and the true or accepted value of a soil characteristic; and

Completeness is the quantity of soil samples and soil sample data that are successfully collected with respect to the amount intended in the experimental design.

Detectability. The primary consideration in the evaluation of detectability is whether a measured sample value can be considered to vary significantly from the measured value of a sample blank. Detectability guidelines commonly are based on the number of standard deviations (SDs) the analytical signal varies from the mean value of

blank responses. In lieu of soil blank samples (which cannot be synthesized), laboratory stock solutions and low-concentration replicate soil samples were used to estimate the variability expected of blank samples.

The required instrument detection limit (RIDL) for each analytical parameter was defined prior to beginning sample analysis. The RIDL was based on the average variability expected among a series of five low-concentration ALDC samples analyzed in each batch of samples, and was calculated as three times the mean SD of numerous initial runs of ALDC samples. The actual instrument detection limit (AIDL) was calculated using real-time ALDC sample data and was used to assess contractual RIDL compliance by the analytical laboratories. Each AIDL served as an estimate of the lowest concentration of an analyte that a laboratory could reliably detect. Each parameter for each batch was required to have AIDL values less than or equal to its respective RIDL. If an AIDL exceeded its respective RIDL, the batch was reanalyzed for the parameter of interest. Incidentally, the RIDL is comparable to the contract-required detection limit (CRDL) used in CERCLA.

Precision and the Two-Tiered Analysis. Precision is the level of agreement among multiple measurements of the same soil characteristic or parameter. Laboratory imprecision was assessed for both within-batch and among-batch variability using data from the ALMR samples. The MQOs were established for within-batch imprecision estimates using data for the ALMRs from an external laboratory. The within-batch variability is usually only slightly lower than the among-batch variability. For comparison purposes, the MQOs were considered to be appropriate for both within-batch and among-batch variability. Imprecision of sample homogenization and subsampling efficiency at the sample preparation phase was assessed using data from the PLHD samples.

A primary feature of precision objectives for solid media like soil is the two-tiered system of characterizing MQOs (9). Some parameters, especially those in the middle to upper portion of the known concentration range, are best evaluated when the acceptable imprecision limits are variable with concentration and are expressed as a relative standard deviation (RSD) in percent. Other parameters, particularly those in the lower portion of the known concentration range, are best evaluated when the acceptable imprecision limits are absolute and are expressed as a SD in reporting units. This approach avoids setting restrictive precision objectives for low concentration soil samples, which generally are more difficult to extract and analyze with a high degree of relative precision.

The concentration scale of the samples is divided into two ranges or tiers separated by a calculated value called the knot (9). A knot value is used to distinguish low-concentration samples from median- and high-concentration samples. Most parameters have a distinct knot value which is determined by dividing the imprecision objective for the lower tier by the imprecision objective of the upper tier and multiplying by 100. The lower tier concentration range below the knot defines the region of the data in which the acceptable imprecision limit is an absolute value and is expressed as a fixed (constant) SD in reporting units. The upper tier of the concentration range above the knot defines the region of the data in which the acceptable precision limits vary directly with concentration and are expressed as %RSD. Imprecision below the knot can rise to quite high levels (if expressed as %RSD) and still satisfy the MQO as SD.

Accuracy. Accuracy is the degree of agreement between an observed value and the true or accepted value of a characteristic. Using data from the ALMR samples, the QA staff investigated accuracy from the perspectives of bias and laboratory trends. Analytical bias was considered to be the quantitative measure of inaccuracy, and laboratory trends served as quantitative and qualitative evaluations of laboratory performance over time. A range of acceptable ALMR values for each parameter was defined by a 95 percent confidence interval accuracy window. The window was calculated as two times the SD of extensive ALMR replicate aliquot analysis conducted by the referee laboratory prior to initiating the routine analysis. From a contractual compliance standpoint, any ALMR observation that occurred within an accuracy window (i.e., between the lower limit (LL) and upper limit (UL)), was considered to be acceptable (Fig. 1).

Fig. 1

The ALMR sample data were also examined for trends that could result from instrumental drift or other problems that might have occurred at the laboratory over the course of the analysis. For this purpose, standard control charts of the ALMR data were plotted in a temporal sequence of batch analysis (Fig. 1).

The external evaluation of accuracy by the QA staff was, by necessity, kept confidential to protect the integrity of the accuracy windows for the soil reference samples. For internal control of accuracy by the laboratory, an ALCR sample with established accuracy windows was provided. The laboratory staff plotted data on control charts to evaluate performance with respect to established confidence intervals. These charts were used to identify systematic or random bias during batch analysis.

Completeness. The laboratory was required to complete the specified processing tasks (criterion = 100 percent completeness) on all soil samples received batch-by-batch as identified on chain-of-custody tracking forms. Analytical completeness (criterion = 95 percent) was evaluated using routine sample data from the verified and validated data bases.

CONCLUSIONS

Quality assurance procedures were efficiently optimized and applied. The results of the study showed that the predefined MQOs were satisfied for detectability, precision, accuracy, and completeness. The quality of the data was thereby assured at the specified levels of acceptability established by the users. In conclusion, the quality assurance concepts first used by nationwide EPA programs were successfully adapted to the site-specific DOE project.

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30-25

FULL-SCALE DEMONSTRATION OF CLOSE-COUPLED SUBSURFACE BARRIER TECHNOLOGY

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ABSTRACT
The primary objective of this project is to develop and demonstrate a close-coupled barrier for the containment of subsurface waste or contaminant migration. A close-coupled barrier is produced by first installing a conventional cement grout curtain followed by a thin lining of a polymer grout. The resultant barrier is a cement-polymer composite that has economic benefits derived from the cement and performance

benefits from the durable and resistant polymer layer. The concept of close-coupled barrier technology is the combination the two technologies that were developed at Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL).

A full-scale demonstration barrier was installed at a cold site at the Hanford Geotechnical Test Facility, 400 Area, Hanford, Washington. The composite barrier was emplaced around and beneath a 15,000 liter tank. The tank was chosen to simulate a typical waste form that exist within the DOE Complex. The stresses induced on the waste form were evaluated during barrier construction. The barrier was constructed using conventional jet grouting techniques. Drilling was completed at a 45 angle to the ground, forming an ice cream cone shaped barrier. One row of cement columns was grouted in a circular pattern followed by a second row of columns between and tangential to the first row, in a honeycomb fashion. The primary barrier was placed by panel jet-grouting with a dual wall drill stem using a two part polymer grout. The polymer chosen is a high molecular weight acrylic manufactured by 3M Company. This paper will discuss the installation of the close-coupled barrier and the subsequent integrity verification.

INTRODUCTION

Numerous historical waste disposal sites located throughout the US Department of Energy (DOE) Complex have resulted in contamination of the subsurface. The waste sites include; underground storage tanks, piping systems, vaults, landfills, and other structures containing hazardous and mixed wastes. Consequently, efforts are being made to devise technologies that provide containment of waste sites either as a safety net to "catch" future contaminant leakage/migration or as an interim step while final remediation alternatives are developed. Subsurface barriers can increase the performance of the waste site and reduce or prevent the possibility of contaminant migration into local geologic media or groundwater. In addition to preventing contamination (and resultant clean-up costs) such in-situ treatment could result in large cost savings and reduced worker exposure, compared to conventional restoration costs (e.g., excavation, re-treatment and re-disposal of the waste). In addition, the legal ramifications for not treating many of these waste sites could be detrimental to the responsible parties.

The primary objective of this project is to develop and demonstrate a close-coupled barrier technology capable of containing waste forms within their existing subsurface transport, disposal, or storage structures. The containment will be a multibarrier of a cementitous grout followed by a polymer grout. The two grouts will be placed in a close-coupled fashion such that the polymer barrier is bonded to the cementitous barrier. The close-coupled barrier is produced by first installing a conventional cement grout barrier followed by a thin lining of a polymer grout. The resultant barrier is a cement polymer composite that has economic benefits derived from the cement and performance benefits from the durable and resistant polymer layer. The polymer barrier is thin (0.15 to 0.3 meters) and serves as the primary barrier to contaminant mobility. The cementitous barrier is thick (1.0 to 2.0 meters) and serves as a secondary barrier to contaminant mobility. It is essential that materials (grouts) and emplacement techniques are compatible; therefore, they must be developed and demonstrated simultaneously. This is not a trivial issue. Barrier materials must simultaneously be emplaceable, i.e., compatible with emplacement equipment and site geology, withstand a wide

variety of chemical, thermal, physical and radiological conditions, and meet acceptable longevity requirements. The concept of close-coupled barrier technology is the combination the two technologies being developed at Brookhaven National Laboratory (BNL) and Sandia National Laboratories (SNL).

BNL has been developing improved polymer-grout barrier materials for applications where impermeability and long-term durability are required (1,2). These materials have been used extensively in many commercial applications such as sewage and brine handling systems and electrolytic baths. Polymer grouts are candidates for high quality barrier materials due to their impermeability to gases and liquids, combined with their resistance to radiation, acidic, and alkaline environments (3). However, improved chemical and/or physical durability and performance increases the cost of the barrier grout.

SNL has been investigating placement methods and cementitious grouts for subsurface barriers. During the summer of FY94 SNL placed several pilot scale jet-grouted cement columns at a clean site near the Chemical Waste Landfill at Sandia. At the same time BNL was invited to demonstrate a polymer grout using the same placement equipment.

For a barrier where zero tolerance in leak rate is required it would be nearly impossible to achieve this goal using a cementitious grout. Large castings of hydraulic cements result invariably in cracking due to shrinkage, thermal stresses induced by the hydration reactions, and wet-dry cycling prevalent at arid sites. The improved, low permeability, high integrity polymer materials under investigation by BNL could achieve the permeability and durability goals, but might be costly. A joint venture was proposed by Brian Dwyer of SNL, John Heiser of BNL, and Steve Phillips (grouting contractor) of Westinghouse Hanford Company (WHC). SNL could design an economical cement grout curtain that would be used as a backdrop for the polymer curtain. A cementitious "bath tub" would be formed and the inside lined or flooded with a polymer binder. The resultant containment is a multibarrier of a cementitious grout followed by a polymer grout. The final composite barrier would have cost benefits from using mostly portland cement grout and have the performance benefits of the polymers from the inner lining.

Close-coupled barrier technology is applicable for final, interim, or emergency containment of subsurface waste forms. Consequently, when considering the diversity of technology application, the construction emplacement and material technology maturity, general site operational requirements, and regulatory compliance incentives, the close-coupled barrier system provides an alternative for any hazardous or mixed waste remediation plan.

This demonstration is currently funded by the Landfill Stabilization Focus Area (LFA) with roots in what is now the Plumes Focus Area (PFA). For the LFA close-coupled barriers have many applications. They can be used to contain buried waste and will provide a lower permeability, more durable and chemically resistant barrier than cement grout alone. The polymers are not expected to crack as easily as cement (wet-dry cycling) or slurry walls (solvent or organics). Close-coupled barriers are also useful in hot spot retrieval for containing contaminants while excavation and removal take place and may serve as shoring reducing the amount of contaminated soil. Utilization by PFA related projects include sealing off a source term (e.g., sealing a leaking UST or containing a subsurface spill of solvent) and preventing continued growth of a plume; thereby,

fixing the volume of waste. A data subset of the technology developed from the close-coupled barrier demonstration will include grouting with polymers. Using polymers by themselves will also prove useful to the DOE complex. Plumes or source terms can be surrounded by an inexpensive (e.g., AC-400 acrylate grout) to improve remediation efficiency for such technologies as in-situ air stripping of VOCs.

BACKGROUND

During FY94 small scale configurations (v-trough, cone, and 7x7 matrix) using cementitious grouts were installed via jet grouting. A single column was installed using a polymer grout. The FY95 demonstration installed a conical configuration barrier that is large enough to simulate numerous real sites. The cement and polymer were installed to form a close-coupled or composite barrier. The barrier was constructed to surround a simulated waste, a small (15,000 l) tank (see Fig. 1). The tank was chosen as a typical waste form that exist within the DOE Complex and also allows access to the subsurface inside the barrier for testing and monitoring.

Fig. 1

Test site

The site selected for the field-scale demonstration is the Geotechnical Test Facility (GTF) in the 400 Area ("Little Egypt") at the Hanford Site near Richland, Washington. This site was selected for several reasons: in geotechnical terms it is typical of many DOE facilities, the GTF was fully characterized and permitted for such a demonstration, the grouting contractor and required instrumentation and equipment (e.g., accelerometers, steel tank, etc.) are located nearby (eliminating mobilization/de-mobilization costs). The GTF was completed in FY82. It was originally designed to test and demonstrate burial ground subsidence control methods. The site is NEPA approved and well characterized and is described in a report Construction and Preliminary Description of a Geotechnical Test Facility at the Hanford Site, Richland, Washington by Phillips and Fischer (Rockwell Hanford Operations SD-RE-TI-048). Potential end users were identified and include BNL (chemical and glass pit remediation), INEL (hot spot retrieval) and Hanford (close-coupled barriers for UST leak repair). The GTF (Hanford) soil is a coarse sand to gravel; BNL is a coarse sand, free of clay lenses or cobble; and INEL is an alluvial/eolian deposit consisting of fine clay sized silts to coarse gravels of carbonaceous origin overlying basalt.

Jet grouting has been accomplished in all of these soil types. Soil type affects the effective diameter (jetting distance) of the column, for example in a clay soil the jetting distance will be slightly reduced due to the energy absorbing characteristics of clay. This effect will be minimal and in the worst case will require slightly reduced spacing of the installation bore holes (columns) or increased jetting pressures. The biggest impediment soil type could impose to jetting would be large cobble that could block the jetting pathway, which could result in a gap in the barrier. It is anticipated that with a close-coupled approach the cobble will become part of one or both of the barrier layers (since the jetting would occur parallel and perpendicular to the cobble; column jetting followed by panel jetting). Therefore the success of the demonstration was virtually independent of the test site soil type.

Prior to the demonstration the site has been prepared by the subcontractor. This includes the burial and backfill of a 15,000 liter tank and the installation of monitoring equipment. Eight monitoring wells

(every 45) were installed inside and outside the area to be enclosed by the barrier. These wells will be used for verification of the barrier integrity using perfluorocarbon tracers and for moisture determinations during water infiltration testing.

BARRIER INSTALLATION

This project demonstrates a systems approach to construction of a subsurface barrier. This includes the integration of barrier materials, emplacement equipment, verification techniques, and post monitoring instrumentation to produce a close-coupled engineered barrier.

The barrier was constructed using conventional jet grouting techniques. Jet grout curtains are constructed by injecting the grout through a pipe into the subsurface. The pipe has a drill tip on it which is used to drill the initial borehole. The pipe is then rotated while injecting the grout and slowly withdrawn from the ground. The high velocity jet masticates and intimately mixes the soil and grout which results in a column ~1 meter in diameter that resembles a pancake stack. The technique requires a pumpable grout that can be injected at pressures up 5-6,000 psi through a small orifice, typically 1 mm. This limits any aggregate additions to fine particle sizes. Most often, the jet grouting uses a low viscosity grout (<50 cps), and incorporates only the existing soils for aggregate. Jet grout curtains can be vertical using conventional drilling, or may be angled, or horizontal, using directional drilling.

The secondary cement layer was installed using column jet grouting as described above. The primary polymer layer was installed using panel jet grouting to reduce grout costs. Panel jet grouting is a simple refinement to conventional jet grouting. The tool is turned back and forth only a few degrees forming a bow tie shaped panel, rather than rotating the jetting tool 360 and forming a cylindrical column. This forms a thin panel, typically 30-40 centimeters wide. Panels are laid side by side with a slight overlap in order to form a continuous barrier. This method requires only a fraction of the grout volume compared to the volume of grout required for a full column.

For the demonstration the drill rig was a Casa Grande C6 unit owned by the Westinghouse Hanford Company. This is a track/trailer mounted jet grouting rig and is depicted in Fig. 2. The cement layer was installed during the summer of 1995. The complete installation of this layer took approximately one week. Drilling was completed at an approximately 45 angle to the ground, the columns are ~28 feet long forming an ice cream cone shaped barrier (see Fig. 1). A row of 60 cement columns was grouted in a circular pattern followed by a second row of columns in front of and touching the first row in a honey comb fashion. Columns were placed 24 inches on center and were ~30 inches in diameter producing a barrier 3-4 feet thick. The final cement barrier is ~40 feet in diameter at the top and extends approximately 20 to 22 feet below grade. This cement layer serves foremost as a backdrop for the polymer layer and secondly as a redundant, albeit less durable, barrier.

Fig. 2

During the construction phase the stresses induced on the waste form were evaluated. Tank wall and floor stress/strain relationships were determined using extensiometers, strain gauges, inclinometers, accelerometers, earth pressure cells, and precision leveling. This is an important part of a barrier emplacement because a miscalculation of the forces exerted on the waste form or structure could result in damage to

the waste form and unplanned releases. No undue forces were observed during the cement layer installation.

Following the installation of the cement layer a novel technique for checking barrier integrity was employed. This method involves the use of gaseous tracers to locate and size breaches and may provide estimates of the barrier diffusivity. The proof-of-concept demonstration of this technique was performed on the cement layer prior to lining the barrier with the polymer. This was done because the likelihood of a crack or gap occurring in the cement layer was greater than in the composite, final barrier. Verification of the polymer/cement composite barrier will be completed in March. A brief description of the technology is given later (see Integrity Verification) and a more complete synopsis can be found in Sullivan et al (4) or Heiser (5).

The primary barrier was to be placed by panel jet-grouting utilizing a two-part polymer grout. The polymer chosen is a thermosetting, high molecular weight acrylic manufactured by 3M Company. 3M provided laboratory formulation and testing to meet the specific needs of the grouting demonstration. They also provided expertise in the field during the construction phase of the demonstration. The resin is polymerized using a catalyst (benzoyl peroxide) in combination with a promoter (N-ethyl, N-hydroxyethyl, m-toluidine). The promoter was mixed in with half the monomer resins (part A) and the catalyst mixed into the second half part B). Two separate tanks held parts A and B and two separate pumps were used to deliver the fluids for grouting. The polymerization reaction begins when parts A and B mix together in the ground. To facilitate jet-grouting of a two part resin the drill rig would require modifications. Off-the-shelf dual wall drill pipe was used so that the two separate grouting media could be simultaneously injected and therefore, the two fluids mixed together in the soil only after leaving the drill stem. The mixing occurred as part of the soil mastication/mixing that occurs from high pressure jetting. Prior to installing the polymer liner the drill rig modifications had to be tested using the polymer grout.

BNL was also involved in a parallel project at Idaho Engineering Laboratories (INEL). This project was to demonstrate in-situ stabilization of buried waste using polymers (see Loomis et al in this proceedings). At this demonstration modifications to the casa grande drill rig were completed. The modifications included the installation and shakedown of a dual wall drill stem, flow metering devices and a low pressure pump. These modifications allow the jet grouting rig to deliver two separate grouts simultaneously. The acrylic resins were injected in two parts. One part contained acrylic resins mixed with the catalyst and the other part contained resins mixed with the promoter. Polymerization of the resins initiates only when the promoter and catalyst are mixed. By delivering the two parts separately the polymerization reaction begins only after the resins leave the drill stem and are mixed in the soil. The system was tested by drilling and injecting a test column in undisturbed soils.

The following day the column was excavated. The column was 16-18 inches in diameter as expected based on results of column grouting using cement in earlier field tests at INEL. The polymer column was more uniform from top to bottom, than the pancake stack seen for cement grout under similar conditions. The column was removed and photographed (Fig. 3). Following the test column, two pits of simulated buried waste were stabilized using the polymer grout with jet grouting. Grouting went smoothly with and both

pits were completed in one day (originally two days were allocated). Results of this field trial were better than expected and proved the equipment would perform properly for the barrier demonstration at Hanford.

Fig. 3

The grouting rig was returned to Hanford for installation of the polymer layer of the close-coupled barrier. The polymer layer was installed via panel jet grouting at a 45 angle, closely following the inner surface of the previously installed cement barrier. The polymer was the same two part system used at the INEL field tests. The resins were delivered to the site by 3M in two lots. Lot A was prepromoted resins (0.3 wt% N-ethyl, N-hydroxyethyl, m-toluidine) and lot B was plain resins. The benzoyl peroxide catalyst (50% active) was added to the lot B drums (16 lbs./55 gal. drum) one to four days prior to use. Each part was transferred to separate holding tanks for delivery to the jetting pumps (see Fig. 4). Flow meters were used to monitor the flow rates of each part thus assuring the proper mix ratio (1:1) of the two parts and that the correct amount of grout was delivered. As a backup to the flow meters the storage tank levels for each part were also monitored.

Fig. 4

The viscosity of the polymer grout resins was near water. This necessitated the lowering of the gel time of the resin mix, additional promoter and catalyst were added to reduce the gel time from 180 minutes to 50-60 minutes. This was done to allow the grout to cure before too much "run-out" occurred. In test panels injected with a 180 minute cure time the resins spread out more than anticipated due to the coarse (near gravel) nature of the sandy soil and the very high porosity (40-45%). A second set of panels injected with a 60 minute gel time resulted in an acceptable column and minimal run-out. For future installations it would be desirable to use a thixotropic agent to increase the viscosity of the resins and thereby reduce or eliminate any run-out. This is a preferable method for two reasons, first the same effect is accomplished but the resin requirements are slightly reduced by the addition of the thixotrope. Second, an more important the gel time could be left at the longer times. This reduces the absolute temperature rise during the polymerization. For a fixed amount of resins the slower the polymerization process takes the lower the temperature. There is a fixed number of watts available from the polymerization reaction and the peak exothermic temperature is dependent on how quickly those watts are released.

Cost

DOE has about 3000 hazardous waste landfills, the representative site is approximately 2 acres. Potential cost savings of close-coupled barrier technology range from a few million to tens of millions per representative 2 acre site.

Subsurface barrier costs vs. Remediation costs for a representative 2 acre site, 20 feet deep

Type of Corrective Action	Estimated Project Cost in $/cubic meter	Approximate Project Cost
Subsurface Barrier Systems Under Development		
Close-coupled Polymer/ Cementitious Barrier**	$24	$1,700,000
Cryogenic Barrier (6)	$90	$6,400,000
Remediation of Organic Compound Contaminated Site		

Thermally Enhanced Vapor Extraction (7)	$150	$10,700,000
Excavate and Treat (6)	$590	$41,900,000

**Project cost data

INTEGRITY VERIFICATION

It is recognized that no suitable method exists for the verification of an emplaced barrier's integrity (5). Because of the large size and deep placement of subsurface barriers detection of leaks is challenging. This becomes magnified if the permissible leakage from the site is low. Detection of small cracks (fractions of an inch) at depths of 100 feet or more has not been possible using existing surface geophysical techniques. Compounding the problem of locating flaws in a barrier is the fact that no placement technology can guarantee the completeness or integrity of the emplaced barrier. In jet grouting the borehole may become misaligned or the jet can be partially obstructed by cobble or varying soil types/densities, leaving a gap in the final barrier. Panel jet grouting may leave gaps between panels and/or at the junctions of horizontal and vertical barrier walls and may be thinner, and thus more prone to cracking. Additionally at the time of gel formation separations or "tears" may occur if localized settling takes place.

As a subtask to the barrier emplacement this project demonstrated a method to verify the continuity of the barrier. Perfluorocarbon tracers (PFT) will be used to locate breaches in the barrier. It is expected that the demonstration will provide a proof-of-concept for gaseous tracer verification of barrier integrity and will give an estimate of the resolution of the technology.

PFT technology consists of the tracers themselves, injection techniques, samplers and analyzers. PFTs have negligible background concentrations of PFTs in the environment, consequently, only small quantities are needed. The tracers are nontoxic, nonreactive, nonflammable, environmentally safe (contains no chlorine), and commercially available. PFT technology is the most sensitive of all non-radioactive tracer technologies and concentrations in the range of 10 parts per quadrillion of air (ppq) can be routinely measured. The PFTs technology is a multi-tracer technology permitting up to six PFTs to be simultaneously deployed, sampled, and analyzed with the same instrumentation. This results in a lower cost and flexibility in experimental design and data interpretation. All six PFTs can be analyzed in 15 minutes on a laboratory based gas chromatograph. Low detection limits allow detection of very small breaches in the barrier. Breaches will be located by injecting a series of PFTs on one side of a barrier wall and monitoring for those tracers on the other side. The injection and monitoring of the PFTs will be accomplished through vadose zone monitoring wells. The amount and type of tracer detected on the monitoring side of the barrier will determine the size and location of a breach. It is easy to see that the larger the opening in a barrier the greater the amount of tracer that transports across the barrier. Locating the breach requires more sophistication in the tracer methodology. Multiple tracer types can be injected at different points along the barrier (both vertical and horizontal). Investigation of the spectra of tracers coming through a breach then gives a location relative to the various tracer injection points.

The initial verification of the cement layer was completed in October 1995. Results were promising (see Sullivan, Gard and Heiser (4) in these proceedings). Verification of the polymer/cement composite barrier will

be completed in January and should provide a method of determining breaches in a barrier and perhaps estimates of the diffusion characteristics of the barrier.

CONCLUSIONS

This project has successfully demonstrated the feasibility of installing a close-coupled, composite barrier. Close-coupled barriers are applicable to final, interim, and emergency loss of confinement conditions. The technology is applicable to any buried or surface waste form that has the potential to release mobile contaminants. Unlike many other subsurface barrier technologies, close-coupled barriers are applicable to a wide range of waste materials and geohydrologic conditions. This is extremely advantageous because nearly every subsurface barrier has site specific conditions that require the flexibility offered by this technology, more specifically this technology offers an ability to place barrier materials that are compatible with virtually any waste form in almost any geologic setting.

End users for this technology include any DOE, state or commercial facility that has buried waste that may release contaminants to the environment at unacceptable levels. Specific end users have been identified and include Idaho National Engineering Laboratory (INEL), Brookhaven National Laboratory (BNL) and the Hanford reservation. INEL and BNL are interested in the full subsurface close-coupled barrier technology. Letters of support of the demonstration have been obtained from Lockheed Idaho for INEL and the DOE area office for BNL. Hanford has expressed interest in the use of polymers to form a close-coupled barrier. This technology could be used to seal leaks in the underground storage tanks at Hanford.

For future installations it would be desirable to use a thixotropic agent to increase the viscosity of the resins and reduce or eliminate any run-out. This reduces the resin requirements and the peak exothermic temperature during the curing of the grout.

Perfluorocarbon tracers will allow locating and sizing of breaches at depth. The technology has regulatory acceptance and is used commercially for non-waste management practices (e.g. detecting leaks in underground power cables). This technology has been used in a variety of soils and locals and will be applicable to the entire DOE complex as well as commercial waste sites. The major use of tracers will be to verify placement continuity of a freshly emplaced barrier and to re-check corrective actions that may be used to seal or repair a breach. It may also be useful to periodically check a barrier to determine the long term integrity of the walls. This would certainly be beneficial if a cementitious grout (portland based) barrier were used. Cementitious grouts are prone to cracking from various degradation modes including wet-dry cycling which is prevalent at many of the DOE sites (e.g. Sandia and Hanford). Tracers would allow determination of performance losses in containment over the life of the barrier.

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30-26

THE SELECTIVE CHELATION OF LEAD IN SOIL WITH A WATER SOLUBLE POLYMER

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ABSTRACT

A proprietary chelating polymer has been synthesized with a high affinity for lead. Batch experiments show the chelator's ability to reduce lead contaminated levels to below EPA requirements in a single washing step. In order to effectively investigate the process feasibility of soil remediation of lead contaminated soil with this polymer, soil transport characteristics must be known. Column experiments using radiolabeled chelator yield information pursuant to vital parameters needed for feasibility studies. Preliminary data suggests that the chelator has a low affinity for quartz.

INTRODUCTION

Lead is listed as a contaminant on roughly a third of the sites on the National Priorities List; in the past, standard protocol for cleaning lead contaminated soil included off-site disposal, solidification and stabilization or capping. These methods of cleanup are not environmentally sound and not permanent solutions. As a result, investigations into treatment alternatives have found chelation technology as a viable choice for lead removal. The advantageous qualities of such an approach include the recovery of lead as a natural resource, the restoration and redeposition of the soil, and the permanence of the solution. Pursuit of this alternative resulted in the development of a proprietary water-soluble polymer, or chelator, with a high affinity for lead by collaborators at Los Alamos National Laboratory (LANL). Batch experiments have shown that the chelator is soluble over a wide pH range, has a high specificity for lead, and is able to remove lead to below EPA requirements. However, before developing the full scale process, investigations of the chelator's transport characteristics through soil must be completed. Column experiments run using radiolabeled

polymer provide information regarding the polymer's transport through soil.

THEORY

The transport of the water soluble chelator through columns will be driven primarily by fluid motion. This condition gives rise to two macrotransport mechanisms 1): convection (also called advection) and dispersion, a diffusive-like process that comes about due to the myriad and tortuous pathways through the column interstices.

Accordingly,conservation of the species will be described by a convective-dispersion equation of the form:

Eq. 1

where jP, the flux of the dissolved polymer, is given by

Eq. 2

and RP is a source/sink term that accounts for processes, such as attachment to soil particles and metal binding, that can alter the local concentration of chelator cP. In Eq. (2), u is the spatially averaged interstitial fluid velocity and DP is a dispersion coefficient. The radiolabeled column transport studies will provide parameters governing adsorption and desorption characteristics of the chelator under the test conditions.

METHODS

Polymer Synthesis

Synthesis is done in small reaction batches of approximately 300 ml. The commercially available base polymer (MW 600 to 750,000) is functionalized to create the desired chelating polymer. Currently, the synthesis process is under patent review. For the column experiments requiring radiolabel, the base polymer is simply functionalized with reactants containing radiolabel. The reaction mixture is then ultrafiltered with a 10,000 molecular weight cut off (MWCO) filter to remove all unused reactant, radiolabel and polymer of low molecular weight, yielding a homogeneous mixture.

Polymer Yield and Quantification

Copper tests are done to quantify the reaction product using a spectrophotometer. Samples and standards are prepared by combining 0.5 ml of 0.1 M copper sulfate ($CuSO_4 \cdot 5 H_2O$), 2 ml Orion pH storage solution and an appropriate amount of sample or chelator stock solution in a 25 ml volumetric flask. The solution is brought up to 25 ml, and a standard calibration curve of absorbance (284 nm) versus polymer concentration is established for chelator concentrations of 0, 10, 50 and 100 ppm. This linear relationship is then used to quantify the chelator concentration of the prepared samples.

Lead concentrations are found by using an inductively coupled plasma unit (ICP) at LANL. After ICP calibration for lead, samples are injected into a chamber where they are exposed to argon plasma at temperatures of 6000 to 10000 K, and atomic emission spectra are analyzed using photo multipliers. This quick and efficient technique has its lowest detection level for lead less than 1 ppm.

MARK Columns

MARK column experiments, in which the bulk of experiments were run, were developed at the University of Arizona (2) and have been used effectively to investigate factors affecting the transport of biocolloids through saturated porous media. The column itself is the bottom barrel of a 3 ml syringe with 0.8 cm ID. The syringes are prepared by inserting a 0.8 cm GF/D filter (Whatman, 2.7 mm nominal pore size) into the bottom of the

syringe barrel. The syringe is then packed with porous media--sand, coated sand or soil. The medium is typically stirred to remove trapped air, ensure homogeneity and provide an even surface at the top of the barrel. The barrel is then rinsed with water of the same ionic composition of the test solution (dependent on experimental objectives) until there is no pH change across the column. The column is now fully prepared.

Chelator transport experiments are initiated by adding a known volume (between 0.4 and 40 ml) of the 0.5% test solution containing radiolabeled polymer at the top of the packed mini-column. The flow conditions are manipulated to ensure laminar flow. The flowrate for all experiments was 4 ml/min. The column is then rinsed with solution of the same ionic strength to remove residual, unattached label from the column pores. The procedure permits tightly controlled application of soluble radiolabeled chelator, semi-continuous measurement of labeled chelator in the column filtrate, and (after sectioning the column) measurement of retained labeled chelator in the column media as a function of bed depth. Radiolabel counts of soil slices and column effluent are made using a Beckman liquid scintillation counter.

As currently configured, the procedure has produced reproducible measurements of radiolabeled microorganisms that are retained on soils or sediments in mm-scale sections of sand columns. In the context of any appropriate transport/reaction model, such data will support parameter estimation, model calibration and prediction for the purpose of large-scale reactor design, pursuit of environmental objectives and design of in-situ remediation procedures.

RESULTS

The chelator was synthesized and tested by collaborators at LANL. Binding capacity tests showed that 10 mg of polymer bound approximately 4 mg of lead. Preliminary soil extraction batch experiments were performed in which 2 g of lead-contaminated Cal-West soil and 8 g of water containing chelator were combined in a sealed tube and agitated for 24 hours. Soil and water were then separated and both were analyzed for lead using ICP. After one extraction, the chelator lowered residual soil lead levels below EPA standards (see Fig. 1), as determined by the Toxicity Characteristic Leaching Procedure (TCLP). Two additional extractions were performed which demonstrated that lead can be removed to a level of less than 1 ppm, even when the initial lead concentration is 6000 ppm. Once the chelator was known to effectively remove lead, the specificity of the chelator for lead was determined. Again, Cal-West contaminated soil was extracted and the concentrations of Ca, Fe, Mn, Mg and Pb were determined both for the liquid phase containing the polymer and the solid or soil phase. As shown in Fig. 2, the chelator is highly specific for lead. The chelator bound 97% of the lead present and below 5% of the Mg, Fe and Mn present. Unfortunately, 9% of the Ca present was also bound by the chelator, however this Ca binding did not appear to affect the ability of the chelator to remove Pb, the target metal. Due to the success of these soil batch experiments, the process for synthesizing this chelator has been submitted for patent review(3).

Fig. 1

Fig. 2

Upon completion of the batch extraction experiments, column experiments were performed to determine the feasibility of using this water-soluble chelator for soil washing applications. These experiments were done in

MARK mini-columns, which measure the fraction of polymer retained in the sediment as a function of bed-depth. The first experiments were done using and ideal soil matrix of quartz sand particles (145 mm average diameter) and 5 pore volumes of chelator solution. No contaminants were present in the bed. As show in Table I, the polymer had a very low affinity for the quartz, less than 1% adhesion under the given flow conditions. Since this polymer is water soluble, this result was anticipated. Examining the concentrations of polymer as a function of depth (Fig. 3) for the 5 pore volume case, it was difficult to determine whether or not the polymer was exhibiting first order adsorption with no desorption. Thus, the volume of the test solution was increased by an order of magnitude to obtain a higher amount of polymer adhesion to the quartz and thus more accurate data.

Fig. 3

When 50 pore volumes of polymer solution were used, approximately the same total amount of polymer bound to the quartz as was observed for the 5 pore volume experiment, see Table I. This suggests that the adsorption is not first order at steady state, instead there may exist preferred binding sites for the polymer and once these sites are saturated, no additional polymer will adhere to the quartz surface. The other possibility is that the flow rate of 4 ml/min is too fast to allow for sufficient binding of the polymer to the surface.

To verify these results the following experiments will be performed. First, and equilibrium isotherm will be generated to determine how much polymer will bind to the quartz surface. This information will verify if the polymer has an extremely low affinity for quartz and if the flow rate was simply too fast to allow for adhesion. Secondly, smaller amounts of polymer solution (0.5 to 1 pore volume) will be used in the MARK experiments to determine if indeed there are preferred binding sites on the quartz surface which are saturated by the polymer. Once these quartz experiments are complete, the water and soil chemistry will be made more complex in a step-wise fashion and further MARK experiments will be performed prior to pilot-scale soil remediation experiments.

Table I

CONCLUSIONS

The water-soluble chelator synthesized at LANL shows tremendous potential for remediation of soils contaminated with lead. The chelator has a high binding capacity for lead; it is highly specific for lead compared to other common metal ions found in soil. Transport experiments performed in MARK columns demonstrate that the chelator has a low affinity for quartz. Further experiments will involve determining the effects of Pb, other metal ions and soil type on chelator transport. The long range goal is to perform an in situ test remediation study in which this chelator will be utilized to was or remove Pb to below EPA requirements.

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ENERGY CONSIDERATIONS AND SOIL TEMPERATURE CHANGES DURING SOIL VAPOR EXTRACTION

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ABSTRACT

Soil vapor extraction (SVE) is a widely used remediation technology for sites contaminated with volatile organic compounds (VOCs). In addition to removing VOCs, SVE can produce secondary effects on soil temperature and moisture content. Soil temperature is important because it controls the vapor pressure of VOCs and the mass transfer rate out of stagnant, low permeability zones in the soil. Soil moisture content affects the removal efficiency of SVE by decreasing the permeability of the soil, by heating or cooling the soil in response to latent heat effects, and by affecting the partitioning of the VOCs. A comprehension of the importance of temperature and moisture content will result in a greater understanding of the energy fluxes associated with soil venting. This is crucial to the development of an optimal design. In this work, a simple model for the enthalpy balance during SVE was used to examine several processes leading to changes in soil temperature and moisture content.

INTRODUCTION

Soil venting or soil vapor extraction (SVE) is frequently used to remediate soils contaminated with volatile organic compounds (VOCs) such as gasoline or chlorinated solvents. The efficiency of a soil venting system is dependent upon the achievable flow rate (1), presence of heterogeneities (2), VOC vapor pressure, and reaction kinetics. Although simple sites with homogeneous materials and uniformly distributed high vapor pressure compounds can be rapidly cleaned in theory, most sites are complex. This leads to initially rapid removal rates followed by a long tail of lower contaminant removal. Cleanup problems at complex sites are the source of a variety of research studies into the enhancement of site remediation.

One potential method for enhancement of soil venting is increasing soil temperatures. Higher temperatures lead to a more rapid diffusional release of VOCs (3) by increasing the vapor pressure. Since the diffusion equation is linear, the diffusional removal rate from stagnant zones is directly proportional to vapor pressure. Henry's Law constants that govern partitioning between aqueous and vapor phases may be important (2). Examination of Henry's Law constants (4) indicates that higher temperatures lead to greater partitioning into the vapor phase for most VOCs. Soil temperature is important when coping with stagnant zones in the soil. Since diffusion rates increase with a rise in temperature, a

higher soil temperature will remove VOCs from stagnant zones faster. Soil temperature also affects bioventing. A rough rule of thumb is that each ten-degree Celsius rise in temperature doubles microbial activity. Thus, moderate elevation of the soil temperature could result in faster biodegradation of contaminants (5). In summary, moderate increases in soil temperature can facilitate cleanup in several ways: raising vapor pressure of residual VOCs, speeding up diffusional removal from stagnant zones, favoring partition from soil moisture into soil air, and promoting microbial activity.

Because of the relationship between soil temperature and cleanup efficiency in soil venting and bioventing, a number of methods have been developed to raise soil temperature. The US Environmental Protection Agency (EPA) has funded studies on methods including injection of heated air, steam injection, and radio frequency heating of the soil (6). Most of these methods have the disadvantage of being energy and capital intensive. The purpose of this contribution is to examine some of the energy relationships leading to temperature changes in the soil during venting and to explore the feasibility of low cost soil heating. Some of the factors in soil venting influencing soil temperature are discussed, followed by the application of a simple mathematical model to a hypothetical contaminated site located in Lexington, Kentucky.

The energy terms important to SVE are: sensible heat, latent heat of vaporization of water and contaminants, vacuum related evaporation, pumping related energy, and biodegradation. As air flows from the atmosphere to the soil, it is heated or cooled to soil temperature. Simultaneously, the moisture in the air equilibrates with the soil moisture and the temperature of the water vapor changes to that of the soil.

Air moves through a soil venting system from high to low pressure, expanding in the process. Lower pressure causes an increase in volumetric flow rate of air and, consequently, an increase in the mass of water vapor exiting the system. The water vapor flux increases inversely with pressure because the partial pressure of water vapor is set by the soil temperature. Vacuum evaporation causes drying and cooling of the soil, especially if the total pressure in the exit well is low. Any volatilization of contaminants would further cool the system.

Pumping of air through the system may lead to energy changes depending upon the method of pumping. Flow of air through porous media in response to a pressure drop is analogous to the Joule/Thompson expansion of classical thermodynamics - an isenthalpic process (7). For an ideal gas, there is no net energy change in passing through the soil or well screen.

In areas of high microbial activity, biodegradation has been reported (8) to lead to increases in soil temperature on the order of 0.028 to 0.065C/day. This effect was found to be more likely in low flow rate regions rather than in high flow rate areas.

MATHEMATICAL MODEL

A simple mathematical model was used to examine a portion of the energy relationships at a hypothetical field site located near Lexington, Kentucky. The mathematical model is limited to the processes of interest. Biodegradation and latent heat effects from contaminant volatilization are excluded from the simulations in order to focus on the latent and sensible heat changes from air and water vapor. Both biodegradation and contaminant volatilization are site and contaminant specific and, therefore, are important in select situations. The model considers a

single control volume (i.e., spatial variations in temperature in the soil are ignored) with one dimensional flow. The simple model gives less detailed information than a full finite difference code. The primary advantage of the model is that a specific geometry for the system does not have to be defined; only the flow rate per unit volume, input pressure, input moisture content, exit pressure, initial conditions in the soil, and heat capacity of the soil are utilized.

Many system geometries could approximate the model's input assumptions. The system could consist of: a) a network of injection and extraction wells in low permeability soils, b) a single extraction well with multiple injection wells in high permeability soils, c) linear air flow between trenches, and/or d) flow between horizontal wells and the surface. The common factor of these geometries is that the simulations assume that the properties of the input air are either known or controlled and that the flow is one dimensional.

The goal for a SVE system, attempting to promote soil heating, is to minimize heat flux out of the system while allowing sufficient time for heat conduction to warm low permeability zones in the soil that are normally resistant to remediation by venting. Heating of low permeability zones should enhance the rate of diffusion of contaminants out to the high permeability regions for removal.

The equation for temperature change of the soil is:

Eq. 1

E = energy imparted by the latent and sensible heats (kJ/s)

V = volume of soil taken as 1 m3

rCp = density of soil times heat capacity of soil, 2.3106 J/m3/K (Jury et al, 1991)

T = soil temperature, C

t = time, s

All simulations assume one atmosphere (1.013 105 Pa) for input air and 0.9 atm in the extraction well. Integration is performed in a series of hourly steps based upon the meteorological data (temperature and humidity) from the National Weather Service, modified as stated in each simulation. Enthalpy and vapor pressure along with flow rates and pressure drops give the net energy flux, (E), into or out of the system. The "lumped" model accurately represents the system over long time periods but cannot consider spatially variable temperature changes.

It is useful to examine the energy dynamics of soil venting using humidified input air prior to viewing the full simulation. The total energy imparted to the soil per cubic meter of air (at soil temperature and one atmosphere pressure) flowing through the system is shown in Fig. 1 as a function of temperature and relative humidity of the injected air.

Fig. 1

The soil temperature in Fig. 1 is assumed to be 15 C. For input air at 100% relative humidity (i.e., humidified input air), the energy balance becomes positive when the input air temperature is slightly above the soil temperature. The slight negative energy flux at 15 C and 100% relative humidity (~ -3 kJ/m3) is caused by vacuum evaporation. Input air at 50% and 0% relative humidity is less energetically favorable and only provides a net heating of the soil system at temperatures significantly above the soil temperature. For this reason, the injection of warm dry air is ineffective in warming the subsurface. Figure 2 gives the number of soil volumes of humidified air (at atmospheric pressure and soil temperature) required to change the soil temperature one degree Celsius.

Fig. 2
The results are given for three different values of soil temperature (5, 15, and 25C) as a function of input air temperature. Figure 2 illustrates that the temperature of the input air is generally more important than pumping rates in leading to warming of the soil. As the temperature of the input air increases, warming becomes much more rapid (i.e., requires fewer soil volumes of air) and nearly independent of the soil temperature. The sharp dependence on input air temperature is caused by nonlinearity in the saturation vapor pressure curve for water. Warm air holds much more water vapor, and thus thermal energy, than cool air. Significant warming rates (i.e., greater than 1C/50 soil volumes of pumped air) occur when the humidified input air is only slightly (15-20) warmer than soil temperatures. The small degree of temperature difference required for soil warming can be provided by several energy sources including: diurnal atmospheric temperature variations, seasonal atmospheric temperature variations, solar heating, and waste heat from industrial processes. Solar heating, for example, is very efficient for low temperature rises. The second figure can be used to compare expected soil warming rates by humidified air with other processes such as heating from biodegradation and cooling by evaporation of contaminants.
SIMULATIONS
Four simulations were performed for a site near Lexington, Kentucky for a period of one year. All the simulations assume a pumping rate of five soil volumes per day at atmospheric pressure and soil temperature. Initial soil temperature is taken as annual average air temperature and initial volumetric moisture content as 0.1 m3 of water/m3 of soil. Figure 3 shows the soil temperature as a function of time for the four simulations. The ambient temperature data is shown in gray. The temperature curve labeled "a" represents ordinary venting of the soil and serves as the reference (control) case for comparison. All of the alternative venting scenarios result in higher temperatures than the control. In simulation "b", the ambient air is heated 40C prior to being pumped through the system. Curve "c" shows the result of heating the input air 40C and saturating it at this elevated temperature. The "d" curve represents an input that is heated 20C, saturated, and heated 20C more.
Fig. 3
Figure 4 gives the predicted soil volumetric moisture content. The control case (ambient conditions) causes a net drying of the soil, primarily from vacuum expansion related drying whereas the humidification cases result in a slight increase in moisture content. Soil drying for normal venting in the simulation is minimized by venting during the warmer months where the normally humid summer air in Lexington condenses in the cooler soil. Venting in the fall and/or winter months, particularly when air temperatures fall below soil temperatures, results in more rapid soil cooling and drying. Changes in soil moisture content track very closely with soil temperature changes, indicating the importance of latent heat of vaporization. Although not shown, other techniques such as pumping only during the hottest portion of the day and simple heating of the air without humidification have been tried. They are of limited effectiveness because the latent heat dominates the energy balance in most cases (Fig. 1).
Fig. 4
DISCUSSION

One of the most important observations is that humidification of the input air does not significantly saturate the soil with water. If the soil were saturated, air permeability would go to zero and the methods proposed would not be effective. The small change in volumetric water content is related to the thermodynamic properties of water. The high value of the latent heat of vaporization of water causes a rapid temperature rise in the soil with only a small volume of condensation water. The change in volumetric water content per degree temperature rise in the soil is given by:

Eq. 2

(hwv - hwl) = latent heat of vaporization of water (~2.4106 J/kg)

rw = density of water (1000 kg/m3)

This equates to a ~10-3 increase in soil volumetric moisture content per degree increase in temperature. Increasing soil temperatures by 10C, for example, only increases the volumetric moisture content by 0.01, an insignificant amount. As soil temperature increases, the vapor pressure of contaminants in the soil increases, resulting in more rapid removal of moisture from the system. For this reason, change in water content of the soil by humidification is self limiting as moisture flux out of the system rises to match input moisture flux. An additional factor is that normal venting tends to dry the soil because of vacuum expansion.

As illustrated, raising the temperature and relative humidity of the input air is an efficient way of heating the soil. Alternative methods of heating and saturation are widely available. Low grade waste heat and low quality waste steam are available at many industrial plants. Solar energy is also available.

For example, one could use the calculations illustrated herein with a smart solar powered remediation system that would inject fresh humidified air when energetically favorable (e.g., when the atmosphere is warm and/or the sun is shining) and recycle scrubbed air from the subsurface during periods when injection of fresh air would result in soil cooling. Warming the air to higher temperatures prior to injection results in greater rates of soil warming than shown in the simulations (Fig. 2), even at very low flow rates.

By significantly warming the soil, relative to normal venting, the enhancement methods suggested can potentially lower cleanup times. The improvement is anticipated to be particularly significant during the long tail period of cleanup dominated by diffusion from stagnant zones, slow kinetics, and other non-ideal conditions that generally control clean-up times. Even when enhancement is not applied, scientists and engineers should benefit from better understanding of SVE caused changes in subsurface temperature.

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30-30

REMEDIATION OF Hg CONTAMINATED SOIL

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ABSTRACT

The position of mercury contamination was determined and content of mercury in solid and pore waters was estimated on the artificial range in Sergiev Posad brunch of Mos NPO "Radon". The range contain 82.6 g of mercury in solid and 172.3 mg of mercury in pore water. The desorption of the mercury by the H+ ions in laboratory experiments was studied. Special laboratory research with electrokinetic cell demonstrated, that mercury in the hole water have cation form.

Electroosmosis and electromigration provide transport desorbed ions between electrodes. The construction of metal collector with cationexchangable membranes gave possibility accumulate up to concentration 1 mol/l. The metal collector was tested during 1 month (current 1 A, potential ~ 30 V). In natural condition was collected ~ 100 mg mercury. Desorption solution must be introduce in order to receive practical results.

INTRODUCTION

The mercury belongs to the group of heavy metals (Hg, Pb, Cd, As, Zn), distinguished by high anthropogenic background. These metals are very toxic and suppress the life of animals and plants (1).

Anthropogenic mercury discharged into environment comes from several global sources: fossil fuel combustion, chloralkali factories, mining industries.

Mercury and its compounds are actively used in an electro-technical industry, pharmacology, medicine, vacuum technology, scientific researches. Therefore the territories around large industrial centers are heavily polluted by mercury.

Electrokinetic soil processing uses low-level direct current (on the order of ampers per square meter of area) is perhaps one of the most promising in situ decontamination processes capable of removing heavy

metals and radionuclides from soil. The significance of the technology appears to be in its projected low operation cost and its potential applicability to a wide range of contamination situation (2).

The procedure of the electrokinetic decontamination is realized in three stages: 1) transformation of a polluting impurity in the soluble forms (anions, cations or neutral atoms), 2)moving of an impurity by electrical current in the anode-cathode space, 3) accumulation of polluting impurity in the collector of metals.

In the present research problems, arising from practical realization of these stages, are considered.

OBJECT OF RESEARCHES AND Hg ANALYSIS

The improvement of Hg decontamination technique was carried out in Sergiev Posad branch Mos NPO "Radon", located in 90 km north-east from Moscow.

An artificial hole (the size 7 x 7 x 1,2 m) has been dug out in the moraine clays and filled up by sand. Physical properties of clay and sand are listed in Table I.

Table I

Very low filtration coefficient of hard moraine clay limits washing by atmospheric precipitation and underground water flows. The granulometric composition of sand and clay are shown in Fig. 1.

Fig. 1

The sand consists of Al_2O_3 and SiO_2 (~80%), oxides of iron (~6,%), carbonates (~4%), oxides Na and K (~3%). Acided by HNO_3 up to pH 2-3 a solution of nitric mercury in natural water has been entered in the hole from 6 m3 capacity through a punched steel pipe, laid in the central part of the hole at the distance of 20 cm from the bottom in October 1992 (Fig.2).

Fig. 2

In result of natural processes (washing by atmospheric precipitation, change of forms, diffusion, sorption) stationary distribution of Hg around the steel pipe at the distance of no more than 40 cm and restricted by the hole bottom has been formed (Fig.2).

The highest concentration has been reached directly near the pipe. The typical character of vertical distribution of concentration of dissolved and suspended mercury is the same, it means that distribution of dissolved mercury is controlled by exchange reactions, but diffusion processes certainly promote mercury redistribution.

The quantitative chemical analysis for the mercury was performed by atomic absorption spectrophotometric measurement.

Mercury was extracted by tioether from 0.5 l of pore water, after filtration tioether was diluted by nitric acid, limit of detection is 0.0005 mg/ml. Mercury from 2 g of solid samples was extracted by treatment of the sample by nitric acid, limit detection is 0.2 mg/g .

In order to reveal position of Hg contaminant content of mercury in 112 liquid and 112 solid samples from 28 position and 4 different depth (60,80, 100 and 110 cm) was analyzed.

Simple calculations show that hole contains 82.6 g of mercury in solid state and 172.3 mg of mercury in liquid state, average value of distribution coefficient (Kd) is ~300 ml/g .

PROCESS OF DESORPTION

The adsorption of Hg on the surface of the solid can be described by the exchange reaction of two species (Hg and X) between their aqueous solution (aq) and solid (s) (3).

Eq. 1
The rate of change of the concentration of Hg is:
Eq. 2
where Hg(aq), Hg(s), X(aq), X(s) - concentration of species dissolved or suspended in aqueous solution (moles/ml);
K1 and K-1 forward and backward reaction rate constants respectively.
Assuming a first order reaction kinetics for a steady state we have:
Eq. 3
Kap - apparent equilibrium constant.
For real systems we must use the thermodynamic activities instead of the concentration, using the conventional relationship between activity, concentration and activity coefficient.
The activity of the dissolved species is a(aq) =C(aq)g , where g is the molal activity coefficient, C(aq) is molal concentration. Activities of the adsorbed specie is a(s) = c(s)l , where c(s) is mole fraction, l-is the activity coefficient in the adsorbed state. The activity coefficients of ions in solution g can be estimated from Debye-Huckel (4) and related equations, but for the activity coefficients of adsorbed species l there is no comparable method and the equilibrium constants Kap are known only for several specific systems (4).
Coefficient Kd is usually used in practice
Eq. 4
Kd - distribution coefficient at steady state [ml/g]
Cp - concentration of solid in aqueous solution [g/ml].
Kd (opposite from Kap) is not on equilibrium constant and can vary as pH, solution composition, physical and chemical properties of suspended particles change.
It is common practice to express the desorption results of metals in terms of the "% desorbed". Experimental measurements are certainly the most reliable way to find out the type of exchange or desorption that takes place in the given solution-solid system under the given conditions.
The desorption of mercury in our laboratory experiments was studied by the H+-ions. We used contaminated soil from the hole; the concentration of mercury in the soil is 4.1 mg/kg. Known amounts of contaminated soil (usually 2 g) were added to 50 ml of hydrochloric acid solution from 0.01 N to 10 N. The desorption was studied both in hydrochloric acid solution and in distilled water and natural water from the hole. Ca2+ (79%), Mg2+ (14%), (Na++K+) (7%) are basic cations of natural water; HCO3-(84%), SO42-(14%), Cl- (2%) are basic anions. Total mineralization is 230 mg/l. The solution with contaminated soil was mixed constantly during 4 hours. After this solution was filtered and concentration of mercury was measured on the filter and in the solution. The results are presented in term h (Table II)
Eq. 5
C(aq) mercury content in filtrate,
Co(s) mercury content in solid before experiment.
Table II
Actually we may transfer considerable part of mercury in solution only after treatment by 1 N solution of hydrochloric acid.
FORMS OF Hg
Special laboratory research to detect mercury form was realized. Plastic 15 l volume electrokinetic cell was filled with natural water from the hole (Fig. 3). 0,3 mg of mercury (HgNO3) and nitric before pH=3 were

added. Solution was mixed carefully for several days. Stainless steel anode and the cathode collector of metals with 2 cationexchangable membranes were place in the cell. During 24 hours for I=0.7 A collector of metals collected 75% of mercury. Mercury in the hole water had cation form.

Fig. 3

TRANSPORT

Mass balance across the system and the assumption of constant flow rate for the bulk fluid (without adsorption, desorption, chemical reaction) describes the movement of different species of medium by the Eq. (5)

Eq. 6

C - mass of solute per unit volume of solution,
z - charge on the ion,
F - Faraday`s constant (96485 coulombs),
R - universal gas constant,
T - temperature (oK),
Y - electrical potential,
D - diffusion coefficient,
x - flow direction,
V - average seepage velocity.

The first, second and third terms on the right-hand side represent the diffusion advection and migration components respectively.

Advection may be used for decontamination, but if the velocity of advection is too large, this process demands very much quantity of desorbent. By advection ions and neutral atoms may be removed.

Electrokinetic is the movement of water (electroosmosis), ions and polar molecules (electromigration) between two electrodes under the action of an applied direct current electric field.

By electroosmosis we may move ions and neutral atoms. In electroosmosis, the flow rate qos [cm3/s] conventionally related to current I (A) by empirical relationship:

Eq. 7

kI - coefficient of water transport efficiency [cm3/A s].

Practically kI change within the limits 10-4 - 10-5 [cm3/A s]. It means that for I = 1 A qos change in interval 0,1 - 1 cm3/s and electroosmosis may have very important role in the decontaminated processes.

Electroosmotic flow could be induced and sustained in all soil types. Electroosmotic flushing is dependent mainly on soil mineralogy.

The flow rate of electromigration qel is:

Eq. 8

Grady is the fundamental parameter, which determine quantity of flow.

Current (I) is sum of separate ions currents

Eq. 9

li - molar conductivity of separate ion.

Cations of Mg2+ and Ca2+ is the basic for the natural hole water and just they determine the molar cations conductivity. For solution of electrolytes l may be estimated by using Nernst-Einstein equation:

Eq.10

where D for some ions may be found in (4); for most of ions (except H+ and OH-) l changes in small intervals (40-80 . 10-4 W-1 m2 mol-1), lH+ =349 . 10-4 W-1 m2 mol-1, lOH-= 197,6 10-4 W-1 m2 mol-1

The maximum meaning of current is dependent on construction of electrodes and ordinary it is no more then 1-2 A.

Steady state of current may be disturb by 2 phenomenons.

1. The primary electrode reactions are:

$2H_2O - 4e^- = O_2 + 4H^+$ (anode)

$4H_2O + 4e^- = 2H_2 - 4OH^-$ (cathode)

Subsequently the hydronium ions produced at the anode migrate toward the cathode and the hydroxide ions produced at the cathode migrate toward the anode. Acid front migration is beneficial for metals extraction from soil. The pH conditions varied at any time during the treatment and can have a significant influence on the local conductance. pH = 12 near cathode region can limit the movement of the metals.

2. In due time the electromigration lead to accumulation of the different sign ions near electrodes and, as consequence, the diffusion act opposite electromigration. It is obvious, that electromigration shell be equal to diffusion , then process of decontamination shall be finish.

In order to overcome this two undesired phenomenons special construction of metal collector was used.

THE COLLECTOR OF METAL AND PRELIMINARY EXPERIMENTS

Schematic construction and hole position of the cathode metal collector is given in Fig. 4. Clear plastic was used for all collector parts. Cationexchangable membranes are placed at each end. Electrode chamber is 2.7 l in volume. Electrode with surface area of 70 cm^2 was constructed of titanic pipe with a 1cm diameter. In all preliminary experiments a constant current 1 A was applied across the electrodes, potential (~ 30 V) was changed in due time. Cationexchangable membranes fixed cations in the collector volume, concentration limit of membrane is 1 mol/l, it means, that solution in collector (natural water from hole) must be change within 3 days.

Fig. 4

In order to neutralize generation of OH^- ions, hydrochloric acid was added in the collector in accordance with rate of primary electrode reaction.

The desorption solution on the first step research is not used and practical possibilities of metal collector was studied. The system worked successfully within one month from 17 August to 15 September 1995. Mainly ions of Ca^{2+} and Mg^{2+} was accumulated in collector of metals. Results of 6 cycles are presented in Table III.

Table III

At an average 1 amp-hr may collect 0.62+0.08 g of cations in collector. The mercury is not revealed in the collector solutions in the limit of detection, but mercury (~ 100 mg) in the end experiment in cationexchangable membranes was fixed. Very low migration quality of mercury was caused by content of natural water in hole (high content of bicarbonates). Experiments with introduction desorbed solution in contaminated soil shall be realized next summer.

CONCLUSION

1. The position of mercury contamination and mercury content in solid and pore water on the artificial range in Sergiev Posad brunch Mos NPO "Radon" were estimated.
2. The desorption of mercury carried out in laboratory by H^+ ions was studied.
3. Mercury in the natural waters of range had mainly cation form.
4. The cathode metall collector was constructed and tested on the range.
5. Desorption solution must be introduced to receive practical results.

ACKNOWLEDGMENTS

The author wish to recognize the contribution made to the project by the following individuals:
A.V.Timofeeva and A.A.Kostrova - analytical chemists,
V.N.Titov - the help in the work,
Y.G.Tkachenko - the useful discussion,
Y.G.Beltyukova - for help in preparation of article

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ISOTOPIC ANALYSIS OF URANIUM IN CONTAMINATED SOILS BY GAMMA RAY SPECTROSCOPY

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ABSTRACT

The total uranium radioactivity concentration (Bq/g) of soils contaminated with enriched uranium can be determined by gamma ray spectroscopy, as opposed to alpha spectroscopy. The gamma spectroscopy method saves considerable time and reduces expenses associated with site characterization and remedial activities. Uranium 234 is the dominant contributor to the radioactivity of enriched uranium (e.g. the U-234 represents greater than 90% of the total uranium radioactivity for uranium at 6 weight percent U-235). Since U-234 is an alpha emitter with no practical gamma yield, alpha spectroscopy is the usual method of choice for the analysis and characterization of soils contaminated with enriched uranium. However, gamma spectroscopy can be used to determine the U-234 concentration (or the total uranium concentration) by using the fact that the U-234/U-235 ratio is nearly invariant for a particular uranium contaminant source. This fact, coupled with accurate U-235 and U-238 measurements by gamma ray spectroscopy, provides a method that meets the site specific remedial investigation sensitivity, accuracy and sample analysis turn-around time requirements. The gamma spectroscopy method is nondestructive, fast, and reliable. A sample turn-around time of 10 minutes can be achieved for a U-234 minimum detectable concentration (MDC) of 0.12 Bq/g (3.4 pCi/g) at a 95% confidence level. A range of uranium contamination levels and contaminant enrichments were encountered in this study: uranium concentrations of 0.03 to 20 Bq/g (1 to 540 pCi/g), and enrichment levels to 6 weight percent U-235. The accuracy and success of the method is dependent on the use of uranium calibration standards (as opposed to multinuclide gamma standards), and applying all appropriate gamma emission interference corrections. The methodology was

validated by comparing 45 soil gamma spectroscopy analysis results with the alpha spectroscopy results performed on the same set of samples.

INTRODUCTION

The need to provide a fast turn-around method for the analysis of uranium radioactivity in soil contaminated with enriched uranium prompted the selection of high resolution gamma ray spectroscopy. Gamma spectroscopy can be used to determine the U-234 concentration (or the total uranium radioactivity concentration) by using the fact that the U-234/U-235 ratio is nearly invariant for a particular uranium contaminant source. The method is nondestructive, sensitive, fast and has been adapted for field/mobile laboratory use. It provides a near real time analytical method to support on-site decisions/actions relative to characterization and the control of remedial activities. The analysis of 450 samples (supporting a recent site characterization) demonstrated the value of the method in saving time and characterization costs. Projected savings amounted to fifty percent in analysis costs compared to alpha spectroscopy, and considerable savings in field sampling and excavation time because a potential need for a future mobilization was eliminated by having analytical results immediately available.

The analysis of uranium in environmental samples by gamma ray spectroscopy is complicated because it requires the use of a number of low energy gamma emission lines, the correction for the U-235/Ra-226 interference at 186 keV, and the correction of a number of other interferences. These complications are a principal reason the method is not widely used for uranium isotopic characterization in environmental soil/material analyses. However, the isotopic characterization of enriched uranium materials (particularly as it relates to control of fissile materials) has been extensively studied and applied (1).

The gamma spectroscopy method can provide analytical information on many radionuclides, in addition to uranium. This broad analytical capability is a valuable asset since it avoids duplication of radioanalytical capabilities. This gamma ray spectroscopy application is the focus of this report because of its importance to characterization and remedial activities. It is fast and reliable, and can save considerable time and expense associated with these activities.

METHODOLOGY AND PROCEDURES

Sample preparation is simple, does not involve chemical processing, and is accomplished by placing (in the field or laboratory) the selected soil sample into a standard geometry counting container. The nominal 500 cm3 Marinelli container was used in this study; the soil sample was mixed prior to being placed in the counting container to ensure a degree of homogeneity. The container was filled completely with soil; about 700 to 1000 grams is typical and depends on soil/material porosity and water content. The analysis of a large sample is desirable to ensure a representative result from the sample location. This sample size is in contrast to chemical or alpha spectrometric methods that utilize about 1 gram, or at most 10 grams, for an analysis. Therefore, success of these later methods depends on careful homogenizing and sampling of the initial sample material.

The mobile high resolution gamma spectrometer system includes a Canberra thin window intrinsic germanium detector (with an energy resolution of 2.1 keV and an efficiency of 20% relative to NaI), a Canberra graded four inch lead shield, an Aptec multichannel analyzer/computer interface card, Aptec preamplifier and high voltage hardware, and Aptec gamma

spectroscopy software for spectroscopic data reduction and quantitation (2,3). The system and other supporting equipment is housed in a mobile trailer provided with atmospheric control for all weather operation. All the gamma spectroscopic measurements reported in this work were performed with this system.

The gamma efficiency calibration of the system was performed using a simulated soil matrix containing a well characterized natural uranium ore standard. The calibration standards were prepared by mixing a weighed quantity of a finely ground uranium ore standard(4) with a simulated soil matrix material; the mixture was thoroughly mixed to ensure uniformity, and then quickly solidified. The uranium ore was of an age where all the U-238 progeny are at, or very near, secular equilibrium. This provides a standard containing all the principal radionuclides of interest (U-238, U-234, U-235, Ra-226, and their progeny). By calibrating the system with the exact radionuclides and gamma energy lines used for the sample analysis, one eliminates quantitation errors related to decay scheme/efficiency complications, and errors associated with the nuclide gamma yields and the fitted/calculated efficiency vs. energy curve for the gamma detector system. The fact that these uncertainties are eliminated improve the accuracy and precision of the measurement, and are key to the overall success of this method.

The uranium (U-234, U-235, and U-238) isotopic radioactivity is derived from measurements of the U-238 and U-235 content of the soil or environmental material. The U-238 concentration is obtained from Th-234 (and at higher concentrations Pa-234m). These progeny are in secular equilibrium with U-238 and can be used as a measure of U-238 for uranium materials that were processed/separated more than 200 days earlier. This is the usual situation encountered for old or inactive sites where characterization and remedial actions are desired. The U-235 concentration is obtained from the several U-235 gamma emission lines; the principal 186 keV line of U-235 requires correction for interference from Ra-226; the radium contribution amounts to about 50% of the U-235 peak area in natural uranium sample with all progeny in equilibrium. The U-234 concentration is derived from the measured U-235 concentration and the nominal U234/U235 activity ratio for the project site. Table I provides nominal U-234/U-235 radioactivity ratios for various uranium-235 enrichments(5,6). The enrichment process (gaseous diffusion, electromagnetic, centrifuge or laser methods) determines the nominal ratio, and variations can be expected. Since gaseous diffusion is the source of most of the enriched and depleted uranium, these nominal ratios reflect this dominant source. The U234/U235 activity ratio is not a strong function of uranium 235 enrichment. A U234/U235 ratio of 22 to 25 is typical for most situations where low enrichment uranium is involved. The U-234/U235 ratio for a site can be obtained from a knowledge of (a) the U-234 and U-235 isotopic compositions of the uranium materials processed or handled at the facility, (b) an isotopic characterization of contaminated materials (alpha or mass spectrometric), or from the gamma spectrometric measurement of U-235/U-238 ratio and a nominal U-234/U-235 value from Table I. The best approach is to characterize the isotopic composition of the contaminated media by one of the direct methods. This provides the needed U-234/U-235 ratio, and also provides an independent verification of the gamma spectroscopy results. This direct measurement (e.g. alpha spectroscopy) would be performed on a limited number of samples, and would be part of a QA/QC data validation protocol.

Table I

The key interference corrections that are necessary to ensure the accuracy and reliability of the uranium isotopic radioactivity measurements involve the 186 keV and 92 keV gamma emission lines. The 186 keV spectral peak is a combination of U-235 and Ra-226; lesser contributors to this peak are Th-237 and Pa-234. For most situations, these lessor contributors can be ignored. The correction for the Ra-226 contribution is derived from the measured Pb-214 and Bi-214 concentrations. The Pb-214 and Bi-214 are progeny of Ra-226; when secular equilibrium conditions exist, their individual concentrations equal the Ra-226 concentration (Bq/g). For the typical situation where processed uranium is the principal radiological contaminant, the radium source is natural occurring radium in the soil. Soil samples placed directly in the counting container and sealed in the field are very near equilibrium. No significant loss of Rn-222 (3.82 day half life) occurs in this situation; the majority of Rn-222 is trapped in the naturally occurring uranium mineral grains or soil matrix constituents. Also, the sample material is not significantly disturbed in the preparation step, compared to drying and grinding of the sample. For situations involving radium contamination from a uranium processing, or radium recovery operation, or some form of aggressive sample preparation, one needs to establish the state of secular equilibrium for the immediate progeny of Ra-226 in the sample. This is easily done by monitoring the Pb/Bi-214 as a function of time for the particular site soil/contaminant situation. The delay time required between sample preparation and analysis can then be established. No delay was found to be necessary for the soil/contaminant condition at the site that was the subject of this study. If the equilibrium with Ra-226 progeny is not attained, the Ra-226 correction is underestimated, and as a result, the interference corrected U-235 value is biased high. Therefore, the U-235, U-234 and total uranium radioactivity values will all be biased high, which provides a degree of conservatism that avoids the situation of underestimating the contaminant radioactivity concentration.

The 93 keV spectral peak is a complex peak involving two lines of Th-234, and contributions from Ac-228, U-235, and Th-231. The Th-234 gamma emissions provide a measure of the U-238 in the sample. The Ac-228 is a major correction when Th-232 is present in the soil naturally or as a contaminant. The interference correction for Ac-228 is determined from the measured Ac-228 concentration (using other prominent Ac-228 spectral peaks) and other Th-232 progeny (Pb-212 and Tl-208), if secular equilibrium is established. The U-235 and Th-231 contribution at 93 keV is obtained from the measured U-235 concentration; Th-231 is a short lived immediate progeny of U-235, and is a minor contributor to the complex peak. The actual interference contribution is derived from the contributor concentrations (described above) and their respective gamma yields, together with the gamma yield of the subject line and it's uncorrected concentration (Bq/g). These data provide the necessary information to obtain the final interference corrected result. The 63 keV emission line of Th-234 is also used for quantitation of U-238 when the U-238 concentration is significantly above the minimum detectable concentration (MDC).

The detection sensitivities for the uranium isotopes or progeny permit low level radionuclide concentrations to be measured. The specific situation depends on the general and normal uranium background spectra

for the materials/soils under investigation, and the count interval for the specific sample. For a typical screening analysis situation (about 3 samples per hour, or one sample every 16 minutes), the uranium radioactivity minimum detectable concentrations (MDC) are given in Table II. The MDC is provided at the 95% confidence level and is calculated from the spectral data using standard methodology developed by Currie (7).

The detection sensitivity and MDC of a measurement is strongly influenced by the method background and interferences, and refers to the statistically determined quantity of radioactivity that can be measured at a selected confidence level. Detection sensitivity at the 95% confidence level is defined as that level above which there is less than a 5% probability that radioactivity will be reported present when it is really absent (, or Type I error), or reported absent when it really is present (, or Type II error). Minimum detectable activity (MDA), or minimum detectable concentration (MDC), is common terminology used when estimating the minimum activity or concentration level which can practicably and reliably be measured under the defined sample and analytical conditions. The MDA and MDC are not the detection levels of the method, but values greater than the detection level (7). The derivation of the MDC is based on the Type II error of escaping detection when activity is present. The MDC values provided in Table II represent the average of individual MDC's determined from the net count/statistical analysis of the spectral data gathered for individual soil samples. The method detection level and the MDC can be lowered with longer count periods; or one can achieve a greater sample throughput with a shorter count period, assuming the MDC for the short count is consistent with characterization or remedial objectives. For some remedial activities where a uranium action guideline of 30 pCi/g is used, one can set the analytical conditions such that the U-234 MDC is 0.13 Bq/g (4 pCi/g), and perform a screening analysis every 3.5 minutes using this method. See Table II.

Table II

RESULTS

The U-238, U-235 and U-234 radioactivity concentrations were determined by the gamma spectroscopy method described above for 450 soil samples. These samples were taken as part of a site assessment/characterization investigation, and involved surface and subsurface soil sampling. Uranium materials processing (involving enriched, normal and depleted uranium) was the principal source of the radiological contamination at the site. the capability to provide analytical results within one to two days was required; timing was dictated by the specific need to support control and direction of the site investigation. This particular timing could not be met using the alpha spectroscopy method, but could be met by this gamma ray spectroscopy method, and also, without having to move the mobile gamma spectroscopy laboratory to the site of the investigation.

The uranium isotopic results were validated by alpha spectrometric analysis; 45 samples were selected from the group of 450 samples previously analyzed by the gamma spectroscopy method. The validation samples were sent to an independent laboratory for uranium isotopic analysis by alpha spectroscopy. The laboratory removed a representative aliquot of soil from the sample container (about 1 gram). No drying or homogenizing of the sample was performed prior to selecting the sample for chemical processing and analysis. This introduces an added source of

variability to the alpha spectroscopy results compared to the gamma spectroscopy results. The gamma spectroscopy method is averaging the gamma emissions over about a 1000 gram sample; this provides a more representative result for the total sample volume.

The 45 samples ranged in total uranium content from 0.03 to 20 Bq/g (0.8 to 540 pCi/g). The samples also represented a wide range of U-234 content because of the enriched uranium contaminant source. The observed U-234/U-238 radioactivity ratios varied by a factor greater than 10 (from 0.7 to 8). The paired results for the 45 samples provided a basis for comparison of the two methods. A U-234/U-235 radioactivity ratio of 22 was used to calculate the U-234 content from the measured U-235; this ratio is site specific and based on an evaluation of the alpha spectrometric results for this site. The alpha spectrometric results averaged 19.3 with a standard deviation of 2.9. The paired comparison of results for the two methods is given in Fig 1. The data has been plotted without including the method error estimate of each measurement pair to clarify presentation of the information. In general, the 95% confidence level counting statistics error (the dominant contributor) for the gamma spectroscopy method, and the particular analytical conditions, ranges from 15% at high concentrations to 35% at low concentrations. For the alpha spectroscopy results, the range is 15% to 50%. The overall agreement between methods for U-238, U-235 and U-234 is good considering the large differences in sample sizes, and the fact that no additional sample homogenization was performed on the sample for alpha spectrometric analysis.

Fig. 1

Fig. 2

In addition, the U-235 weight percent was calculated for the gamma spectroscopy and the alpha spectroscopy results. The paired results for this calculation are given in Fig. 2. The agreement in U-235 weight percent (between the two methods) is also good, considering it is a ratio of measurements which has a larger error caused by the compounding of the individual measurement errors.

Statistical tests of the data pairs were performed to evaluate the extent of agreement for the methods (8). The Pearson product moment correlation coefficient provides a dimension less index (that ranges in absolute value from 0 to 1) to measure the extent of a linear relationship between two data sets; the index is derived from the linear regression line fitted to the data points. The Pearson correlation coefficients for U-238, U-235, U-234 and total U results sets were 0.99, 0.82, 0.92 and 0.92, respectively. This indicates a strong linear correlation, as expected. A test of the significance of the differences in the paired results for the 45 samples (using the Paired Difference Method) shows no significant bias between methods at the 95% confidence level for the complete data set. A test of a reduced data set of the pairs obtained by discarding pairs at high and low concentrations to minimize the possible effects of homogeneity and the larger errors at low concentration showed a small positive bias for the U-234 pairs at the 95% confidence level. This small positive bias is the result of using a value of 22 for the U-234/U-235 ratio (to convert U-235 to U-234 for the gamma spectroscopy method) compared to a value of 19.3, which is the average of the alpha spectroscopy results for the site samples. The value of 22 represents the average plus the standard deviation of the alpha spectroscopy results; this provides a degree of conservatism in the U-234 content of the soil

samples and is the cause of the small positive bias indicated by the Paired Difference Method test.

SIGNIFICANCE OF RESULTS

A practical example of using gamma ray spectroscopy to provide isotopic and total uranium radioactivity measurements in soil contaminated with enriched uranium has been demonstrated. The use of a constant U-234/U-235 site specific radioactivity ratio is satisfactory to provide a reliable assessment of U-234 content using gamma ray spectroscopy. This methodology should be considered as an important assessment tool for the characterization and remediation of radiological situations involving normal uranium, enriched uranium and depleted uranium. The principal advantages can be summarized as follows:

The method is nondestructive and the cost per analysis is low compared to alternatives (40 to 75% less); and the QA/QC for this methodology is less cumbersome and more direct compared to chemical or wet radiochemical analyses.

Large sample volumes are directly analyzed, thus providing analytical information that is a better representation of the field sampling location, and less dependent on homogenizing the sample compared to methods that involve chemical processing.

The sensitivity of the method for uranium isotopes permits low level concentrations to be measured; the method is an alternative to alpha spectroscopy for uranium concentrations near or above 0.03 Bq/g (1 pCi/g).

Sample analysis turnaround can be as short as 10 minutes for a field located/mobile laboratory, to 1-3 days for the off-site laboratory. The analytical productivity is controlled by the specified MDC; a throughput of 20 to 50 samples per day can be expected using one spectroscopy system.

Quantitation of other observed gamma emitting nuclides can be performed on the same sample with no significant increase in the cost per sample.

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30-33

HEAP LEACH PROCESS FOR REMEDIATION OF PLUTONIUM- AND AMERICIUM-CONTAMINATED SOILS

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ABSTRACT

As an extension of previous successful studies for remediation of uranium-contaminated soil using adaptation of the heap leach process, results documenting the removal of plutonium and americium from soils are reported. In addition, results of leaching schemes from soils co-contaminated, not only with plutonium and americium but also with uranium, are reported. The first soils tested that contained plutonium and americium as contaminants were obtained from the Rocky Flats Plant site in Colorado, whereas soils that contained uranium, in addition to plutonium and americium, were obtained from the site of the former plutonium processing facilities at Los Alamos National Laboratory (which were active from 1945 to 1978). Heap leach has proved to be a relatively fast, low-tech, economical solution for the remediation of plutonium- and americium-contaminated soils at laboratory scale.

Using different leach reagents, more than two thirds of plutonium and/or americium present in the soil samples were successfully removed using the heap leach process. Uranium was also removed. Leach solutions used in experiments consisted of various compositions of citrate, bicarbonate, and/or dithionite. Comparative studies of the different reagents are reported and the results from Rocky Flats and LANL are compared.

The laboratory experiments show promise for remediation of soil by a simultaneous leach of multiple radionuclides.

INTRODUCTION

Several sites exist in the Department of Energy (DOE) complex with plutonium (Pu), americium (Am), and or uranium (U) soil contamination, including the former Rocky Flats Plant (RFP) and Feed Materials Production Center (FMPC) sites in Colorado and Ohio, respectively, and the Los Alamos National Laboratory (LANL) in New Mexico. Uranium-contaminated soils also exist, among other places, in the Czech Republic. Plutonium was previously successfully leached in the laboratory from soils samples from the Rocky Flats Plant (1). Soil samples from the RFP were from an area where drums containing Pu and Am contaminated liquids were stored and subsequently leaked. (The liquids consisted mineral, pump, and hydraulic oils. Other liquids in the drums included carbon tetrachloride, trichloroethylene, perchloroethylene, and ethanolamine.) Pu and Am were the primary alpha emitting radionuclides in those samples. Uranium was also successfully heap leached from uranium-contaminated soils from the FMPC (2), from the Czech Republic, and from LANL (3). Uranium contamination of soils from the FMPC was the result of both aqueous liquid spills and airborne emissions.

Simultaneous leach of U, lead (Pb), and beryllium (Be) from LANL soils was accomplished as well (4). The uranium-contaminated soils are associated with a non-nuclear firing site where highly oxidized depleted uranium (DU) metal came into contact with the soil surface. Simultaneous leach of Pu, Am, and U, however, has not heretofore been reported.

Actinides (such as Pu, Am, and U) tend to have a strong affinity for minerals in most soils (1). The mechanisms that normally dominate radionuclide sorption are surface complexation and ion exchange. Since the radioactive metals are Lewis acids (i.e., acquire electrons to reach an inert state), complexants that act as Lewis bases (i.e., have electron pairs that can be shared with the metal) can be utilized to leach Pu and Am from contaminated soils.

Triay (1) used several leach solutions for the removal of Pu and Am from RFP soil samples in batch desorption experiments. Those leach solutions consisted of various concentrations and mixtures of sodium citrate, ascorbic acid, sodium ascorbate, sodium dithionite, sodium persulfate, sodium bicarbonate, hydrogen peroxide, sodium hypochlorite, sodium hydroxide, sodium ethylenediaminetetraacetic acid (EDTA), and diethylenetriamine pentaacetic acid (DTPA). Best plutonium removal (62%) was attained with 0.1 M sodium citrate (Na3C6H5O7) + 0.1 M sodium dithionite (Na2S2O4) leach solution at 20C. Best americium removal (77%) was with 0.3 M + 0.01 M sodium bicarbonate (NaHCO3) + 0.5 M Na2S2O4 leach solution at 20C.

Turney (2) utilized 0.5 M NaHCO3 in unsaturated heap leach for removal of greater than 90% removal of uranium (U) from the FMPC soils at room temperature. Kitten (3) successfully removed U from the Czech Republic soil samples by use of unsaturated heap leach with 0.5 M leach solution at room temperature. Dander (4) used 0.5 M NaHCO3+ 0.1 M leach solution in unsaturated heap leach at room temperature for removal of U and Pb from LANL soils. Dander (4) also used 1.0 M acetyleacetone (C5H8O2) leach solution for removal of Be from the LANL soils by heap leach.

EXPERIMENTAL PROCEDURES

Based upon the work of Triay (1), Turney (2), Kitten (3), and Dander (4), the objective of this study was to investigate the removal of U, Pu, and/or Am simultaneously from LANL and RFP soils by use of unsaturated heap leach with leach solutions consisting of different combinations of 0.1 MNa3C6H5O7, 0.1 M NaS2O4, and 0.5 M NaHCO3.

Samples from LANL were taken from locations at the former plutonium processing facility, and included soils from beneath (denoted 3S) and near (denoted 27a) a former uranium processing laboratory and from building debris (denoted BD) associated with the decontamination and decommissioning of that facility. Soil samples from the RFP were the same as those samples described above.

The leach solutions used in the various experiments are shown in Table I.

Table I

Soil samples were all ~ 60 g and were heap leached in a downflow direction in 60 mL glass columns in an unsaturated environment (Fig. 1). Experiments were conducted at room temperature. Leach solution application rate was 0.012 mL/cm2/min (0.012 cm/min).

Fig. 1

Radioactive decay of Pu and Am emits primarily alpha radiation. Post-leach solutions were analyzed for Pu and Am by use of a liquid scintillation counter (LSC) that measured only gross alpha radiation. As such, LSC determined the total activity Pu and Am in solution. That activity is reported here as pCi/g. Post-leach solutions were analyzed for uranium by use of kinetic phosphorimetry analysis (KPA). Uranium is reported as mg U/kg soil.

RESULTS AND DISCUSSION

Alpha radioactivity was removed from both RFP and LANL soils with (1) sodium bicarbonate (hereafter referred to as bicarbonate, or B) and (2) sodium citrate and sodium dithionite (hereafter referred to as citrate-dithionite, or CD) leach solutions. Gross alpha of untreated RFP samples used in the experiments ranged from 45 pCi/g to 67 pCi/g. Approximately 10 pCi/g gross alpha was removed from the RFP samples by use of bicarbonate, and ~ 40 pCi/g gross alpha removed by citrate-dithionite leach solution. This suggested between 60% and 89% of the gross alpha activity was removed by the citrate-dithionite leach solution (see Fig. 2). Bicarbonate solution removed from 15% to 22% of the gross alpha activity in the RFP samples.

Fig. 2

While the bicarbonate solution removed up to 22% of the gross alpha activity in the RFP samples, bicarbonate solution removed nearly 70% of the alpha activity associated with the building debris samples from the LANL site (Fig. 3).

Fig. 3

Cumulative U removed from building debris samples from LANL was greatest with citrate-dithionite leach solution (Fig. 4). Sodium citrate-sodium dithionite-sodium bicarbonate (hereafter referred to as citrate-dithionite-bicarbonate, or CDB) removed less than 50% of what the citrate-dithionite leach solution did. Bicarbonate leach solution removed about 40% of the amount of uranium that the citrate-dithionite leach solution did. Sodium citrate-dithionite leach is solution was more effective for the removal of Pu and U than bicarbonate and CDB leach solutions.

Fig. 4

Dithionite is a strong reducing agent that will reduce oxidized plutonium (Pu(VI) and Pu(V)) to its more reduced oxidation state (Pu(IV) and Pu(III)). Dithionite will also reduce U(VI) to U(IV). At reduced oxidation states Pu and U will form strong complexes with citrate ions (5,6). The formation of citrate-Pu and citrate-U complexes would enhance the solubilities of reduced Pu and U and, thus, accelerate the removal of Pu and U from the soils.

A total of 37 mg U/kg was removed from the 27a LANL soil (Fig. 5). Uranium concentrations of 027a samples ranged from 8 to 231 mg U/kg. It is noted that RFP samples were soil samples containing clay, silt, and sand size particles. Building debris samples from the LANL consisted of cement (pH ~ 11) and were gravel sized with little, or no, clay, sand, or silt sized particles. Characterization of soils from the FMPC showed the U contamination in the soil (soil pH ~ 7) was associated with the clay sized particles (7). Large surface areas and strong electrostatic attractions make this scenario possible (8).

Fig. 5

CONCLUSIONS

The results of the experiments performed indicated that Pu, Am, and U can be removed simultaneously by the use of the 1) bicarbonate, 2) citrate-dithionite, or 3) citrate-dithionite-bicarbonate leach solution in unsaturated heap leach. Almost 70% of gross alpha activity associated with building debris from the former plutonium processing plant at the Los Alamos National Laboratory was removed by bicarbonate leach solution. As much as 89% of the gross alpha associated with soil samples from the Rocky Flats Plant soil samples was removed by the unsaturated heap leach method with sodium-citrate-dithionite leach solution. Uranium was also

removed simultaneous from the LANL samples during the leach process. The results show promise for remediation of soil by a simultaneous leach of multiple radionuclides.

ACKNOWLEDGMENTS

Thanks to Garry R. Allen of the Los Alamos National Laboratory Environmental Restoration Project for funding this project. Thanks also to Ins R. Triay, Leader of the Environmental Systems and Waste Characterization Group at the Los Alamos National Laboratory, for providing background studies and laboratory space for the execution of these experiments.

GLOSSARY

Am	americium
Be	beryllium
C5H8O2	acetyleacetone
DOE	Department of Energy
DTPA	diethylenetriamine pentaacetic acid
DU	depleted uranium
EDTA	ethylenediaminetetraacetic acid
FMPC	Feed Materials Production Center
kg	kilogram
KPA	Kinetic Phosphorimetry Analysis
LANL	Los Alamos National Laboratory
LSC	Liquid Scintillation Counter
mg	milligram
Pb	lead
pCi	pico-Curie
RFP	Rocky Flats Plant
Na3C6H5O7	sodium citrate
Na2S2O4	sodium dithionite
NaHCO3	sodium bicarbonate
Pu	plutonium
U	uranium

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URANIUM MILL TAILINGS : PHYSICOCHEMICAL PROPERTIES AND RADON EMANATING POWER

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ABSTRACT
Uranium mill tailings resulting from an acidic treatment were sampled at a Cogema site (Jouac, France) and separated into 5 particle size fractions by wet sieving. The morphology, the chemical and mineralogical compositions as well as radium-226 content were determined for bulk tailings and 2 selected fractions. The coarse particles were shown to be rough shaped and to form important aggregates when fine particles were present. Quartz, mica and feldspars were identified as the major components of the investigated materials in addition with clay for the fine fraction, but mineralogy and particle size fractionations were found to be dependent characteristics. Although the fines contained more radium than the coarses, direct correlation did not exist between radium concentration and particle size.
An experimental procedure was developed to determine radon emanating power using a sweeping type of collection. Measurements were performed at various moisture content and showed a slight effect of water on radon emanation from the coarse fraction while the emanating power of saturated bulk tailings and saturated fine fraction were respectively about 5 and 3 times those from dried materials. The apparent influence of size fractionation on radon emanation was found to be small for wet tailings. However, this study indicated that interferences between the properties of mill tailings may hide their respective effect on radon emanation process.
Key Words : Uranium mill tailings, radon-222, emanating power

INTRODUCTION

The uranium extraction from ores leads to large amounts of mill tailings, presently about 50 million tons in France. Although more than 90 percents of the uranium is generally extracted, those materials still contain radionuclides as thorium-230 and radium-226, which generate a radioactive

gas, radon-222. Without protective action, the dose associated with the ingestion and inhalation of short-lived decay products of radon may be high enough to cause concern for health of populations living in the vicinity of an uranium mill tailings (UMT) disposal. Knowledge of the radon source term is therefore required in order to assess its environmental impact. In the remainder of this paper the term "radon" will refer to radon-222 isotope. Release of radon to the environment involves two steps, escape from the individual particles in which it is formed and migration through the bulk medium to the free surface. Emanation is not to be confused with exhalation. Emanation only refers to the escape of radon from individual particles to the free pore space, whereas exhalation relates to macroscale radon release. The free pore space includes, in addition to the interparticle pores, the intraparticle pores that allow all radon atoms which enter it to diffuse out of the particle. In this paper, we treat the radon emanation and do not consider radon exhalation. The radon emanating power EP (also called emanation factor, emanation coefficient, escape to production ratio and radon leakage (1)) is defined as the ratio between the number of radon atoms entering the free pore space and the total number of radon atoms formed. The emanation of radon from various types of materials have been reviewed by Tanner (1,2). Four mechanisms have been proposed to account for radon emanation from solids. The recoil energy (86 keV) gained by the radioactive decay of radium-226 allows radon-222 atom to cover a finite distance, named the recoil range (40 nm in minerals, 100 nm in water, 65 m in air). The radon atom may therefore end its recoil in the free pore space. This mechanism is referred to as direct recoil. A second mechanism which may contribute to the emanation is diffusion through the solid matrix. It concerns the radon atoms located at the surface of the particle and becomes significant for very small particles with high diffusion coefficient (of the order of 10-16 cm2s-1). This mechanism is not likely for minerals, as has been reported by most of the investigators. A further mechanism which may be considered is the so-called indirect recoil : the radon atom goes through the free pore volume by recoil, lodges in the opposite pore wall and is released at a later time in the free pore space. Finally, the knock-out effect, as has been summarized by Semkow (3), has been proposed to explain high emanating powers of some substances which decompose easily into H2O or CO2, like hydroxides and carbonates.

Bossus (4), Semkow (3) and Morawska (5) have developed theoretical models of radon emanation from single mineral grains. The grain geometry was simplified for the sake of calculations. Bossus has considered the simplest case of a smooth sphere without any inner porosity. Morawska has included cylindrical pores of same diameter, parallel in each planes and Semkow has come close to the real structure of minerals with, among other things, a roughness surface concept. But UMT are produced by grinding rocks and have not a regular crystal structure. They are much more complicated than assumed for the models. Moreover, the parameters involved in those models (recoil range, intraparticle diffusion coefficient, shape of the surface, intragranular pore size, radium localization) are difficult to investigate with experimental methods. Emphasis was given to the recoil mechanism in those models, the diffusion through the solid phase being considered as negligible. Grain aggregation nor radon embedding into neighboring grains were included, so that those models might be seen as basis for a more complete model of emanation from

materials. Those models give important information about radon emanation from individual particles but cannot be used to determine the emanating power for UMT.

Elsewhere, numerous experimental studies (6-10) have been conducted on radon emanation from rocks, soils, minerals, or building materials for most of them. EP have been found to depend on physicochemical factors such as particle size, specific surface area, moisture content and temperature. The radon emanation from common accessory minerals have been shown to be very low (6). There are fewer published data on EP for UMT. The reported values (11-13) for UMT originating from various types of ores and treatments range from 0.07 to 0.31. Landa (12) has demonstrated that no consistent pattern enabling the prediction of EP of tailings on the basis of ore type was evident. The author has also shown that where wet sieving of UMT has occurred, fines tend to have somewhat EP than the coarse tailings. Studies by Strong (11) indicated that increasing moisture content enhances EP of Australian tailings, but different correlations - linear or more complicated - have been found between water content and EP depending on the type of the investigated materials. The various approaches taken to the problem and the lack of experiments descriptions make in fact comparisons difficult between all of this studies. The generalization of a given parameter effect on radon emanation to any UMT requires therefore further investigations.

The main objective of this laboratory study was to provide the magnitude of radon EP for one type of French UMT. Experimental measurements were also undertaken to assess the effect of the two factors that could be controlled in the investigated material, moisture content and particle size. The method used to separate UMT into particle size fractions, physicochemical properties of the bulk material and of two selected fractions are described in a first part. The experimental procedure to measure EP and the results obtained are presented in a second part.

MATERIALS

To avoid misunderstandings, it should be specified that "dry" materials refers to materials dried at 105C and cooled in a desiccator. The moisture contents are reported as the weight fraction of water, determined by subtracting the weight of the dry tailings from the weight of the moist tailings and dividing the difference by the weight of the moist tailings.

Sample Collection and Preparation

Tailings were collected at an active, acidic-leach uranium mill at Jouac in the Limousin, center of France (Socit des Mines de Jouac, COGEMA's subsidiary). They derived from granitic ores from Margnac (~80%) and La Besse (~20%), both next to Jouac. The processing, which included mechanical (f(95%) < 500 mm) and chemical (H_2SO_4, 130 kg per ton) treatments had removed about 98% of the initial uranium content. Tailings were sampled on filters before to be mixed with chemical slurries on the 17-05-94. Afterwards the moist tailings (19 wt %) were dried at 105C, hand crushed and stored per 1000 g aliquots in sealed containers at ambient temperature. The resulting material - called bulk tailings- were homogenized prior to sampling for any use (3D-mixer, Turbula). The bulk tailings had a mean moisture content of 0.3% at room temperature.

Size Fractionation

A wet sieving procedure has been derived from the one proposed by Landa (12).100 g aliquots of bulk tailings were shaken overnight (62 rpm) in 700 ml of deionized water and then submitted to ultrasonic waves for 30

mn in order to disrupt aggregates. The slurry was put through a nest of 20 cm in diameter stainless steel sieves (500, 200, 125, 45 mm diameter opening). The nest was mechanically shaken for 2 mn using an electric sieve vibrator (500 excursions per minute, with intermittent tapping). The material on the 500 mm sieve was abundantly washed with a fine stream of deionized water (1000 ml minimum). The fractions on the 4 sieves were washed one after the other above the nest of lower sieves and dried at 40C. The particles passing the 45 mm sieve were collected by centrifuging (59600 m s-2, 45mn) the entire slurry volume. These fine solids were dried at 105C and the clods were hand crushed to pass a 125 mm sieve. The particle size fractions resulting from that wet sieving are presented in Table I with size limits or theoretical cut diameters and weight proportions (percent of total weight). About 6.5% of the bulk tailings are water-soluble materials removed during the wet dispersion and sieving processes.

Table I

Particle Size Distribution

Fractions #2 and #5 were selected to test for the effect of particle size on radon emanation. We will therefore focus hereafter on the physicochemical properties of those fractions, as well as bulk tailings. The particle size distribution of fractions #2 and #5 were determined using a laser granulometer (CILAS HR850). The apparatus works in the range 0.1 to 600 mm and does not accept particles with diameter greater than 600 mm. Hence bulk tailings were sieved on a 500mm sieve before analyses. Table II summarizes some characteristics of the particle size distribution with the median diameter - by which is meant that 50% by volume of particles have a diameter smaller than this median diameter or f(50%) - as well as f(10%), f(90%) and the mean diameter. The reported results are averaged over at least 3 measurements.

Table II

Fraction #2 had a median diameter of 300 mm with 30% (volume) maximum of particles with diameter smaller than 200 mm and 7% (volume) maximum with diameter above 500 mm. Fraction #5 had a median diameter of 10 mm with at least 95% (volume) of particles with a diameter smaller than 45 mm. The apparent gap between the real particle size distribution of the investigated materials, the one determined with the granulometer and the theoretical sieving cuts might in fact be due to the shape of the particles. They are particularly rough indeed, most of the time without real similarity with ordinary geometric shapes (see Fig.1), when the separation and the analysis method are based on spherical particles hypothesis. It was then difficult to assess the real size distribution of those materials.

Morphologies

Small amounts of bulk tailings and fractions #2, #5 were deposited under a stereomicroscope without any sample preparation. Although the images observed by optical microscopy did not allow to probe details of the particle shapes or surfaces, they were sufficient to illustrate the complexity of UMT morphologies (see Fig.1).

The observation of fraction #2 (Fig.1c) showed various and not regular particle shapes and allowed to recognize crystalline structure such as quartz, mica and feldspar. The sample also included polycrystalline particles as can be seen in Fig.1c-d. The particle size distribution of fraction #2 appeared relatively wide. This is in agreement with the results obtained using a laser granulometer. Nevertheless, evidence was

given that no fines were aggregated with those particles. The particles in fraction #5 aggregated together in slightly larger entities as well as much larger ones (Fig.1b). The thickness of those aggregates did not allow to have a sharp image of the whole, but small and large cracks could have been observed. The surface of the fines could not have been examined. However, a rough estimation of the specific surface area was performed using methylene blue method (NF 18-592) and yielded 65 m2g-1. As could be expected, very important aggregation occurred in bulk tailings (Fig.1a - much smaller particles aggregated at the surface of coarse ones). This might suggest the presence of cementing phases between the particles in the materials.

Fig. 1

Chemical and Mineral Composition

Table III presents the chemical composition (major elements) expressed in % of oxides. Complete chemical analyses including minor and trace elements were also performed but did not give interesting information with regard to the present study. The analyses were carried out by Inductively Coupled Plasma / Mass Spectrometry (ERM, Poitiers, France). Table IV summarizes the semi-quantitative estimate of main identified minerals from chemical composition and X-ray diffraction analyses. Fe-bearing phases might be oxi-hydroxides or sulphurs.

Table III

Table IV

The particle size separations actually led to variations in the resulting fractions composition but no clear cut selection was made. Generally speaking, the same phases were found in bulk materials and fractions #2, #5 : mica, quartz and feldspars. The strong mechanical and chemical treatments involved by uranium extraction have led to a relatively homogeneous mineralogy (except for clays) with particle size. Nevertheless, mineralogy and particle size were not independent properties of the investigated materials.

Activity Concentration of Radium-226

Radium concentrations were determined by gamma ray spectrometry (214Bi ray). 15 to 25 g samples of bulk tailings or size fractions were sealed in 20 ml boxes and left for a minimum of 38 days for equilibrium to be reach between radium and progenies. The samples were then counted on a hyperpure germanium detector for 12 hours. The results shown in Table V are the average values over measurements on at least 3 different samples of the same material.

Table V

The radium concentration of the fractions increased with decreasing particle size, as have been reported by other investigators (5, 12), by more than a factor of 10 between the coarse and the fine fractions in the present case. However, fractions #1, #3 and #4 presented a radium concentration of the same order of magnitude than fraction #2 and no direct correlation between radium content and particle size fractionation was found in the present study. It is of interest to note that the fine particles (<45 mm) which did not pass a 125 mm sieve after drying and crushing contained 320 Bqg-1 and had a median diameter less than 1mm. The major part of cementing phases contained in the bulk materials might in fact be hold in this subfraction.

RADON-222 EMANATING POWER

Experimental Procedure

A known mass of tailings, dried at 105C and cooled in a desiccator, was deposited in a glass reactor (800 ml) to form a thin layer of thickness less than 1 cm. To avoid problems with hysteresis in the uptake of water (8), the sample was first supersaturated with deionized water (several millimeters of water above the layer surface) for at least one night. Water was then removed stepwise by evaporation at room temperature, so that experiments were started with saturated materials, carried on with intermediate moisture contents and finished with the sample dried at 105C (inside the reactor) and cooled in a desiccator. For radon measurements, the reactor was inserted in the setting up depicted in Fig.2. The surface of the sample was swept by an inert gas (N2, 1 lh-1), which carried the emanating radon to a collecting bag (20 l). If the investigated material was wet, the inert gas was first saturated with water by bubbling in a saline solution in order to avoid any drying up of the layer. A small fan prevented radon from accumulating above the layer. In such conditions the system was assumed to be at the atmospheric pressure. Temperature and relative humidity of the system were controlled by a probe all along the experiment. 7-9 hours were required before the system to reach a steady state as regard to radon concentration, depending on the sample type. After a sufficient time, the collecting bag was sampled using 2 evacuated scintillation cells (500 ml). These scintillation cells have been constructed by Algade (Bessines, France) and were standardized by air-bubbling in radium-bearing solutions. The scintillation cells were left for a minimum of 3 hr (for the secular equilibrium to be reach) and then counted for 2 hours using flask counters (PABS, Electronic Laboratory, CEA-
Far) and standard electronics. The gas circuit was then closed for a short time (<1 mn) in order to empty the collecting bag and opened again to allow gas collection. A second sampling was then performed about 2 hours after the first one.

Fig. 2

The values of the emanating power (EP) are calculated by Eq. 1 :

Eq. 1

The uncertainties on EP values were estimated according to a recommendation (INC-1) from the french NSB (BRPM). These uncertainties consist of a "systematic" component estimated by classical statistic methods on the basis of one standard deviation (gamma ray countings) and a "random" one, estimated from the maximum uncertainty of each parameter. The mean uncertainty on EP was 13%, the larger contribution being introduced through the dependence on gamma-ray spectrometric determination of radium concentration.

The thickness of the layer had to be chosen in order to allow all the emanating radon atoms to reach the surface, that is to avoid a loss of radon atom by decaying during the migration process. For this purpose, radon concentrations were measured from 10, 20, 30 and 40 g samples of moist (30%) bulk tailings. The same test was performed from 10 to 60 g on particle size fractions #2 and #5. The EP were found to be constant with these layer thicknesses. This is reinforced by the radon diffusion lengths reported in the literature (1, 14) as ranging from 0.1 to 3 meters. The effect of moisture content on EP was then performed with 30 g samples of bulk tailings and 40 g samples of fractions #2 and #5.

RESULTS AND DISCUSSION

Table VI presents the means of EP measurements for different values of moisture content of bulk tailings and particle size fractions #2 and #5.

They are plotted versus moisture contents in Fig. 3. Those results were obtained with stepwise evaporation from one saturated sample of each investigated material. The other EP measurements performed for sweeping time and thickness determinations were consistent with those series of results.

Table VI

Fig. 3

EP of bulk tailings was found to increase sharply with moisture content in the range 0 to about 5%. Measurements have not been completed with moisture content close to water saturation but the existing results suggested that EP did not vary so much between 5 and 16% moisture content. EP of fraction #5 increased also but slighter with moisture content in the range 0 to about 10% and then reached a plateau to complete water saturation. The ratio EP(wet)/EP(dry) was 4.75 for bulk tailings and 3 for fraction #5. They were relatively high values compared with the typical ones between 1 to 4 reported by Semkov (3). This water effect can be qualitatively explained by the direct recoil mechanism of radon emanation and the difference between the radon recoil range in air (65mm) and in water (100 nm) (1). When air fills the free pore space, the recoiling radon atoms may be embedded in an opposite solid surface. On the other hand, when water is present in the pores, many recoiling radon atoms will stop in it and increase the EP. In contrast, EP of fraction #2 was rather constant with moisture content up to 15% and then increased slightly to the saturation limit. This observation suggests that the pore sizes might be sufficiently large so that the recoiling radon atoms will stop in it, no matter if air or water fills the pore. The very low EP value measured for the dry bulk tailings may also be explained by the effect of pore size. As have been seen in studying the nature of those materials, they were composed with very fine particles and water-soluble compounds. When evaporating water from the sample, those particles were drawn closer to each other as well as to the larger particles than they were at higher moisture contents. This was confirmed by the increasing apparent densities from 0.60 g cm-3 for the wet sample to 0.85 for the dried one. The radon EP would therefore decrease because of the increased probability of the embedding process to occur. Cemented crusts were formed on the surface of the sample when drying the sample. This might also have led to a loss of radon by decaying before to reach the sample surface.

With the exception of the dry state previously discussed and generally speaking, no clear effect of size fractionation on radon EP was observed (Fig.3) Although radon concentrations measured from fraction #5 were much higher (8800 Bq m-3) than from other samples (2800 Bq m-3 maximum), its EP was only slightly higher than that of bulk tailings or fraction #2. Contradictory results exist in the literature : Greiner (8) did not measured any effect of particle size on radon emanation while Baretto (6) did. However, comparison with results obtained from coals or single minerals should in fact be avoided because variations in the particle size of UMT have been shown to involve changes in other properties of the materials such as their mineralogical compositions. The present study indicates therefore that it would be more suitable to consider the effect of the size fractionation rather than the particle size one for such materials. Moisture content might also interfere with chemical properties such as the localization of radium. Conclusions about the effect of a given parameter on radon emanation are in fact difficult to draw because

of relations of dependence between UMT properties. In these conditions, the contribution of the different emanation mechanisms previously reviewed could not have been further assessed.

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STUDY OF AN URANIUM MILL TAILINGS SITE: CHARACTERIZATION OF THE SOURCE TERM AND MODELING

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ABSTRACT
Uranium mining and milling have left behind great amounts of tailings containing both radionuclides (radium and uranium) and toxic elements (arsenic, lead, nickel,...) which may be released through aquatic pathway. The main exposure from uranium mill tailings is currently the inhalation of radon daughters. The contaminants release towards subsurface compartment and groundwaters is not of concern at present date. Nevertheless, potential modifications of such release in the long term have to be evaluated.
An uranium mill tailings site impact assessment related to aquatic pathways involves at first a complete definition of the source term (contaminants inventory, determination of the mechanisms responsible of their mobility in the tailings). The second step is the understanding of the numerous processes involved in the weathering of the tailings (pH and redox buffers, dissolution and precipitation of minerals).
In this paper, we present data obtained for the Lengenfeld site (located in the south of Dresden in Germany) which contains 300,000 m3 of tailings deposited 30 years ago and covered with 5 to 10 meters of sandy soil. No information was available about uranium ores, milling treatments and disposal conditions. From six drillings performed on the site, samples of the cover, the tailings and the underlying soil have been analyzed. From the results, we can assume that the tailings are issued from granitic and/or sedimentary ores which have been attacked by an alkaline solution (Na_2CO_3, $NaHCO_3$). The comparison with a well known site containing similar residues (basic treatment) may be used to improve present knowledge of the evolution of the Lengenfeld tailings. The analysis of the tailings, specially the determination of the amounts of the different minerals present at different depths, has allowed to point out a beginning of weathering through dissolution and precipitation reactions. The experimental data will be compared to simulations performed by means of geochemical codes, based on thermodynamic equilibrium.
Key Words : uranium mill tailings / radium / uranium / geochemical modeling
INTRODUCTION
For more than a quarter of century, tailings and waste rocks resulting from uranium mining and milling operations have been deposited in impoundment areas throughout the world (about 50 millions of tons until 1991 in France)(1). The tailings issued from milling and chemical processes have the texture of a fine sand and represent about 98% by volume of the extracted material with 85% of the radioactivity found in the original rock. Many of the remaining radionuclides have very long half-life times (80,000 years for 230Th and 1,600 years for 226Ra). As a result, although the radioactivity may be very low, some radiological hazards can be considered to last forever. Moreover, the uranium ores generally contain metallic elements (arsenic, copper, cadmium, silver, lead,...) which remain in the tailings. These elements may cause concern for health, and environment for long periods of time because of their permanent toxicity. The health hazards occur in a number of pathways via radioactive gas or dust inhalation, external exposure and through contamination of surface and groundwater. The radionuclide concentrations in waters are controlled partly by the tailings which act as a source

term, and partly by the hydrochemistry resulting from the mixing of surface, tailings and shallow waters.

This work is devoted to the term source definition considering the aquatic pathway. This includes the determination of the amounts of contaminants present in the tailings and the mechanisms responsible of their release in the geosphere. Their knowledge requires a complete characterization of the tailings, the surface layer and the underlying soil.

HISTORY AND DESCRIPTION OF THE LENGENFELD SITE

The uranium mill tailings site is located in Lengenfeld (Vogtland) at 120 km of Dresden in the east part of Germany. It is closed to the uranium ore mill, which has been in production from 1946 to 1961. The mill and the tailings site were managed by Wismut Society. Until 1991, no details about the organization and the productivity of Wismut were available. After the German unification and because of public pression, some information have been obtained from Wismut's old workers, specially concerning the amounts of uranium produced in east Germany (220000 tons between 1946 and 1989) (2). But the origins of the uranium ores and the milling processes are still unknown.

In Lengenfeld, the tailings (volume of about 300,000 m3) have been deposited 30 years ago in a valley whose river has been diverted in the east part of the site. These residues are surrounded by a dam and covered by a 7 meters thickness of sandy soil.

EXPERIMENTAL PART

Field Experiments

Six boreholes drilled from 0 to 20 meters deep have given a total of 60 core sections including surface layer, tailings and underlying soil (Fig. I). The water table is very close to the surface layer and the cut cores were saturated in water. These solutions in contact with solids have been obtained by pressing down with a 100 bars squeezer. After a filtration through 0.22 mm pore size filters (Millipore), the pH of the solution was immediately measured with a Portatest 655 pH-meter equipped with an Ingold combined electrode, and then was adjusted to 1-2 with Merck 65% Suprapur HNO3 for further analysis.

Fig.1

Laboratory Experiments

Eq. 1

Solid analysis : A laser apparatus (Cilas HR850) was used to measure the distribution of the particles as a function of their size. The chemical and mineralogical composition were determined for several samples of surface layer, tailings and underlying soil. Radionuclides contents (226Ra, 238U and 210Pb) were measured by g spectrometry with a high resolution hyperpure germanium spectrometer. The detection limits (DL) were 0.1 Bq.g-1 for 226Ra, 0.7 Bq.g-1 for 238U and 0.4 Bq.g-1 for 210Pb.

Solution analysis : classical chemical analysis were performed on solutions obtained at different depths. 226Ra and total uranium were analyzed respectively by 222Rn emanometry (DL = 20 mBq.l-1) and laser fluorimetry (DL = 20 mg.l-1).

MODELING

The numerical simulation of geochemical reactions has been performed with the code chimere (3) and the software The Geochemist's Workbench (4). Both are founded upon thermodynamic equilibrium in aqueous solution in order to deal with species speciation in solution, dissolution-precipitation of minerals or sorption reactions. The thermodynamic

database of The Geochemist's Workbench is based in large part on the supcrt data compilation and on the Dzombak and Morel data set (4). Whereas, the chimere thermodynamic database originates from data compilation achieved within the European Communities chemval project (5).

RESULTS

Analysis have been conducted on numerous samples and only a mean composition of each component (surface layer, tailings and underlying soil) is presented in Table I.

Surface Layer

The mineralogical phases are mainly quartz (65%), micas (7.5%), fluorite (9.5%) and carbonates (3.5%). Kaolinite and feldspars appear as minor components (respectively 2.5% and 1.5%). High amounts of barium (closed to 10,000 ppm) have been determined in all samples of this layer. The particles mean size is closed to 150 mm.

The evolution of the cores colour (greenish to reddish) occurring during the two hours following the drilling is probably due to the oxidation of ferrous ions in the presence of oxygen. A detailed analysis of iron in the solid has shown that this element is present partly as ferrous ions (40% of the total iron) and partly as ferric ions (60% of the total iron). Organic carbon has been detected in solid samples (0.1% in weight) and is probably responsible for the weakly reducing conditions of the surface layer.

Considering the water table, the first centimeters of the surface layer (below one meter deep) constitute probably an unsaturated zone in which precipitation/dissolution reactions can take place because of water evaporation. Therefore, the high concentrations measured at this depth are not representative of the covering layer.

Table I

Solutions and solids obtained at different depths in the surface cover have been analyzed (Table II and Table III).

Table II

Table III

Calcium, magnesium and sulphate concentrations decrease with depth while sodium and potassium concentrations increase, iron concentration remaining constant. The pH increases slightly (7.0 to 8.2). Several processes can be involved to take into account these variations like ion exchange, dissolution and/or precipitation of mineral phases, sulphates reduction. Different hypothesis have been tested through speciation modeling. The solution compositions have been introduced into the two geochemical codes under the following constraints: pH fixed to the measured values and redox potentials Eh slightly below 0 mV. In both cases, the total concentrations in calcium, magnesium and carbonates are mainly controlled by the carbonate phases, calcite $CaCO_3$ and dolomite $CaMg(CO_3)_2$. The carbonates also act as pH buffers.

Of course, the computed total concentrations in calcium, magnesium are dependent on the equilibrium constants introduced in the geochemical database. The solubility constant of calcite is well known ($K_s = 10^{-8.3}$) indeed. But large discrepancies occur in literature data for the dolomite solubility product ($K_s \sim 10^{-16.5}$ to $10^{-19.5}$). In fact, dolomite found in nature is seldom pure (stoichiometric composition) because of the incorporation of numerous divalent cations (Ca^{2+} in most of the cases), leading to the formation of a magnesian calcite ($Ca_xMg_{(1-x)}CO_3$). For solutions in equilibrium with both calcite and dolomite, the $[Ca^{2+}]/[Mg^{2+}]$ ratio is constant and equal to 1.3, which corresponds to

the solubility ratio (with a solubility constant for dolomite about 2x10-17). The variations observed in the different samples may be due to the evolution of carbonate phases between two pure phases (calcite and dolomite) through the formation of solid-solution (6).

Tailings

Several samples obtained at different depths have been analyzed and their chemical compositions are presented in Table IV.

Table IV

The mineralogy of the tailings is quite different from those of the surface layer with high amounts of micas (35%) and feldspars (7.5%). Carbonate is present only as dolomite (7.5%) and 60% of iron is present as ferrous ion. Organic carbon amount is higher than in the surface layer and is probably responsible for the reducing conditions found within the tailings (a colour evolution, from greenish to reddish, has also been observed on the cores after drilling). In the opposite of the surface layer, a very low amount of sodium is measured in all the tailings samples. The clays amount is significant with kaolinite (9%) and swelling clays (2.5%). The granulometry of the tailings is quite homogeneous and the mean particle size is closed to 10 m. This particle size has forbidden any more detailed mineralogical studies through microscope observations. Therefore, considering the results presented above, it is not possible to conclude about the origin of the ore (granitic or sedimentary). Keeping in mind that this mill facility has been chosen because of the existing structure and the important railway network, we can assume that all kinds of ores have been treated in this mill inducing a possible heterogeneity in the tailings.

The radionuclides contents are presented in Table V.

Table V

The concentrations of 226Ra, 238U and 210Pb decrease slightly with depth, but these variations are not significant. In the ore, the two ratios 238U/226Ra and 226Ra/210Pb are closed to the unity because of the secular equilibrium. Whereas in tailings, uranium has partly been removed in consequence of the ores treatment and thus the secular equilibrium between 238U and its daughters has been disrupted. From these data, an estimation of the extraction yield gives a value of 84% which is relatively lower than those obtained in France (between 93 and 99%).

The volumes of solution obtained by squeezing are very low (less than 20 ml in most of the cases) and only few elements have been analyzed (Table IV). Among others, the bicarbonate contents in solution still remains to be measured by this method. However, charge balances and in field results from boreholes lead to carbonate concentrations which are close to 10-2 mol.l-1.

High concentrations of sodium (6x10-2 mol.l-1) with high pH values (closed to 10) are obtained. These data suggest that a basic treatment has been applied to the uranium ores, the chemical reagent being probably NaHCO3. Such a treatment is usually applied to sedimentary ores combined with a more pronounced milling than for granitic rocks. For example, the mean size of tailings obtained in Lodve (sedimentary ores in France) is 17 mm (7). The particle size measured for Lengenfeld tailings is in good agreement with a basic treatment. However, this result is insufficient to assert that all the ores were issued from sedimentary rocks.

Only two radionuclides, U and 226Ra, have been measured in solutions (Table V). The distribution coefficients of radium (closed to 20,000 l.kg-1) are in the range proposed in the literature (7). The

radionuclides concentrations differ strongly between the two samples with an increase of pH with depth. The chemistry of radium (as all the alkaline earth elements) is dominated by their ease of oxidation and the ionic nature of their metal-oxygen bonds. Only the divalent stable form is known. Geochemical simulations show that concentrations in anions (such as SO42-, Cl-, OH-) as are not high enough to form complexes with radium. This element is thus mainly present as the divalent species Ra2+. The chemistry of uranium is more complex because of the multiplicity of oxidation states, organic and inorganic complexes and insoluble compounds.

The estimated permeability is about 10-11 m.s-1, which induces a very low water flow. In these conditions, we can reasonably assume that equilibrium between solid and solution is reached in any point of the tailings. A first modeling of the water-rock interactions within the tailings has been performed under the constraint that pH values were fixed to the measured ones. As previously discussed, the alkaline leaching of the uranium ores has led to a pH which is about 10 and to a Na2CO3 solution type. The magnesian calcite, through precipitation-dissolution reactions, acts as a pH buffer and controls the calcium and magnesium solution composition. The perennity of this pH buffer is ensured in tailings by a high carbonate contents and a low permeability. Although the high content in organic matter measured in the tailings is probably responsible for the perennity of the reducing conditions, redox in solution is often controlled by Fe(II)/Fe(III) reactions. The Fe(II) content in the tailings is relatively high. Thus, for the solution compositions encountered within the tailings, the following reaction seems to be reasonable: FeCO3(s) + 3 H20 Fe(OH)3(s) + HCO3- + 2 H+ + 1 e-. The combination of this reaction and of the total iron concentration measured in solution leads to a redox potential Eh which is - 220 mV. Regarding the uncertainty about the bicarbonate content in solution, another mechanism can be proposed for such a high pH: Fe(OH)2(s) + H20 Fe(OH)3(s) + H+ + 1 e-. This reaction gives a Eh which is quite similar to the previous one, about - 250 mV.

The stability of alumino-silicate minerals has also been computed relatively to the low temperature and the chemical aqueous conditions found in the tailings. For such an alkaline pH, a sequence of stability can be propounded: micas and swelling clays > kaolinite, quartz and Na-K-feldspars > SiO2(am) and Ca-feldspars. This is just a theoretical proposal based on thermodynamical considerations, and which needs to be confirmed by further experimental studies.

For such reducing conditions and high carbonate contents, it appears that uranium is present as an anionic species UO2(CO3)34- at pH equal to 10.2 (> 10) (8). But for pH < 10, the most stable form of uranium is the tetravalent state which tends to precipitate as UO2, xH2O. Therefore, for the sample obtained at 9 m (pH = 10.2), uranium is present as an anionic species, weakly sorbed on solids. For the sample obtained at 11 m (pH < 10), uranium concentration in solution is controlled by the solubility of UO2, xH2O, inducing a lower concentration in solution. These modeling data are in rather good agreement with experimental results.

Radium is present at trace level in the tailings, respectively 1.60x10-10 mol.l-1 at 9 m and 3.25x10-11 mol.l-1 at 11 m. So, it cannot precipitate as a pure solid compound. The behavior of radium in these samples can be controlled by different processes : sorption onto solids, cation exchange on clays and co-precipitation on carbonate phases. With acid uranium mill

tailings, stress has been put on the co-precipitation with barite ($BaSO_4$). In Lengenfeld, the geochemical modeling indicates that the SO_4^{2-} aqueous concentration is not high enough for such a process to occur. On the contrary, the interaction of radium with carbonate phases (especially on magnesian calcite) is not well-known and further attention has to be paid on it because of the chemical behavior analogy of Ca, Mg and Ra.

Underlying Soil

Few samples have been analyzed (Table VI). The characteristics of the solid phases are quite representative of a river sand : wide range of particles size (2 to 500 mm), smooth shapes of these particles and high content of silica.

The amount of carbonate is lower than in the tailings but the amount of clays is quite similar. The sodium concentration is higher in the soil than in the residues.

Table VI

Radionuclides concentrations have been measured in this layer up to a depth of one meter (Table VII).

Table VII

The $^{210}Pb/^{226}Ra$ ratio is about 0.9, which is slightly lower than in the tailings. The contamination plume measured in the underlying soil may be due to the migration of small tailings particles or to the migration of uranium and radium as ionic species. The $^{238}U/^{226}Ra$ ratio is higher (0.60) in the soil than in the residues (0.18). Within the tailings, the distribution of particles sizes is narrowly centered on small radius. And so, the distribution of Ra and U relatively to the particle size is probably quite homogeneous. Therefore, the contamination in the underlying soil is not only induced by the migration of small particles. In this case, the ratio would be the same than in the tailings. The discrepancies observed for the different samples are probably due to the combination of the two processes: migration as particles and as ionic species. During migration in solution, sorption and precipitation reactions are different for Ra and U, and a shift between the Ra and U migration fronts occurs.

Only one solution has been analyzed and the data are presented in Table VI and Table VII. The pH is lower than in the residues (8.3) and no uranium is detected in this solution. This result is in good agreement with the uranium speciation. In reducing conditions, for $pH < 10$, uranium is present at the tetravalent state and precipitates as UO_2, xH_2O. Therefore, the concentration level in solution is lower than the detection limit. The distribution coefficient KD of radium in underlying soil is about $1x10^4$ $l.kg^{-1}$. The radium is always present as a divalent cation which can be sorbed on solids like clays, feldspars and micas. Few data are reported in literature (10, 11) and it is difficult to elucidate the mechanisms responsible of this high KD value.

DISCUSSION

Over centuries, the rich deposits of metallic ores in the Vogtland county were mined intensively. Wolfram ores currently are encountered and surrounded with barite veins. After the extraction of this metal, the residues are mainly constituted with sands containing barite. Before to be devoted to the treatment of uranium ores, the plant located near the site was used for wolfram extraction. Therefore, it is not surprising that this sand, probably present in high quantities in the near environment, as a covering layer. The modeling indicates that solution composition is partly controlled by carbonate phases. Calcium and

magnesium concentrations being different in the three samples obtained at different depths, one can assume that the carbonates phases which control the system are slightly different at the different levels (calcite, dolomite, magnesian calcite). The permeability of this layer (assuming a granulometry closed to 150 mm) is about 5x10-8 m.s-1 and it is possible that an evolution of the system (dissolution and precipitation processes) occurs, inducing variations in major elements concentrations. More detailed characterizations of these carbonate phases will allow a more accurate modeling of the covering layer system. This is an important topic for the prediction of the solution composition issuing from this layer and flowing down the tailings.

The tailings are constituted of very fine particles (10mm) in contact with reducing basic solutions (pH > 10). Considering the solution composition, we have assumed that a basic treatment (probably NaHCO3) has been applied to extract uranium. In these conditions (reducing medium, pH > 10), the solution composition is controlled by carbonates (magnesian calcite), by Fe(II)/Fe(III) reactions, and by slower and more complex organic matter and alumino-silicates evolutions. Uranium is present as an ionic carbonate species ($UO_2(CO_3)_3^{4-}$). As soon as the pH slightly decreases (< 10), uranium precipitates as UO_2, xH_2O, inducing a lower amount of mobile uranium. Considering these data, one can assume that as long as carbonate phases will be present in the tailings, the pH will be fixed at a value closed to 10. In these conditions, uranium will be present as an anionic mobile species. But after a full consumption of this solid, the pH will decrease and uranium will become less mobile due to its precipitation.

The speciation of radium in solution is easy to predict but few data about its sorption onto different kinds of solids are available. Therefore, complementary studies have to be conducted on the sorption mechanisms which may be responsible of the retardation factor of radium in the migration process.

Finally, the analysis of few samples of the underlying soil has shown that the radionuclides have soon reach one meter of depth. It is difficult to elucidate the mechanisms responsible of this plume. However, the $^{238}U/^{226}Ra$ ratio was sufficiently different from those measured in the tailings to led us to assume that this plume cannot be due to the single movement of fine tailings particles. More probably a combination of several processes is involved.

This study has also allowed to build a part of the history of the site. A basic chemical treatment has been applied to different kinds of ores which have been submitted to an intensive milling. Therefore, a comparison with others sites containing analogue residues will be possible, in order to improve our knowledge on the history of this site. Moreover, the tailings have been deposited in this valley 30 years ago. They are saturated with water and we can assume that the weathering of the solids have soon begun. Particularly, the microscopic observations of the different minerals will bring some information on the behavior of radium in these weathering process (incorporation, surface sorption). These data are necessary to predict the long term behavior of the contaminants and therefore, to build a strategy for the long term management of this kind of residues disposal.

ACKNOWLEDGMENTS

This work was supported by European Communities (Contract N F13P-CT93-0075). We thank Mr G. Biton for his technical assistance and Mr J. Larue for helpful discussion.

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30-36

EVIDENCE OF DIAGENETIC PROCESSES IN URANIUM MILL TAILINGS FROM THE ECARPIERE URANIUM DEPOSIT (COGEMA-FRANCE)

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ABSTRACT

An integrated analysis of a tailings impoundment, ECARPIERE (Western part of France) was performed in order to get a good understanding of the behavior of radionuclides and associated heavy metals.

Three drill holes performed on the tailings impoundments (former open-pit and ring dyke) give a good outline of the mineralogical and geochemical modifications accruing in the tailings which, in some cases, have been stored for nearly 40 years.

Mineralogical and geochemical investigations of the solid fraction and isotopic charac
terization of the tailings pore water were made from the top to the bottom of the storage all along the drillcore ; they lead us to the conclusion that the stored tailings have undergone a significant diagenesis.

Mineralogical Modifications consist of :

- precipitation of new mineralogical paragenesis: smectite, gypsum, barite, hexavalent uranium minerals ...,
- dissolution of residual minerals such as sulfides, feldspars, biotite, monazite, etc....
- close relationships between some heavy metals, radionuclides and the newly formed minerals.

Leaching Test performed on both stored and fresh tailings lead to the same conclusions as mentioned above. The radionuclides and associated heavy metals releases are higher in a fresh tailings than in a stored one ; development of smectite does not appear in fresh tailings.

The 230 Th/226 Ra and 226 Ra/210 Pb ratios are in desequilibritum in the acidic tailings.

The main conclusion of these studies is: a tailings impoundment is not inert, deep modifications appear short time after the deposit especially in the saturated zone of the tailings. These modifications will have an influence on the mobility of the radionuclides and heavy metals in the tailings and consequently on their release toward the environment.

INTRODUCTION

Uranium mining activities started in FRANCE in the early fifties and during more than 40 years 70.000 tons of uranium have been produced; uranium was mined in there main uranium districts located in hercynian leucogranites, permian black shales and tertiary sandstones. These deposits are mainly vein types, episyenite body and stratoid lenses. Uranium was recovered by dynamic or static leaching using acidic and alkaline solution, 46M.Tons of waste (29 M.Tons of tailings and 17 M.T of heap leaching wastes) were produced and stored on the mill sites or in the vicinity of them.

In order to reduce the impact of our mine sites (openpit - underground mines, tailings impoundments, mill facilities, heap leaching dumps) remediation works have been undertaken on the 170 mine sites with the following principles and objectives :

Principles:

Highest efficiency
Final impact must comply with regulatory constraints
Residual impacts are made as low as reasonably achievable
Large information and participation of the public.

Objectives:

Long terms stability of the remediated area
Prevention of intrusion
Control and limit the long term radiological impact
Choice of natural barriers to rely on passive controls and reduce future technical supervision requirements

Reduction of the total land consumption and need for institutional control

Integration into the surrounding landscape, and if possible future use of land

Project technically and economically workable.

Ecarpiere Uranium Deposit

ECARPIERE is located in VENDEE W.FRANCE

Mining (underground - open pit) 1953 - 1990

Milling (9300 kt 1,5 /) 1957 - 1991 (acidic treatment)

Heap leaching (4000 kt 0,4 /) 1967 - 1991

- Remediation 1991 - 1995 (Recontouring, Cover, Drainage + Water treatment)
- Open pit mining.....115 ha
- Mill.....6 ha
- Heap leaching facilities.....16 ha
- Underground mining installations.....12 ha
- Heap leaching waste dump.....9 ha
- Mill tailings impoundment.....73 ha
- Waste water collecting zone.....9 ha

Tailings (7 M tons) from l'ECARPIERE have been stored behind a ring dyke for more than 30 years; to reduce radon emanation and water infiltration the tailings are covered by a multilayer cover.

PETROGRAPHY AND MINERALOGY

There drill holes (C1 - C2 - C7) were performed on the tailings impoundments in order to get a representative sample which allowed us to study the petrography, mineralogy and geochemical characteristics of the tailings from the top to the bottom.

Two types of minerals can be observed :

The main clastic minerals representing the nature of the initial hostrock and the ore paragenesis; these minerals are : quartz, feldspars (Na - K), muscovite, biotite, smectite1 (Al rich), calcite, zircon, pyrite, monazite, tourmaline, fluorite, iron and titanium oxides ; other rare minerals such as marcasite, covellite and graphite are sporadically found.

The second type of minerals can be considered as authigenic, they appear as a cement in all the samples ; gypsum, barite, jarosite, uranium phosphate, uranium vanadate, uranium lead silicate, aluminous rare earth phosphates, iron oxydes and hydroxides, iron phosphate, amorphous silicate and smectite 2 (Fe rich). All these minerals cristallize after deposition of the tailings and a mineralogical zoning can be observed from the top to the bottom of the storage : the smectite to illite ratio increases with depth.

There is a close relationship between the percentage of clay minerals and the radionuclides (226Ra) content.

The dissolution of pyrite, the formation of ion oxyhydroxides around clastic and authigenous minerals, the recrystallization of hexavalent uranium minerals and the development of Fe smectite indicate the diagenetic evolution of the tailings. All these authigenic minerals are not observed in fresh tailings coming from the mill.

LEACHING TEST DATA

Leaching Data

Results on the contaminant releases from the cored mill tailings show that sulphates (mainly gypsum) are mobilized. Uranium and other heavy metals are not released.
The release of 226 Ra is correlated to gypsum and not to barium. After gypsum is totally leached, radium mobility is weaker.
Some tests were carried out in the same conditions on a fresh and stored mill tailings of a Limousin plant (same acid hydrometallurgical process). They revealed the same significant difference towards the mobility of 226 Ra.
In fresh mill tailings, 226 Ra is released in significant amounts, even after the sulphates leaching, whereas it is much lower in an aged mill tailings sample. As mentioned above no recrystallization of authigenic minerals such as smectite, gypsum, barite etc..... have been described in fresh samples.
This behavior can be explained by the diagenesis that the tailings have undergone during their storage.

GEOCHEMISTRY

Major and Trace Elements Data From the Solid Tailing Samples

The geochemistry of the tailings depends on 1) the nature of the ore, 2) the chemicals added during the milling process such as H_2SO_4, $NaClO_3$, and $CaCO_3$ and $Ca(OH)_2$ and 3) possible elements added through water treatment such as Ba.
A relatively regular evolution with depth is noted. The Mg, Ni, V, Cr, Co, Cd, Ti and Fe contents increase from the base to the top of the tailings pile whereas the Cs, Be, Bi, Ge, In, Ga, Rb, Sn, Ta, W and Nb contents decrease. This reflects a change of the processed ore type from hosted granitic ore to hosted metamorphic ore.
The behavior of Ba, Mo and Sr is variable and there is no relation between Ba and K_2O. This is probably due to the occasional addition of sludges resulting from the water treatment on the top of the tailings.
In solid tailings sample, 226 Ra is best correlated with PO_4.

Geochemical Data From the Porewater Samples

The shape of the vertical profile of radium is roughly similar to the gamma ray profile and to the Sr profile. They linearly increase from the top to about 14 meters deep and then decrease in the water satured zone. At this depth, there is also a higher Ba and U content in the water coupled with a depletion in the solid fraction.
Some elements increase with depth such as Si and Al. For several elements the last two porewater samples are different.

URANIUM SERIES DISEQUILIBRIUM

Analytical Methods

The 238U, 230Th, 226Ra and 210Pb activities were measured by gamma spectrometry on 20g of dried samples.

238U, 230Th, 226Ra and 210Pb Activities

The 238U, 230Th, 226Ra and 210Pb activities vary from one layer to another.
As an example for C1, the average activities of the mill tailings are :

- 1475 Bq/kg of 238U,
- 23900 Bq/kg of 230Th,
- 20500 Bq/kg of 226Ra,
- 25400 Bq/kg of 210Pb.

The residual 238U activity represents from 1 to 12% compared to the activity of 230Th.
It increases at the bottom of the tailings.

The 230Th/226Ra and 226Ra/210Pb activity ratios deviate from 1 in the mill tailings. In the C1 and C2 holes the 226Ra/210Pb activity ratio increases with depth from about 0.75 to 0.95 whereas in the third one it remains constant at about 0.8.
The disequilibrium rapidly reverses in the substratum and the activities decrease drastically. The 230Th/226Ra activity ratio are higher than 1 in the mill tailings and strongly increase in the surface altered zone of the granite, just below the tailings.
The values of 226Ra/210Pb ratio below 1 indicate that the isotopes have been leached differently in the tailings.

CONCLUSION

The main conclusions from this study are

1. The mill tailings have undergone a significant diagenesis during the last 30 years with the neoformation of minerals (smectite, gypsum, barite, hexavalent U-minerals) and the dissolution of residual minerals (pyrite, feldspars ?...).
2. The 230Th/226Ra and 226Ra/210Pb activity ratios are in disequilibrium in all the mill tailings indicating a different behavior of the radio-isotopes during diagenesis. The radium content of the porewaters range from 1Bq/l to 12Bq/l.
3. The leaching behavior of radium and other metals is different in fresh and stored tailings.
4. The geochemical characteristics of the porewater in the unsaturated and the saturated zones are different and the contact is the location of some metals deposition.

ACKNOWLEDGMENTS

This study was undertaken under a CCE contract entitled FISRAMUT.

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30-38

ENVIRONMENTAL DISTANCE LEARNING AWARENESS TOPICS FOR DOE, INDUSTRY AND UNIVERSITIES

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Technical Training Materials Using the Latest Instructional Systems Development (ISD) Technology

ABSTRACT

Most organizations are experiencing changes which include downsizing, funding cutbacks, shifts in workforce, new missions, new funding sources which demand a swift reaction to train or re-educate the workforce. Those organizations that continue to remain competitive have been able to accommodate to the changes imposed by a technological society using technical training materials incorporating the latest Instructional Systems Development (ISD) technology.
Only 10% of all trainers are familiar with the latest technology in distance education. The remaining 90% prefer traditional tools for

teaching and training (stated at a '95 American Society of Training and Development [ASTD] Conference). Recently, NETN broadcast a "meeting of the minds" to teachers across the U.S., wherein academics came together to discuss the pros and cons of technology and media in the classroom. They determined that chalkboards, flip charts, and pen and pencil tests far outranked ISD technologies such as interactive advertising on television, CD-ROM-based training, in popularity and usage. Yet today's workforce is exposed to new technologies and their uses on a daily basis. Talk show hosts use video telephones, industrial moguls are experimenting with selling products using interactive advertising on television, and business executives have equipped their satellite offices with videoconferencing equipment to cut down on travel time. More and more information from executives and their employees is prepared and presented in new formats involving multi-media.

Environmental issues have had such a substantial impact on the way we live and do business over the past several years, it is imperative that up-to-date training is available. This training must be easily accessible and affordable. Distance Learning using ISD meets this criteria.

INTRODUCTION

Only 10% of all trainers are familiar with the latest technology in distance education. The remaining 90% prefer traditional tools for teaching and training (stated at a '95 American Society Training and Development [ASTD] Conference). Recently, NETN broadcast a "meeting of the minds" to teachers across the U.S., wherein academics came together to discuss the pros and cons of technology and media in the classroom. They determined that chalkboards, flip charts, and pen and pencil tests far outranked ISD technologies such as interactive television, CD-ROM-based training, in popularity and usage. Yet today's workforce is exposed to new technologies and their uses on a daily basis. Talk show hosts use video telephones, industrial moguls are experimenting with selling products using interactive advertising on television, and business executives have equipped their satellite offices with videoconferencing equipment to cut down on travel time. More and more information from executives and their employees is prepared and presented in new formats involving multi-media.

It is incumbent upon both the student and the employee to stay abreast of the new communication technologies. With the digital revolution, anything can be rearranged, cut apart, and mixed with anything else that is digital. It is possible to create a digital manual that can be called up on a home computer. So many things are possible: interactive television, video phone calls, long distance learning. The question that must be answered first is what will the audience members readily accept? The questions of cost, duration, quality and effectiveness must follow, and be answered in light of the changing mission, culture, and objectives. Traditional classroom/workshop training methods, while familiar and proven, cannot supply training that crosscuts all EM components with the same efficacy as ISD technologies. The computer literacy of the workforce is on the rise, and the availability of travel/training dollars is in decline, thus ISD technologies are the best hope for softening the culture shock of organizational change, as well as using the existing skills of the workforce in the most efficient manner possible, while demonstrating considerable cost savings. As Secretary Hazel O'Leary pointed out in her May 3, 1995 address, an annual savings of $175 M can

be achieved for DOE in part by the use of videoconferencing (an ISD technology) for training.

Just as on-line software companies are struggling today to define themselves, to come up with something distinct that will make them stand out, so too, must learning programs carry a distance and unique label, focusing on both the network and the content. Learning objectives must not only be delivered by the latest technology but produced in a visually (if the technology is television) exciting manner.

In the business arena marketing strategies have moved from highlighting the technologies to heralding the services. The spotlight is now upon the customer. At NETN, we always begin with the customer.

The University of New Mexico has established a history of ISD technology courses for two national laboratories in the Albuquerque area: Sandia National Laboratory and Los Alamos National Laboratory. These full-credit academic courses are broadcast to 30 industry sites in the Albuquerque area via a line of site microwave antenna utilizing an instructional television fixed system (ITFS). Over 60 engineering courses are broadcast each semester, establishing UNM as a leader among universities with total quantity of academic course broadcast. Selected academic courses are also broadcast through the National Technological University (NTU) in which UNM is honored to participate as a member university supporting their national engineering curriculum utilizing distance education. As Lionel Baldwin, NTU President stated at a recent meeting, NTU has become a "virtual" university.

The following phases are suggested to identify, modify, and develop technical training materials (distance learning, computer-based training, video-based courses, interactive television courses, etc.) to maintain an informed, effective DOE workforce. The process will facilitate selection of the most appropriate ISD technology based on material, audience, budget, and timeframe.

Research/Data Collection

Conduct a needs assessment of receiving locations by utilizing comprehensive survey methods including telephone, facsimile machines, computer, electronic mail, etc. Using the strategy to study the technology most effective for education and training, the material can then be customized to fit the exact needs of the intended audience.

Hardware Inventory and Trainers

Implement the same comprehensive technological survey techniques to assess the equipment inventories at the receive sites, which would be used to determine the methods of receive capability. The hardware would facilitate accessible technologies in five major areas:

a) Originating Site - Broadcast Capabilities, Uplink/Broadcast/Production
b) Receiving Field Offices - Receiving Capabilities/Downlink/Monitors/Phones/Faxes/Computers
c) Training Rooms - Physical Room Arrangements/ISD Equipment Available
d) Individual Work Stations - How many and composition at each facility
e) Trainers - To participate (EM & Field)
  - Background of experience in ISD
  - Organizational reporting issues
  - Time available to participate in new and innovative ISD training opportunities

Audience Evaluation

Evaluate ISD technologies available to all levels and organizations to determine the access and availability to every individual as well as

groups. Establish Advisory Group composed of trainers from field offices, representatives, and contractors to oversee projects. Technologies surveyed will include traditional chalkboards, as well as computer programs, faxes, modems, VCR's, and monitors, CD ROM capability, Internet, electronic mail, two-way interactive computers, and downlink of satellite feed.

Competency Level of Employees

Employees will be evaluated for basic understanding of equipment usage by the following: a) determining employee preference for and use of instruments/media, b) testing on familiarity with equipment, and c) logging usage of various technologies

Competency Level of Trainers

Trainers/facilitators will receive evaluations parallel to those of employees to determine their competency and skill in using the various technologies for general business purposes. Trainers would appropriately demonstrate the same or greater competency and skill as the average employee.

ASSESSMENT OF ISD ACCESS AND FAMILIARITY WITH TECHNOLOGY

After completing the assessments determining general business usage of ISD technologies by employees, basic training would be initiated to achieve a common skill level on all the media identified for use by trainers for ISD technology training. It is essential to consider the preferences of both students and trainers in identifying these courses. After the control group has been established in which the media/technology/instrument of teaching is the tool, a survey of courses currently initialized and developed would be compiled. A small number of target, sample courses (with a mix of student levels) would be established as a pilot project. A survey instrument would be developed to determine the elements that would result in effective training. Various ISD technologies would be introduced to existing courses to discover their effectiveness for participants at various skill levels.

EQUIPMENTAL DESIGN EVALUATION

An experimental design would be developed to include concurrent evaluation of both a control group (the group that takes the course without the new introduction of ISD), and an experimental group (introducing the newer ISD for evaluation).

Qualitative evaluation of training would include (but not limited to): utilizing the new skills in the job (changing the way they are currently doing their work as a result of the training), and basic comfort level of employers with the new ISD training technology.

Quantitative issues evaluated would include the number of employees taking the course in what period of time, the test/evaluation scores, and the basic knowledge tests comparing the results of pre-test and post test. Different learning styles/preferences of various groups include: ethnic background and gender, geographic location, and workers job descriptions/titles relative to ISD preference.

CONCLUSION

In addition to learning about the technology, familiarity with the software packages used is important for those who will be anxious to improve their presentation skills. There are many software packages (Adobe, Persuasion, Microsoft PowerPoint, Gold Disk Astound) that are suitable for multi-media presentations. Training will depend upon the time and expertise available by both the customer and the trainer. Presentations have progressed beyond placing transparencies on an

overhead projector or writing (with one's back turned to the audience) on a chalkboard. Today live and full color video can be incorporated in a presentation, along with computer demonstrations and interactive modes. Exploring the technologies available is important in today's changing society.

Standardizing training materials by converting to ISD formats would assure quality and content of material, and maintain a consistency of message within the customers area. ISD technologies can help students cope with the culture shock of technological change, and assist them in using their existing skills in the new, high-tech environment. Computer-based learning methods are similar in presentation to traditional methods, thus aiding student receptiveness to new technologies. Students can work at their own pace and in their own style with the adaptable ISD technologies, thereby reducing the impact of individual levels of competence. Dissemination of information via telephone, cable, satellite and television should be considered with the audience in mind. A criteria for selection of the technology should include cost, duration (is the technology likely to change quickly), quality and acceptability.

The outcome of this phase approach is projected to concur with existing research literature on ISD technology. However, this approach considers the employees' unique work environment as well as the effect of decentralization of responsibilities, and the types of training necessary and desirable skills. A variety of options could include the combination of several technologies, and the length of the learning process.

Combining the human and the hardware in the system is of the utmost importance. Today the ability for interaction is growing, thus coming closer to the ideal communication situation: one-on-one human communication where the element of direct feedback is present.

The phased approach can facilitate a "mix and match" technology reflecting the resiliency of government and industry in adsorbing and retraining current and new employees. The survey and assessments ensure the appropriateness of the material to the technology and determines the preferred choice of instructors and students. There is no one tool that can solve all problems of information dissemination. It is best to arrive at a degree of competency on an arsenal of tools, to include both the familiar and the unexplored. With timelines, budget, priorities, audience size, necessity, and the availability of technology, clearly defined, ISD technologies can be implemented to help institute necessary change in mission and culture, and while lining up with Secretary O'Leary's goal of saving millions of dollars and maintaining a highly effective and well-trained workforce. ISD technology is a means to such goals.

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IDENTIFICATION OF UNLABELED GAS CYLINDER CONTENTS FOR PROPER WASTE CHARACTERIZATION

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ABSTRACT

The management of waste gas cylinders with unidentified contents is an issue of far reaching significance. Directly involved and potentially at risk are thousands of workers at DOD, DOE, commercial and academic facilities where poor storage practices of waste gas cylinders have continued for decades. It is largely accepted that proper

characterization of the cylinder contents calls for a chemical analysis. Criteria for these analyses have been discussed in Section 5.3.3 (Compressed Gas Cylinder Handling) of EPA Publication 625/R-93/013 (Approaches for the Remediation of Federal Facility Sites Contaminated with Explosive or Radioactive Wastes). The Publication states that examination of the cylinder's exterior (color, markings, etc.) is insufficient for identification of the contents. Rather laboratory techniques such as Infrared spectroscopy (IR), gas chromatography (GC) or mass spectrometry (MS) are required. Though it is widely agreed that spectroscopic analysis of the unknown material is essential, this paper will argue that confident identification of the unknown requires chemical expertise and cylinder expertise. Analyses based solely on GC, MS, IR or any chemical data can lead to incorrect identification of the unknown regardless of the analyst's skill in spectral interpretation. The erroneous conclusion will lead to incorrect waste classification and may cause serious worker injury.

INTRODUCTION

Compressed gas cylinders are routinely used at an enormous number of industrial, commercial, military and research facilities. The great majority of cylinders undergo the intended cycle: test, fill, use, return (recycle, dispose), retest, refill... etc. Cylinders managed this way tend to remain clearly labeled and pose none of the special problems of those with unknown contents. The "unknowns" are usually old, abandoned containers whose labels are missing or have become illegible due to poor storage practices. This includes those left to rust in factory corners and university basements or those waiting for years "to be processed" at gas facilities. Also included are cylinders that have been damaged by fire or natural disasters, and those excavated from chemical "cemeteries" at government remediation sites. But even clearly labeled home-made cylinders or those with modified valves should be considered to have suspicious and potentially dangerous contents, especially if the labels are hand written. Cylinders bearing contradictory signs, such as conflicting labels or the presence of liquid where none should be present must be considered unknown as well.

The dangers associated with gas cylinders arise from the fact that these sturdy containers are used to store materials under high pressure as well as many of the most reactive, and most toxic low pressure compounds (see Table I). Further, it must be recognized that cylinders are not just used for pure gas storage. Cylinders have also been used to package liquified compressed gases, liquids, solids and a great number of commercial, industrial and specialty mixtures. Together with the durability of the package, which generally exceeds the durability of the label and valve, it is clear that old, unmarked cylinders with failing valves may contain any of a large number of hazardous materials. The safe management of such cylinders calls for a three step analysis: 1) a detailed inspection of the cylinder and its valve, 2) a chemical analysis of the cylinder's contents, and 3) an evaluation of everything learned from the first two steps to determine if the clues paint a consistent picture.

CYLINDER EVALUATION

Before laboratory work begins, it is important that the cylinder be examined by a specialist for clues about the intended contents of the cylinder. This requires being able to identify and interpret cylinder markings as well as recognizing the cylinder and valve construction materials - eg. steel, brass, stainless steel, Monel etc... The analyst

must be able to identify the valve outlet's Compressed Gas Association (CGA) designation and recognize whether the valve is the packed or packless type. It must be noted whether a pressure relief device is present and, if so, where it is located and whether it is the plug, disk or valve type. It is important that all of this information be noted, as the following example illustrates: a rusted cylinder with a crumbling label was found at a DOE facility. The external, left hand threaded valve outlet was measured and correctly identified as a CGA 350. This outlet has been designated for use with many flammable gases. Had the cylinder evaluation been concluded at that point, the contents may tentatively have been identified as methane or ethane because the remaining portion of the label seemed to bear the letters "ane." Further inspection of the valve showed that no pressure relief device was present. This additional fact indicated that the cylinder was intended for toxic, flammable gas service. Accordingly, the contents were tentatively identified as silane. Laboratory results confirmed this. Of course, one can not be certain what a cylinder contains until the laboratory analysis is completed. But the knowledge that a cylinder may be leaking silane instead of methane not only simplifies the work, it may also prevent serious worker injury.

SAMPLE COLLECTION AND MANIPULATION

In the Field

The preferred way to collect a sample of an unknown gas is to bring the unidentified cylinder to the lab. Doing so eliminates uncertainties in the following; the original cylinder's pressure, the ambient temperature and pressure, and the state(s) of matter in the original cylinder eg. gas only, gas and liquid... In most cases, though, this is not allowed by the U.S. Department of Transportation. Once transferring becomes necessary, it should be done with careful attention to safety issues and in such a way that the sample is truly representative of the unknown. That means the manifold (the system of valves and tubing) through which the sample is transferred, and the receiving vessel be properly cleaned and evacuated. Likewise, the materials from which the manifold components are made and even the valve lubricants must be judiciously chosen for compatibility. The fact is, many highly reactive materials will be chemically altered by the presence of small amounts of air or water or on contact with so-called non-reactive polymers. The result is that the micro sample that finally reaches the spectrometer may not be the same as the remainder of the unknown.

In the Lab

Gas samples are best handled with a manifold. This stands in contrast to liquids and solids which often are prepared as solutions and delivered to analytical instruments using syringes. The idea is to attach the cylinder to the manifold in a leak tight fashion and deliver the sample to each of the instruments via plumbing - not by walking through the lab with the sample in hand. The manifold should be located in a forced-air fume hood and be equipped with gauges for coarse and fine pressure/vacuum measurement, emergency backup valves and check valves, a finely controlled needle valve, and chemical scrubbers on both the vent and vacuum side. The sampling manifold should be constructed according to the same guidelines as the transfer manifold, with careful thought given to chemical compatibility. A separate manifold should be freshly passivated and used any time the unknown is suspected of being a potent oxidizer eg. fluorine, oxygen difluoride, chlorine trifluoride or other interhalogen

compounds. The reaction of strong oxidizers with most organic compounds (oils, grease, residue from last sample) is generally explosive.

CHEMICAL ANALYSIS

For an analysis to be done correctly, it must be recognized that cylinders may contain gases, liquids, solids, or combinations of materials in different states (see Table I). Further, there may be an equilibrium between the states, such as that with liquid and gaseous ammonia. Or there may be no gas-liquid equilibrium, as with nitrogen and pyrophoric liquids. Or there may be more complex equilibria at work, such as those between liquid phosphine, gaseous hydrogen and gaseous phosphine. Because of these equilibria, it is possible that a cylinder with high pressure may actually be almost empty, whereas one with no pressure may be full. All of this must be considered in order to properly interpret the results.

Table I

Confident identification of the gas in an unlabeled cylinder is best achieved by using two unrelated spectroscopies. The best tools known to the author for this purpose are Infrared (IR) spectroscopy and mass spectrometry. Gas chromatography is certainly an important analytical technique but it is by no means the starting point of unknown gas analysis. The principle reason is that GC data provide quantitative, not qualitative information. Analysis of the liquid or solid may also be done spectroscopically or may require wet chemistry methods.

Infrared Spectroscopy

A Fourier transform infrared (FTIR) spectrometer, equipped with a robust gas cell, is ideal for unknown gas analysis for several reasons. To begin with, the vast majority of materials packaged in cylinders are IR active . Further, the computer, which is present in all FT-type instruments, makes it simple to scan a sample -say 30 times, signal average, display, and store the result to computer memory in less than five minutes. The same computer can be used to quickly compare the sample spectrum to spectral libraries in search of a best match. To be done effectively, the libraries should contain 10,000 gas phase spectra and an additional 10,000 or so condensed phase spectra. Such libraries are available commercially.

Once the library search results are presented, however, the analyst must decide whether the results are believable. For non-reactive, pure compounds spectral interpretation is straightforward. The job is more difficult with non-absorbing or weakly absorbing materials and with mixtures. "Mixtures" includes compounds deliberately mixed by the chemical supplier, materials deliberately or accidentally mixed by the previous user, and the reaction of the sample with manifold residues. Very little deviation from a library spectrum is all that is required to make the search results confusing. Clearly the analyst must be skilled in spectroscopic interpretation.

There are several shortcomings with the use of IR for unknown gas analysis. To begin with, not all materials are visible in the IR. The diatomics, hydrogen, nitrogen, oxygen, fluorine, chlorine and bromine are invisible. Likewise, the Noble gases, helium, neon, argon, krypton, and xenon are invisible. Further, compounds that are related structurally or chemically, like propane and butane, are quite difficult to distinguish. Next, the absorption strength of all compounds is not equal. For example, dichlorodifluoromethane (a refrigerant) absorbs very strongly at low concentration. A much higher concentration of hydrogen cyanide (highly

toxic) is required to observe the same size signal. Therefore, mixtures of compounds with differing absorption sensitivities will be disproportionately represented in the spectrum. The only way to know the true concentrations is to compare the sample spectrum to a reference mixture spectrum. It follows that if a weakly absorbing gas is present in a mixture at low concentration, it may be missed altogether.

Mass Spectrometry

As mentioned above, it is important that a second, unrelated analytical technique be included before reaching a final conclusion. Mass spectrometry (MS) is an excellent candidate for this position for three reasons. 1) MS is an extremely general technique - useful for the detection of hundreds of thousands of compounds, 2) it is not "blind" to any of the gases listed in the previous paragraph and 3) as required, the principles of operation are completely unrelated to those of IR spectroscopy.

Standard benchtop mass specs equipped with electron multiplier detectors are sufficient for the job. Likewise, the so-called "mass spec detectors," basic, computer operated mass specs that are marketed as detectors for gas chromatographs, are also quite adequate, but modifications must be made to accommodate direct sample introduction. As with the IR, multiple scans are easily averaged and the result compared to commercial libraries. One of the more common libraries has roughly 70,000 spectra. Once again, the computer search algorithm will assign a low match quality if the sample and reference spectra do not match very closely. Analysis of a cylinder of tetrafluorosilane (SiF_4), for example, contaminated with a little moist air will give a single (confusing) mass spectrum that combines all the features expected for SiF_4, in addition to those of hydrogen fluoride, oxygen, nitrogen, carbon dioxide, water and the minor components of air. It is essential, therefore, that the analyst be skilled in spectroscopic interpretation.

RECONCILIATION OF CYLINDER EVALUATION AND CHEMICAL ANALYSIS

After interpreting the chemical data, it is imperative to review the cylinder evaluation to determine if the results are consistent and if there is a need for chemical analysis of the condensed phase. There are innumerable examples of how failure to properly reconcile all observations can lead to an incorrect conclusion and disaster. Here are two:

Example 1 A low pressure cylinder containing liquid is too large to be transported as an unknown (USDOT guidelines). A gas phase sample is collected and brought to the lab for analysis. The results are reported as a few percent ethane in nitrogen at <5 psig. The incorrect conclusion is that the sample and original cylinder contain ethane in nitrogen. This is either a calibration mixture or accidental contamination. A disposal or recycling plan should be developed according to the flammability limits of ethane.

A careful cylinder evaluation would reveal the following inconsistencies: neither ethanenor nitrogen tend to be packaged in low pressure cylinders. The cylinder contains liquid and has a 5 psig head pressure, yet the equilibrium vapor pressure above liquid ethane is 543 psig @ 70oF. The correct response is that the gas phase sample is not representative of the liquid. This cylinder may contain a pyrophoric material such as an ethyl aluminum compound. Suppliers typically ship these liquids in low pressure cylinders slightly pressurized with nitrogen. It is common to find a little methane gas above trimethyl compounds, ethane gas above

triethyl compounds, etc... Opening such a cylinder in the presence of air will cause a spectacular fire, spontaneously. A chemical analysis must be done directly on the liquid phase.

Example 2 A small, 30 year old cylinder with a CGA 320 outlet is known to contain liquid. Laboratory analysis clearly indicates the cylinder contains methyl chloride. This is determined by two factors: 1) the close match of the unknown FTIR spectrum with reference spectra (Fig. 1) and 2) excellent agreement of the measured vapor pressure (59 psig @ 70oF) with the literature value. The hasty conclusion would be the following: based on the certainty of the spectral results and the match with the expected vapor pressure, the cylinder contains compressed, liquified methyl chloride. Handle accordingly.

An experienced cylinder analyst would note that CGA 510 or 660 outlets are typically used for methyl chloride. The absence of the prescribed outlet for this compound may indicate that the cylinder contains an additional low pressure material not observed in the gas phase. Pesticide manufacturers often included methyl chloride as a propellant for the "active" ingredients. The mixture was packaged in cylinders often fitted with CGA 320 outlets. For example, 5-10% mixtures of highly toxic organo-phosphates in methyl chloride have been marketed under the trade names Phosdrin or Nifos T. These compounds have LD50s of 1-2 milligrams per kilogram of body weight. A proper analysis requires checking for a second component. To do so, the cylinder should be inverted in a laboratory fume hood and a few drops of liquid carefully delivered. Failure of the entire sample to evaporate instantly indicates a material other than methyl chloride is present. The FTIR spectrum of the residual liquid (Fig. 2) verifies the presence of the toxic material. Clearly even a skilled chemist who lacks cylinder experience is likely to conclude the analysis with the identification of methyl chloride. If the precautions taken during the disposal of this improperly identified cylinder are suitable only for methyl chloride, the workers are likely to be poisoned.

Fig. 1

Fig. 2

These are some examples of how a disaster can arise from incorrect conclusions based upon sound scientific information. It is clearly seen then, that in order to properly and safely manage cylinders with unidentified contents, it is essential that a thorough chemical analysis go hand in hand with a thorough cylinder analysis.

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PROACTIVE PUBLIC INVOLVEMENT STRATEGIES FOR THE ENVIRONMENTAL RESTORATION PROJECT AT LOS ALAMOS NATIONAL LABORATORY

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ABSTRACT

The Resource Conservation and Recovery Act mandates formal public involvement at the corrective measures stage of any environmental restoration activity. At Los Alamos National Laboratory, the Environmental Restoration Project has exceeded legislative requirements in its approach to public involvement.

However, formal public meetings are not particularly successful, especially in an area of substantial economic and racial diversity. In the past, activists have monopolized the Laboratory's formal public meetings.

To ensure that a broad cross-section of the public participates, the Environmental Restoration Project at Los Alamos has developed an approach that involves a much broader variety of stakeholders in making recommendations on various remediation activities. This approach consists of attending community meetings and familiarizing different constituencies with the purpose and goals of the Environmental Restoration Project. Attending community meetings has helped Los Alamos and our neighbors build a more positive relationship based on mutual respect and trust.

As a direct result of this different approach, community representatives have volunteered to participate in a focus group that will work together on issues of concern and interest.

PROACTIVE PUBLIC INVOLVEMENT STRATEGIES FOR THE ENVIRONMENTAL RESTORATION PROJECT AT LOS ALAMOS NATIONAL LABORATORY

Located in the northern mountains of New Mexico, approximately 32 miles northwest of Santa Fe, Los Alamos National Laboratory occupies a unique position in the Department of Energy's nuclear complex. It was at Los Alamos during the "Project Y" years that scientists from Europe and the United States worked together to develop and test the first nuclear weapon, thus heralding the beginning of the "nuclear age."

Development of the first nuclear weapon also initiated the problem of addressing the radioactive and hazardous waste generated by this project. Although the dangers of acute radiation were known at that time, no one knew with any certainty the environmental problems that could be created if waste products were buried for extended periods of time. The long-term health and environmental effects of liquid waste drained into the canyons also was not known. The canyons that cross the Los Alamos National Laboratory site all flow to the Rio Grande.

Los Alamos National Laboratory and Los Alamos county are surrounded by the communities of Santa Fe, Espaola, and several Indian Pueblos, the closest of which are San Ildefonso Pueblo, Santa Clara Pueblo, San Juan Pueblo, and Cochiti Pueblo all located adjacent to the Rio Grande.

To address past releases from research and development activities at the Laboratory, the Department of Energy in 1989 established the Environmental Restoration Project (ER Project). The ER Project established the following goals:

to protect human health and the environment from exposure to releases of hazardous and mixed wastes from historical treatment, storage, and disposal practices at the Laboratory;

to meet the environmental cleanup requirements of the Laboratory's permit to operate under the Resource, Conservation and Recovery Act (RCRA), specifically Module VIII, which governs ER Project activities (Module VIII was issued in May 1990 and modified in May 1994); and

to perform timely and cost-effective cleanup activities.

Module VIII requires the Laboratory to establish a community relations plan for the ER Project. The following elements were required as part of the community relations plan:

1. establish an active mailing list;
2. conduct informal public meetings;

3. prepare fact sheets, news releases, work plans, final reports, and quarterly progress reports;
4. establish a public information repository and reading room;
5. place updated materials in the information repository and reading room;
6. conduct public tours and briefings, and solicit public concerns;
7. prepare quarterly technical progress reports for the Administrative Authority; and
8. establish procedures for immediate notification of San Ildefonso Pueblo or other affected parties in case of a newly discovered off-site release that could have a potential impact.

In addition to the requirements of the Hazardous and Solid Waste Amendments (HSWA) permit, the ER Project follows the requirements of the Comprehensive Environmental Response, Compensation and Liability Act (Superfund) by establishing an administrative record for the ER Project. The Laboratory also follows the public involvement requirements under Superfund, although it is not a Superfund site.

The first public information meeting for the ER Project was held in May 1992 at three locations: Santa Fe, Espaola, and Los Alamos. The format of these meetings was informal: It consisted of an open house with project exhibits and staff available to answer any questions. A limited amount of time was available for a general briefing and a question and answer period. An average of three meetings each year were held until March 1994. These meetings were different from the formal public hearings mandated by the HSWA permit. Although initial attendance at these public meetings was good, attendance decreased significantly during 1994. The public meetings generally attracted representatives of special interest groups whose agendas did not necessarily include environmental cleanup issues. The ER Project felt that in holding these public meetings it did not really reach its intended audience. As a result, the ER Project decided that this approach was not cost effective and discontinued these meetings; it also decided to find an approach to reach a broader section of the interested public, including state regulators, local government officials, community service organizations, Indian Pueblos, and ordinary citizens.

In its first effort, the ER Project conducted a day-long brainstorming session for the northern New Mexico public. ER Project staff and the public addressed their information needs, how the public could provide meaningful comments on decisions, and how ER Project staff and the public could interact more effectively in the future.

With this information, the ER Project then formed a working group consisting of ER Project staff, DOE representatives, and interested Laboratory organizations. This working group designed an approach that consisted of informal conversations based on a standard set of questions. The working group felt that this approach, rather than conducting formal surveys, would provide the ER Project with better qualitative information and would better serve the public involvement process in the complex, multicultural environment of northern New Mexico. The working group identified a cross section of the northern New Mexico population and organizations that represented the diversity in northern New Mexico. Interviewees included elected officials, agency staff, ethnic and peace organizations, the media, educators, representatives of the business community, and other interested citizens. Approximately 42 individuals

were identified; these individuals were interviewed between June and July of 1994.

A team consisting of an experienced interviewer and a technical staff member from the ER Project conducted the interviews. Appointments were scheduled at the interviewees' choice of location and time, which necessitated travel as far south as Albuquerque and as far north as Taos. Including technical staff as part of the interviewing team enriched the information received from the interviewees and created positive feelings about the ER Project and Laboratory staff.

The interviews provided information about the Laboratory's impact on the northern New Mexico communities, assessed the interviewees' awareness and knowledge about the ER Project, identified issues that were of concern to them, and explored their ideas about how to encourage community participation in the public involvement components of the project. Most importantly, the interviews broadened the dialogue between ER Project staff and representatives of organizations or communities.

Community concerns focused on the Laboratory's profound economic and social impact on northern New Mexico. Whether supportive or antagonistic toward the Laboratory, interviewees agreed that the Laboratory's presence in northern New Mexico is of benefit to many of the surrounding communities. It is very important to the public that the Laboratory and its employees be sensitive to the indigenous people and their cultures.

Almost all interviewees addressed trust and honesty issues. The culture of secrecy that has been associated with the nuclear complex has been detrimental to any trust the neighboring communities and individuals representing them have in the Laboratory and the Department of Energy.

Health and environmental concern also rated high in the interviews. Contamination of aquifers, surface waters, and air and soil is of great concern to our neighboring communities. The interviewees voiced their concerns about Laboratory operations affecting their families' health through possible contamination of their food supplies. Many people in the communities surrounding the Laboratory have vegetable gardens and raise their own livestock or hunt deer and elk for their food supply.

In talking about these issues with the interviewees, their feedback almost unanimously showed that they

- much appreciated the personalized interview approach;
- would welcome presentations at meetings of community service organizations and local government agencies;
- trust their local community leaders and encourage the ER Project to foster closer relationships with them; and
- strongly favor one-on-one contacts with the Laboratory technical staff as a means of exchanging information.

Suggestions from the interviewees to improve public involvement activities included

- working with existing groups and identifying community and business leaders in the community. Public meetings are good but not effective in reaching the larger community. "Why not come to us instead of having us come to you?"
- emphasizing to the leaders in the community the need for their communities to participate in the Laboratory's environmental restoration and waste management activities.
- limiting public meetings. If public meetings are necessary, set a fixed time limit; involve the public in setting the agenda; establish boundaries of discussion before the meeting and request that people stick

to selected topic; value the opinions of people who attend meetings and ask people to submit questions before meetings.

designing tours that specifically target community leaders and school children. Tours would include cleanup sites (both undergoing cleanup and restored sites) and science (walking) tours, such as geological tours. Interviewees suggested that people in northern New Mexico appreciate and actively enjoy their environment. It would be a useful tool to use this appreciation to generate interest in this and other public involvement processes.

communicating scientific information. It is not easy to present to the public scientific information in understandable language. The interviewees asked that materials and presentations be in "plain language." Interesting suggestions for improvement included using local science teachers to help present scientific information, using Laboratory employees as conduits for conveying information to the public, and using the storytelling traditions of the indigenous people to communicate environmental restoration information.

As a result of the information received during the interviews, ER Project staff made an effort to visit different organizations in the northern New Mexico communities. ER Project staff visited with service organizations such as the Kiwanis and Rotary Clubs, as well as culturally diverse organizations such as Cultura del Norte and the Espaola Women's Library Club. During these visits, ER Project staff asked for volunteers to help form a committee to address issues of particular concern to the ER Project, such as site ranking, cleanup methods, and budget reductions. A number of people were very receptive to working on a committee with ER Project staff (fourteen people volunteered).

The ER Project is presently establishing a committee consisting of volunteers from the northern New Mexico community; this committee will help the ER Project make recommendations on a variety of environmental issues. These volunteers will familiarize themselves with the Laboratory, the ER Project, and its various potential release sites. The committee will have a representative voice in how these issues will be addressed.

Taking into consideration both positive and negative comments, three factors have facilitated public involvement in environmental restoration at the Laboratory. First, because the communities generally believe that the Laboratory is important to the economic well-being of northern New Mexico, they are willing to work toward solutions with the Laboratory. Second, there is considerable trust in the integrity and ability of the individuals working in the ER Project. And third, the communities believe the Laboratory has the technological ability to address cleanup issues.

Because the ER Project has changed its public involvement activities to accommodate the desires of the public, the Laboratory has gained more credibility with a far broader section of the public and not just public interest groups. In addition, ER Project staff have established many personal contacts with individuals, thereby resulting in a group of volunteers willing to spend the time and effort to work with the ER Project on a variety of environmental issues.

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APPLICATION OF RISK-BASED CORRECTIVE ACTION TECHNIQUES FOR ENVIRONMENTAL RESTORATION PROJECTS AT DEPARTMENT OF ENERGY (DOE) REMEDIATION SITES

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ABSTRACT

The nuclear technology community has struggled for years with the concept of "how clean is clean?", and how to approach the decontamination, decommissioning, cleanup, and release to the public of radiologically contaminated facilities and lands formerly used to support the national defense mission. Astronomical cost estimates for remediation efforts have refocused the evaluation of these programs to consider the issue of "how clean is clean enough"? Compliance to established performance-based environmental standards presents an undue burden on the national cleanup budget, since it is not practical or necessary that properties be restored to the same standard (dependant upon projected future use). An alternative to compliance with predetermined regulatory standards is the implementation and utilization of a site-specific risk-based corrective action (RBCA) process.

The concept of utilization of a risk-based approach to establishment of remediation goals has been developed primarily for RCRA sites contaminated with petrochemical contamination (e.g., ASTM Emergency Standard ES 38-94). Parsons has been instrumental in assisting a number of DOE sites in developing a basis for utilization of an analogous approach for cleanup of radiological contamination. In the event of soils and groundwater cleanup, this approach can result in progressive remediation efforts that maintain realistic health and environmental protection goals. This process can be a functional element of DOE programs involving a streamlined approach for environmental restoration (SAFER).

Risk-based decision making is used to help determine the appropriate type and level of corrective action. Based on the principles of protection of human health and the environment commensurate with actual risks (rather than strict compliance with assigned performance-based goals), and expedites corrective action in a cost-effective manner. The basis for the Parsons approach to RBCA programs integrates the framework established by ASTM, including the provisional standards for cleanup developed by the Voluntary Cleanup Task Group.

There are a number of computer-based tools and models that may be utilized to assist in defining the requirements for RBCAs at DOE sites. Computer codes such as RESRAD and GENII, modifications in hydrological and geohydrological models, and modeling computer systems such as ERMA/GIS have been utilized by Parsons and have been instrumental in preparing defensible RBCAs.

The concept of RBCA for radiologically contaminated sites could be very important in view of recent rulemaking efforts within the nuclear technology community. For example, rulemaking efforts by the Nuclear Regulatory Commission (e.g., 10 CFR 20.1400 and NUREG-1500) are being promulgated to define free release criteria for nuclear facilities and related properties. These rulemaking efforts are being conducted to provide codification of radiological criteria for site cleanup, which will allow the NRC to more efficiently carry out the functions of public health protection and environmental protection by more efficient use of resources. The intent is to allow consistent application of standards across all types of licenses and to provide a predictable basis for cleanup activities. The implementation of a uniform RBCA process to support attendant risks for site remediation would be a key step in moving forward with these goals.

BACKGROUND AND OVERVIEW

With changing congressional directives and limited funding to be allocated for environmental, safety, and health protection programs, it is more imperative that ever that governmental and industrial organizations optimize remediation and prevention budgets. In order to address the most pressing concerns, it is necessary to develop a method of ranking and/or prioritizing actions such as site remediation. A risk-based approach to site remediation is a means of ensuring that the remediation problems that are of the highest concern receive the highest priority for action.

The impetus for environmental and human health protection risk assessments may be traced to the enactment of the National Environmental Policy Act (NEPA) of 1969, which mandated the requirements for analysis of environmental impacts for major federal actions affecting the quality of the human environment. Risk analysis has developed into a major tool for identifying and quantifying hazards and the associated consequences of federal and industrial activities.

The proliferation of federal laws and regulations to protect public health and safety and to provide for the protection of the environment has generated rapid growth in risk management activities in numerous and diverse fields of applications, such as occupational safety, toxicology, industrial hygiene, nuclear safety, environmental impact assessment, engineering/reliability studies, weather prediction, epidemiology, and the social and behavioral sciences.

The science of risk analysis is a discipline that has received a great deal of attention in recent years. The body of science associated with risk has been subdivided into many interrelated (and sometimes interchangeable) components such as risk analysis, risk assessment, risk evaluation, and risk management; however, the general topics of risk analysis and management may be used to address all of the subdivisions associated with risk measurement and management. All of these concepts are integral to the development of technical bases for RBCAs.

THE ASTM STANDARD FOR RBCA

An ASTM standard has been established for petroleum release sites that can be adapted as a basis for RBCA at DOE locations. ASTM ES 38-94, entitled "Emergency Standard Guide for Risk-Based Corrective Action Applied at Petroleum Sites", promotes a three-tiered approach to risk and exposure assessment, where successive tiers require more rigorous analytical treatment and more detailed data collection and validation. This approach involves:

Tier 1 -- Qualitative risk assessment that is based on general site assessment information. The data at this level of quality should identify obvious environmental impacts (if any), potentially affected and sensitive receptors, and significant exposure pathways. Information should be sufficient to categorize sites and to determine acceptable time periods for implementation and completion of necessary corrective actions.

Tier 2 -- Requires collection and validation of more site-specific data in order to determine appropriate risk-based actions. At this higher level of analysis, the reasonable maximum impacts of the contaminants on a site are evaluated by used o site-specific characterization and monitoring data, conservative projections of expected contaminant levels after treatment and potential plume migration, and reasonable maximum exposure scenarios. This information is used to set conservative

corrective action objectives that are protective of human health and the environment.

Tier 3 -- Focuses entirely on assessment of site-specific conditions and requirements. At this level of analysis, more sophistical statistical/mathematical evaluations of transport, uptake, and fate phenomena may be utilized to generate a range of possible exposures and related risks. Site-specific risk assessment models may be developed.

The goal of the tiered approach is to achieve the appropriate level of detail and rigor of analysis that will ensure protection of human health and the environment. The advantage of this technique is that, in moving from lower to higher tiers, corrective actions can be rendered more cost-effective and efficient since the conservative assumptions of lower tiers are replaced with more accurate and realistic analyses that involve site-specific assumptions. The ASTM three-tiered approach can be adapted easily to accommodate the multitude of DOE site remediation problems, and can be evolved by each site to manage corrective actions and to facilitate the decision-making process.

BASES FOR RISK ASSESSMENT

Risk assessment is the body of technology that utilizes qualitative and quantitative analysis as methods of defining the risks associated with actions or activities, and planning for the consequences of these risks. The risk analysis can serve a variety of risk management purposes; for example, it can provide input to decision makers to:

Determine safety, environmental, and health concerns associated with specific activities and substances (e.g., hazardous waste disposal, radiological exposure, or the use of industrial chemicals); Compare new and existing technologies or determine the effectiveness of controls and mitigation techniques designed to reduce risks;

Select sites for potentially hazardous facilities and/or operations; or

Establish management priorities, such as which of several activities should be considered first for regulatory or corrective action.

As such, the risk analysis and management process can be an important factor in the routine management of operations or activities, in the establishment of criteria for abnormal or accident situations, and in the estimation of costs and related benefits for a proposed action.

METHODS OF RISK ANALYSIS

Risk analyses encompass all aspects of a safety/risk assessment, including accident scenario formulation, initiating event identification, the probabilistic risk of failures of components and systems that are intended to contain hazardous, explosive, nuclear, or chemical materials, the risk associated with fires in areas containing high explosives, and the risk to facilities from seismic events and tornados. For the formulation of RBCAs for site remediation, the predominant concerns are release, mobilization, transport, exposure, uptake, and ultimate fate of hazardous, toxic, and/or radiological contaminants.

Risk analysis/assessment is used to estimate future losses and the effectiveness of additional controls in order to provide management with information to make sound decisions regarding hazards and dangers associated with risks. Risk analysis can provide or permit:

Probability estimates of large or catastrophic accidents/events;

Summation of loss estimates (e.g., actuarial predictions) to provide a more complete risk estimate;

Cost-effective safety programs, by concentrating on high risk areas;

Optimization of the combined cost of safety programs and the cost of accidents which are present at a given level of control;

Evaluation of the effects of codes, standards, orders, directives, and regulations;

Consideration of various types of risk on a consistent basis.

The science of risk analyses encompasses analytical applications (e.g., design basis accident analysis and evaluations pertinent to applicable modeling codes). These analytical applications are to be complemented by reliability analysis and risk assessment/analysis methodologies applied to a range of release/accident conditions. Risk analyses are used in safety analyses to provide insight into those aspect of design and operational characteristics that are vulnerable to potential accident (particularly risk-dominant sequences).

Qualitative Risk Analysis

Qualitative risk analysis is used when the information available on a process/systems under consideration is inadequate to provide a numerical basis for quantitative risk calculation, or when the requirements for risk analysis are to provide an idea of the types or risks involved rather than the quantities of the related hazards. In many cases qualitative analysis is used as a precursor (the first step) to a quantitative evaluation.

If the qualitative analysis indicates that the nature of the risk is not of a type or source that would generate unacceptable accident consequences, further analysis may not be required. In this manner the qualitative risk analysis may serve as a screening method for risk management programs.

Qualitative risk analyses would be applied most often to Tier 1 RBCAs, but may apply to more simple projects at the Tier 2 level.

Quantitative Risk Analysis

Quantitative risk analysis methods are utilized when the assignment requires that the risks associated with a remedial action (or other process under scrutiny) are to be numerically specified based on information available or that must be gathered. Utilizing experienced safety analysts and sophisticated computer codes such as HAZARD-I, IRRAS, and IMPORTANCE, the knowledgeable risk analyst can reduce laborious numerical manipulations to efficient, workable tasks.

The approach to the quantification of risk is determined by the desired results of the evaluation and the information that is available. There are a wide spectrum of analytical techniques; types of basic quantitative risk analysis and projection methods include:

Gaussian Normal Projection
Subjective Risk Analysis
Log-Log Distribution of Frequency/Severity
Survey Methods
Log-Normal Distribution of Accident Data
Insurance Risk Analysis
Extreme Value Projection
Life Shortening Effects Time-Loss Analysis
Trend Analysis
Actuarial Risk Assessment

Quantitative risk analysis involves the quantification of the expected or probable loss per unit time or unit of activity and is equal to the probability of loss multiplied by the magnitude of the loss. For any operation, the risk is the sum of the individual risks for each potential

loss. Since there may be an infinite number of both probabilities and consequences for a given risk/accident scenario, an accurate quantification of risk requires consideration of the entire accident cost-frequency spectrum. In quantitative terms, risk is expressed in values ranging from zero (representing the certainty that no harm/accident will occur) to one (representing the certainty that harm/accident will occur). As an example, a risk expression of 10-6 or E-6 (as expressed in the U.S. Environmental Protection Agency IRIS database) equates to a risk of one chance in one million of harm or accident.

Quantitative risk assessment methods for complex site remediation projects might be most aptly applied at the Tier 2 or Tier 3 levels of RBCA.

Probabilistic Risk Assessment

Probabilistic risk assessment (PRA) techniques represent the most sophisticated types of hazard and risk analysis. These analysis methods have been implemented and verified by the commercial nuclear industry for complex accident scenarios involving release of nuclear materials. These techniques must meet extremely stringent requirements for the evaluation of nuclear and non-nuclear facilities compliant with U.S. Nuclear Regulatory Commission requirements. The basic PRA process involves the development of a set of possible accident sequences and the determination of their outcome. In support of this process, sets of models are developed and analyzed.

The development of sequences for the PRA generally can be divided into two sets of models: those relating to operating systems (such as waste management facilities or contamination control systems) and those relating to the containment of these systems. Models for these analyses usually consist of event trees, which depict initiating events and combinations of system successes and failures, and fault trees, which depict ways in which the system failures represented in the event tree can occur. These models are analyzed to assess the frequency of each release or accident sequence. For very complex analyses, the fault and event tree methods may be combined for both a `top-down' and `bottoms-up' analysis. The CAFTA and IRRAS computer codes typically are used to develop and evaluate event/fault tree models.

The containment models represent the events occurring after the accident but before the release of radioactive or hazardous material from containment. These models cover the physical processes induced in the containment by each accident sequence as well as the transport and deposition of materials released within containment. The analysis examines the response of the containment to these processes, including possible failure modes, and evaluates the releases of hazardous materials to the environment.

The outcome of the accident in terms of public health effects and economic losses may be assessed by the use of environmental transport and consequence models. These models use site-specific meteorological, hydrologic, topographic, and geologic data to assess the transport of hazardous materials from the site. Local demographic and health effects models are used to calculate the consequences to the surrounding population.

An integral part of the approach to PRA is uncertainty analysis. This involves not only uncertainties in the data but also uncertainties arising from modeling assumptions. The results of the risk assessments

are analyzed and interpreted to identify the plant/facility features that are the most significant contributors to risk. Numerical manipulation and prediction models such as CRYSTALBALL and APIDSS may be used to manage intricate calculations.

Throughout the analysis, it is important to use realistic assumptions and criteria. When information is lacking or controversy exists, it may be necessary to introduce conservatism or evaluate bounds. The goal for a PRA is to produce as realistic an analysis as possible.

Although PRAs typically have been conducted for nuclear safety evaluations, recent efforts have been established to apply this methodology to environmental restoration actions. A standard guide for probabilistic risk assessment is being developed by the ASTM Committee E-47 on Biological Effects and Environmental Fate. The guide will be prepared to assist risk analysts by providing an approach for developing and using probabilistic estimates of risks due to exposure to chemical and non-chemical stressors at contaminated sites. As stated in this guidance, PRAs use simulation techniques to generate probable ranges of risks and their likelihood of occurrence, providing more information to the analyst than standard point estimates of risk.

PRAs would be used most effectively in the Tier 3 level of RBCAs for complex sites involving multiple contaminants and multiple release pathways.

ENVIRONMENTAL RISK

When applying the RBCA method to a DOE site remediation project, it becomes important to assess (in an integral manner) the risks to air, land, water resources, and fish and wildlife resources. For this reason, analyses of environmental effects may be conducted in support of environmental risk analyses. Assessments of effects to the biota in the environment require information/data such as:

- Concentrations of specific radionuclides;
- Probable sources and the pathways or the movement of the radionuclides;
- Potential target species which may be harmed; and
- Toxicity/dose data for the pertinent species.

In some instances standards may be established that can provide the basis for evaluation of effects to the environment. However, for certain radionuclides, specific environmental standards may not exist. In these cases it is necessary to define evaluation criteria from various sources, such as toxicity or health physics data. The risk analyst may utilize computer codes such as GENII-S, EXREM-III, PAVAN, AIRDOS, and MESODIF to analyze the activities and pathways of radionuclides in the environment.

Human/environmental risk assessments for RBCAs at DOE sites are conducted in accordance with applicable EPA requirements and DOE Orders. Under the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA), risk assessments are to be conducted based on the requirements defined in EPA/540/1-89/002, "Risk Assessment Guidance for Superfund, Volume 1, Human Health Evaluation Manual Part A," and EPA/540/1-89/001, "Risk Assessment for Superfund, Volume 2, Environmental Evaluation Manual (Interim Final)." DOE Order 5400.4 specifies that human/environmental risk assessments be conducted in accordance with the CERCLA requirements. DOE Order 5400.5 entitled "Radiation Protection of the Public and Environment" is used to establish allowable risk levels from radionuclide contaminants.

The typical approach to human health and environmental risk assessments is twofold, and requires:

Establishment of a baseline risk
Assessment of future risks based on predictive modeling.

The baseline assessment is designed to evaluate potential threats to human health and the environment. The analysis requires an understanding of the nature of chemical and radionuclide release from the site and method of transport, the potential pathways for human and wildlife exposure, and a quantification of the potential threat from the exposure. In general, there are four steps are performed to determine the baseline risks: 1) data evaluation; 2) exposure assessment; 3) toxicity assessment; and 4) risk characterization

The first step in the baseline evaluation is to collect analytical data from relevant media (air, surface water, groundwater, soil) and to evaluate its suitability for use in a risk assessment. Tentatively Identified Compounds are identified and evaluated to determine those constituents that are potential threats to human health and the environment.

The second step identifies potential human and environmental receptors and exposure points associated with the site. The potential pathways for exposure are determined. These are called exposure scenarios, that identify how the receptor becomes exposed (eating, drinking, breathing, dermal exposure to contaminants, etc.) Chemical releases from the site are estimated and environmental fate and transport are modeled as required to estimate exposure levels via air, groundwater, surface water, and soils. Chemical intakes are then estimated from the exposure scenarios. The third step involves the performance of a toxicity assessment, hazard identification, and dose-response assessment. Reference doses and slope factors are identified for systemic toxicants and carcinogens, respectively. This information is obtained from EPA approved data bases such as the IRIS or the HEAST. Toxicity values for specific constituents may also be identified in the scientific literature.

The final step compares calculated intakes to acceptable intakes for noncarcinogens, and compares calculated risks to target risks for potential carcinogens. The comparisons determine whether the site or facility presents an adverse human health or environmental risk.

Future risks are assessed by utilizing computer models which predict future chemical and radionuclide concentrations that can be evaluated through the toxicity assessment and risk characterization. Analysts may employ ground water flow and transport models such as the U.S. Geological Society approved method of characteristics (MOC) model to predict the flow and concentrations of contaminants in ground water. Models such as MODFLOW, MODPATH, QUAL2E, MT3D, SWIFT, PTC, QUICKFLOW or many others may be used to model chemical transport in streams and rivers. VADOSE may be utilized to model the subsurface unsaturated zone.

A number of models are useful for calculating radiological site contamination, doses, releases, dispersion, and effects; these models are especially useful when evaluating radioactively-contaminated DOE sites. Some of the most commonly-used models include RESRAD, SKYSHINE (and MICROSKYSHINE), PAVAN, GENII-S, PART 61, AIRDOS (and MICROAIRDOS), HOTSPOT, EXREM, and ALLDOS.

SITE CLASSIFICATION

Once the analytical portion of an RBCA has been performed, it is necessary to classify the site according to the required actions. A typical site classification scheme may involve the following four levels,

based on the calculated threat to human health, safety or sensitive environmental receptors:

Class 1 -- Immediate threat
Class 2 -- Short-term (0-2 years) threat
Class 3 -- Long-term (>2 years) threat
Class 4 -- No demonstratable long-term threat

These classes may be utilized in general for all types of human health and environmental threats, including combined and cumulative effects of radiological and hazardous contaminants.

SUMMARY

The application of RBCA techniques are becoming increasingly important to DOE environmental professionals. The complex task of estimating specific risk and cleanup levels requires the estimation of a number of input parameters, including those which are:

Media-specific (e.g., hydraulic conductivity, radionuclide content, wind speed, etc.);

Contaminant specific (e.g., decay rate, toxicity, mobility, etc.);

Receptor-specific (e.g., body weight, metabolism, ingestion rate, etc.); and

Landuse-specific (e.g., restricted use, unrestricted use, etc.).

These inputs are typically variable and uncertain; they are best represented by probability distributions. In applying a graded, tiered approach to RBCA, the risk analyst can apply uncertainty propagation techniques to quantify the uncertainty in the estimated risk, and utilize this information to prioritize risks, optimize usage of limited funding, and utilize information to make effective decisions.

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SITING AN ONSITE WASTE MANAGEMENT
FACILITY AT ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE

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ABSTRACT

The Rocky Flats Environmental Technology Site (Site) has initiated a project to permit, design, construct and operate an onsite Waste Management Facility (WMF) with a capacity to receive 100,000 to 400,000 cubic yards of low-level radioactive and/or hazardous waste. The function of the WMF is to provide a repository for the remediation waste generated from Environmental Restoration risk reduction activities and Deactivating and Decommissioning activities at the Site. The WMF would potentially receive remedial waste from closure activities starting in calendar year 1996. Planned remedial actions require that a viable remediation waste management facility be operational prior to any removal actions, to eliminate additional interim waste storage concerns.

The objective of the WMF study is to select the best site and design alternative for the management of remediation waste. After viable design alternatives were developed, they were evaluated using a three-phased screening process. The first phase of the project evaluated onsite vs offsite options. The second phase evaluated onsite locations based on specific siting criteria and the third phase involved a comparison of onsite design alternatives.

This presentation focuses on the second phase of the study, the onsite selection process. In each of the first two phases, the best alternative was selected and used as a basis for the next phase. Selection criteria reflected the project objectives, siting and Corrective Action Management Unit (CAMU) criteria as well as the Interim Measure/Interim Remedial Action (IM/IRA) process. Consideration was also given to the National Environmental Policy Act (NEPA) throughout the decision process. Based on the three phases of the evaluation, the best approach was determined to be an above grade Concrete Lined Cell (CLC) designated as a CAMU and located in the eastern portion of the Protective Area near Operable Unit (OU) 4, the Solar Ponds. The CAMU cell will be designed with a geosynthetic and clay liner system which will comply with the Resource Conservation and Recovery Act (RCRA) Subtitle "C" requirements as defined in Code of Colorado Regulations (CCR) 6CCR 1007-3, Part 264 and 6 CCR 1007-2, Part 2. A leachate collection system will provide further protection between the two liners. Upon closure of the cell, an impermeable cap will be placed on the CAMU and the site will undergo post closure monitoring.

INTRODUCTION

The necessity to reduce overall risk at the Rocky Flats Environmental Technology Site (Site) dictates the need to design and construct an onsite Waste Management Facility (WMF) as a designated Corrective Action Management Unit (CAMU). The function of the WMF is to provide a repository for the remediation waste generated from Environmental

Restoration risk reduction activities and Deactivating and Decommissioning activities at the Site. The WMF will have to receive remedial waste from closure activities which are starting in 1996. Planned remedial actions require a viable remediation waste management system be operational prior to any removal actions, to eliminate additional interim waste storage concerns.

This presentation focuses on the second stage of the study, the onsite selection process. In each of the first two phases, the best alternative was selected and used as a basis for the next phase. Selection criteria reflected the project objectives, siting and Corrective Action Management Unit (CAMU) criteria as well as the Interim Measure/Interim Remedial Action (IM/IRA) process. Consideration was also given to the National Environmental Policy Act (NEPA) throughout the decision process.

SITE DESCRIPTION AND BACKGROUND

The Site is located in northern Jefferson County, Colorado approximately 16 miles northwest of Denver. Other surrounding communities include Boulder, Westminster, and Arvada, all of which are located less than 10 miles to the northwest, east, and southeast of the Site, respectively. The Site consists of approximately 6,550 acres of federal land in sections 1 through 9 and 15 of T2S, R70W, 6th Principal Meridian. Most Site structures are located within a protected central area, referred to as the Industrial Area (IA) of approximately 400 acres, and surrounded by a buffer zone of approximately 6,150 acres.

The majority of residential use within five miles of the Site is to the northeast, east and southeast. Commercial development is concentrated near residential developments north and southwest of Standley Lake as well as around Jefferson County Airport, approximately three miles northeast of the Site. Industrial land use within five miles of the Site is limited to quarrying and mining operations. Open space lands are located northeast of the Site near the City of Broomfield and in small parcels adjoining major drainages and small neighborhood parks in the surrounding communities. Irrigated and non-irrigated croplands, producing primarily wheat and barley, are located north and northeast of the Site and in scattered parcels adjacent to the eastern boundary. Future land use in the vicinity of the Site most likely includes continued urban expansion, increasing the density of residential, commercial and perhaps industrial land use in the area.

The operations at the Site consisted of fabrication of nuclear weapon components from plutonium, uranium, and non-radioactive metals (principally beryllium and stainless steel). Fabrication operations were terminated and the plant was placed in shut-down mode in 1991. These operations resulted in liquid and solid wastes containing hazardous and radioactive constituents that were managed at waste processing units across the plant. Parts made at the plant were shipped elsewhere for assembly. In addition, the plant reprocessed components after they were removed from obsolete weapons for plutonium recovery. Both radioactive and non-radioactive wastes were generated in the production process. Current waste handling practices involve onsite and offsite recycling of hazardous materials, onsite storage of hazardous and radioactive mixed wastes, and offsite disposal of Low Level Waste (solid radioactive materials) at the Nevada Test Site and Low Level Mixed-Waste (LLMW) at the Envirocare facility in Utah. However, Site operations historically included both onsite storage and disposal of hazardous waste, LLW and LLMW.

PHASE 1 - OFFSITE VS ONSITE EVALUATION

The purpose of Phase I of the project was to select the best disposal option for environmental remediation waste originating at the Site. The three options are:

Onsite disposal
Offsite disposal, or
the No Action Alternative.

This selection process is part of the decision making process for an Interim Measure/Interim Response Action (IM/IRA) to address remediation waste, and the intent is to reflect the current waste and environmental strategies and to be consistent with input from the stakeholders. Onsite disposal was selected as the best of the three options. The selection of onsite disposal was based on the following:

Projected costs for onsite disposal are significantly lower than for offsite disposal, Because of the lower cost, there would be more risk reduction activities accomplished in support of the Accelerated Site Action Project (ASAP) and the Rocky Flats Conceptual Vision (the Vision),

There would be less public exposure (human health risk) during transportation, along with less involuntary risk,

Less risk of spills in handling and transportation,

Greater capacity for onsite disposal compared to offsite disposal. An onsite facility would be more accessible and more available when needed,

Fewer schedule restrictions for an onsite facility, and

Fewer analytical requirements for an onsite facility because there is less redundancy in sampling requirements.

Ultimately, the most important difference between the onsite and offsite options is total cost and the effects of cost for reducing risk at the Site. In essence, the more it costs to dispose of a cubic yard of contaminated material, the fewer cubic yards of material the Environmental Restoration (ER) program would be able to clean up in a given time frame. This prolongs the cleanup efforts at the Site and allows contaminated materials to remain uncontrolled for much longer periods of time. In turn, this increases overall risk at the Site to human health including the offsite population, the onsite workers, and the environment.

As part of the evaluation, it was necessary to select the waste management option that could best reduce the overall risk to human health and the environment while remaining fiscally responsible. Budget restraints affect the degree of risk reduction possible. Because of lower costs, the onsite option is clearly the selection for reducing the overall risk to both the public and the environment.

With a given budget to perform work onsite, more cleanup actions can be performed more effectively, and therefore, the Site can be made safer with onsite disposal.

PHASE 2- ONSITE WASTE MANAGEMENT FACILITY SITING EVALUATION

As onsite disposal became a viable alternative for waste management at the Site, Phase 2 of the project commenced. The objective of Phase 2, the Siting Evaluation, was to: 1) identify and rank criteria to be used for WMF site selection, 2) develop a methodology for comparative analysis of the different sites and then 3) select a site or sites that would be a suitable location for a WMF within the boundaries of the Site.

This site would be a location for a WMF that would accept remediation wastes with low-level radioactive and/or hazardous constituents, but not preclude the shipment of remediation waste that can be more effectively

and economically managed offsite. The facility would be designed and constructed to meet all the applicable federal, state and local regulatory requirements.

Waste streams would include, but not be limited to:

Contaminated soil and debris collected from accelerated actions and hot spot removals,

Asphaltic materials and pondcrete (solidified sludge from the solar ponds),

Pond sludge,

Sediments from onsite ponds,

Toxic Substance Control Act (TSCA) waste such as asbestos and polychlorinated biphenyls (PCBs),

Treatment by-products from groundwater, surface water, and/or soil remediation actions,

Investigation-Derived Materials (IDM) generated during remedial investigations at the Site not suitable for disposal at the onsite sanitary landfill,

Debris from deactivation and decommissioning activities

Identification and Ranking of Siting Criteria

A number of categories of criteria were considered in developing the siting criteria. It was agreed that the criteria had to include at a minimum Applicable or Relevant and Appropriate Requirements (ARARs) as well as general guidelines that had been discussed with the various stakeholders meetings regarding a WMF at the Site. The criteria were then placed into six major categories and further divided into specific issues within each of these major categories, as outlined below:

Ability to designate the WMF as a Corrective Action Management Unit (CAMU). Key points of the CAMU Rule include:

- The CAMU should facilitate the implementation of reliable, effective , protective, and cost-effective remedies,
- Associated waste management activities shall not create unacceptable risks to humans or the environment resulting from exposures to hazardous waste or hazardous constituents,
- The CAMU should not include uncontaminated areas of the facility unless the inclusion of such areas is more protective than management of wastes at contaminated areas of the facility,
- Areas within the CAMU where remediation waste will remain in place after closure should be managed to control, minimize, or eliminate future releases to the extent necessary to protect human health and the environment,
- The CAMU should expedite the timing of remedial activity implementation when appropriate and practicable,
- When appropriate treatment technologies should be used that enhance long-term effectiveness of remedial actions by reducing the toxicity, mobility or volume of waste that will remain in place after closure of the CAMU

Ensure the Protection of the Public, per Code of Colorado Regulations (CCR) 6 CCR 1007-2 Part 2, which is the Requirements for Siting of a Hazardous Waste Disposal Site. Key points include:

- Geological and hydro-geological conditions of a site in which hazardous waste is to be disposed of should be such that reasonable assurance is provided that the wastes are isolated within the disposal area and away from pathways to the public for 1,000 years,

- Structural related issues evaluated will include slope and geo-technical stability,
- The immediate area of the site should be in an strata of minimal ground water flow,
- Geological strata combined with engineered barriers should provide a minimum permeability of 10-7 cm/sec or there should be sufficient thickness and distance between the disposal location and nearest usable aquifer to isolate any materials that are disposed of in the facility,
- Consideration should be given to the relative depth to bedrock and ground water, including seasonal fluctuations for ground water,
- The relative distance to the nearest discharge area shall include consideration of ground water flow direction and travel time

Issues that are Specific to the Site and support the Rocky Flats Conceptual Site Vision (the Vision). Key points evaluated include:

- The ability to support the Site Vision,
- The impacts from existing utilities, sewer lines, process waste lines and communication lines if the selected WMF site is with the Industrialized Area

Cost Criteria. Key points evaluated include:

- Cost of pre construction activities, including building demolition, subsurface utility line removal and rerouting, access requirements and power/facility requirements,
- Cost of engineering and construction of protective measures

Regulatory Support focusing on using State principles for onsite management of contaminated materials. Key points evaluated include:

- Minimization of the number of disposal sites and of consolidation of contaminated materials,
- Site a centralized disposal facility in an area of optimum geologic parameters preferable within or close to the Industrial Area,
- Any WMF should be located in an area having limited future land use potential and would be controlled by DOE until the interred waste no longer presents a risk to human health or the environment

Other Stakeholders concerns which included:

- General public perception and acceptance,
- Municipal or County acceptance,
- Department of Energy (DOE) Orders,
- The National Environmental Policy Act (NEPA)

Methodology for Comparative Analysis of the Onsite Locations

A basic assumption was that the entire Site, both within the buffer zone and the Industrial Area would be included in the siting evaluation. A series of maps were produced to assist in this evaluation. These maps included:

A Site Location Map showing the location of building, roads and other plant infrastructure,

A map showing hydrologic conditions including the depth to water table and the area encompassed by the 100-year floodplain,

A map showing the geological and geo-technical conditions including inferred faults traces, and areas with greater than 15% slope,

Alluvial thickness map, showing the thickness of the alluvium, and by inference potentially economic thickness of gravel,

A Ecology and NEPA map showing the location of seeps, wetlands and the Preble's Meadow Jumping Mouse (PMJM) probable habitat. PMJM is a wildlife species that is being considered for listing as a species of concern, or as a threatened and endangered species

The different elements from these maps were combined onto one map called the Adverse (for WMF siting) Conditions map. Areas highlighted on this map contained one or more conditions that would be major obstacles in siting a WMF and thus were removed from further consideration in the evaluation. In the Industrial Area, areas with minor existing buildings were kept in the study.

This initial screening of the Site reduced the number of locations being evaluated to seven, four in the buffer zone and three in the Industrial Area.

Industrial Area sites include:

Industrial Area - West (IA-W), an area on the west side of the Industrial Area,

Industrial Area - East (IA-E), an area on the east side of the Industrial Area, and

Solar Ponds - an area adjacent and east of the Solar Pond in the northeast section of the Industrial area.

The buffer zone sites are:

The New Sanitary Landfill (NSL),

An area within Operable Unit 2 (OU2),

An area in the southeast quadrant (SE Quad) of the buffer zone, and

An area in the southwest quadrant (SW Quad) of the buffer zone.

The methodology that was then applied first began by developing a relative weighting factor (%) for each of the six general categories of criteria as shown in Table I.

Table I

Next, each category was divided into specific issues. There are a total of 38 separate and distinct issues in the six categories (Table II). Each of the issues was subjectively assigned a value between 1 and 3, with a 3 being a more important issue and 1 being less important. As an example, for Category 2, Public Protection, the first issue is geological and hydro-geological conditions of a site in which hazardous waste is to be disposed of should be such that reasonable assurance is provided that the wastes are isolated within the disposal area and away from pathways to the public for 1,000 years. The second issue is structural related issues would include slope and geo-technical stability.

Table II

In this instance the first criteria was subjectively assigned a value of 3 because it was decided that issue was a more important issue then the second issue relating to slope stability which was assigned a value of 2. The next step in this methodology was to develop a matrix form with the 7 location versus the 38 criteria. The 38 criteria were then compared to the 7 sites (Table III). The sites were evaluated against the criteria and relative to one another. Each issue was assigned a score between 0 and 1. A score of 0 for any of the 38 issues would signify a fatal flaw and the site would be withdrawn from further consideration in the siting evaluation.

Table III

A weighted average was arrived at for each of the categories and the values were summed. The results of the evaluation are shown on the following table.

Overall, the Solar Ponds, the site adjacent and to the east of the Solar Ponds, ranked highest with a score of 68.5%, followed by the IA-West site (67.1%), and the NSL (65.1%) as the location to site the WMF. The Solar Ponds site was then carried on to the third phase of the evaluation.

PHASE 3 - COMPARISON OF ONSITE DESIGN OPTIONS

This third phase of the project was to select and evaluate different design options for an onsite WMF. The following design concepts were considered:

- Pyramid Design
- Butler Building
- Slab on Grade
- Hardened Concrete Vault
- Above Grade Concrete Lined Cell (CLC) with Bulk Storage
- Above Grade Concrete Lined Cell with Cargo Containers
- Above Grade Disposal Cell (landfill)
- Entombment
- No-Action

An initial concept design screen using the same criteria as was used in the siting study resulted in the following four final design alternatives selected to be carried on through the final design evaluation.

- Butler Building
- Above Grade Concrete Lined Cell (CLC) with Bulk Storage
- Above Grade Disposal Cell (landfill)
- No-Action

The final design alternative comparison used the following Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) nine criteria to select the best alternative:

- Protection of Human Health and Environment
- Compliance with ARAR's
- Long Term Effectiveness and Performance
- Reduction of Toxicity, Mobility, and Volume through Treatment
- Short Term Effectiveness
- Implementability
- Cost
- Regulatory Agency Approval
- Community acceptance

The above grade Concrete Lined Cell (CLC) was selected as the best alternative as the WMF several reasons. The primary reason is the high cost of fill material which must be imported to the site to provide the berms or side support for the waste. The use of concrete walls substantially reduces the amount of back fill material and the cost of the facility. The CLC adds another layer of protectiveness (the concrete walls and floor) to ground water from the leachate generated during operations. The CLC would be built to be compliant with RCRA Subtitle "C" standards. The above grade construction prevents ground water infiltration. The liners and leachate collection system provides the ability to detect leaks and recover contaminated leachate prior to entering the environment.

Also, this design allows for modular installation which would optimize the sizing of the cell and timing of the installation as waste is generated. This design will speed up risk reduction activities under the Site Vision because the facility can be built in sections. The first module of the 100,000 cubic yard facility would be sized for 25,000 to 33,000 cubic yards and could be installed quicker in order to handle remedial waste ready for disposition early on. Even though the design offers the benefits of a permanent facility, it still allows for retrievability. This yields the additional flexibility of being able to utilize the facility as either storage or closure as a CAMU at any time.

Even though it was one of the most versatile, protective and reliable designs, it was also the most cost effective.
The Above Grade Disposal Cell is a standard waste landfill, compliant with RCRA Subtitle "C" standards. This facility would be built above the existing grade to prevent ground water infiltration. Like CLC, the liners and leachate collection system provides the ability to detect leaks and recover contaminated leachate prior to entering the environment. This facility offered some degree of flexibility and retrievability and still remains one of the least expensive alternatives.
The Butler Building alternative was selected for the final screening because it allowed for storage of the waste until the final disposition is determined. The waste would be stored in monitored cargo containers and would be retrievable. It is believed that public acceptance would be greater for this alternative despite the tremendous cost, lack of useful life, and reduced protection to the environment.
The No-Action alternative was included in the final screening because it must go through the entire screening process as required by NEPA.

30-44
DETERMINATION OF MECHANICAL AND ISOLATION PERFORMANCE OF GRANITES
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ABSTRACT
In this study, several mechanical and isolation tests are applied on granite samples in laboratory. Cylindrical granite samples, 54mm (NX core size) in diameter are prepared for mechanical tests. 50 mm in diameter samples are prepared for isolation tests. Uniaxial compressive strength, triaxial compressive strength, tensile strength and deformability of granite samples are determined under different thermal loads by using a hydraulic compressive machine. Cohesion and internal friction angle are determined. Thermal load effects on granites are shown in graphs. Thermal loads cause decreasing in mechanical strength of granites and increasing in elasticity. For isolation performance tests, radionuclide transport in granite sample are determined under different heat and pressure by using a test apparatus. Transported radionuclide amounts and activities are shown in graphs.
INTRODUCTION
The geological environment of the repository is a key element in the multi-barrier system approach to nuclear waste isolation (1). Special attention should be given to ensure long term stability and isolation of the repository. Decay heat emitted by the waste following closure of the repository, will effect the mechanical properties of surrounding rock. Small scale effects (microcracks etc.) can occur near to sources of heat in granites.
Radionuclide migrates in host rock with flowing groundwater. Connected pores in rocks have important effect on migration of radionuclide. Temperature of groundwater is increased due to the increasing pore water pressures . For these reasons, radionuclide transport is not only effected by host rock properties but also effected by main medium parameters, such as temperature and pressure (2).

MECHANICAL PROPERTIES

In this study, heat effects on mechanical strength of granites are determined in laboratory. Granite samples are prepared; F= 54 mm (in diameter) and W= 120mm (in width) for uniaxial and triaxial tests, F=54 mm, W= 54mm for tensile strength. Granite samples are heated at different temperatures (24 hours) than covered by wax to conserve their internal parameters and cooled. These samples are tested by hydraulic compression machine according to ISRM (3) suggested test methods. Cohesion and internal friction angle of granite specimen, which are determined by triaxial compressive strength and several properties are shown in Table I.

Table I

Uniaxial compressive and tensile strengths determined after different thermal loads; 20C, 200C, 300C, 400C and 500C. According to thermal loads, Uniaxial compressive strength and tensile strength of granite samples are shown in Fig. 1. Axial deformability of specimen is determined for each thermal loaded samples. Elastisity modules are shown in Fig. 2.

Fig. 1

Fig. 2

ISOLATION TESTS

A special testing equipment (Fig.1) is designed for monitoring radionuclide transport in granite samples. A stainless steel column is developed for introducing radionuclide solution to the one face of granite. Temperature of the radionuclide solution is increased by heaters. Heat is applied externally on the stainless steel column by two heaters. Heater surrounds radionuclide transfer column. Heaters are controlled by relay and switches on the control panel. Temperature is measured by a thermocouple near the contact zone of granite and radionuclide solution. Pressure values are taken by two manometers in different scales which are located on the upper lid. (2)

Upper lid is opened and cylindrical granite sample is emplaced into the bottom of column. Isolation, between the cylindrical surface of the sample and column circumference is ensured by isolation material and controlled by the bottom lid. 350 ml. radionuclide solution is poured into column. Both lids are closed tightly. Required temperature is adjusted from control panel. When contact temperature is increased to the required temperature level, sample container for transported solution is located under the discharge hole. After a constant time period, solution that passed through the granite sample is collected in sample container. Volume of this transported solution is determined. And analyzed to determine its activity by gamma spectrometry.

TEST RESULTS

Transported volume against to temperature and pressure, transported activities against to temperature and pressure are shown in Fig. 2. and Fig. 3.

Fig. 3

Fig. 4

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Session 31 -- HEALTH & SAFETY - LESSONS LEARNED
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31-2

COMBINING REQUIRED TRAINING PROGRAMS FOR MORE EFFICIENT TRAINING

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ABSTRACT

Many DOE workers have similar training requirements imposed by multiple agencies. Refresher training for Treatment, Storage, and Disposal (TSD) Facility workers is one example. Both Subpart B of 40 CFR 264 & 265 (EPA's RCRA regulations) and paragraph (p) of 29 CFR 1910.120 (OSHA's HAZWOPER regulations) require annual re-training. To better serve our workers, LANL's ESH Training Group created an annual refresher program that addresses issues and topics from each of the initial training courses. This paper describes how our course for TSD Facility workers fulfills tha annual training requirements for two regulatory agencies in one eight-hour session.

INTRODUCTION

When rating the efficiency of their training, most participants are influenced both by how applicable the training is to their current job duties and by the amount of time they spend in training sessions. Time spent covering similar information in different courses is usually viewed as non-efficient training. The fact that different regulatory agencies mandate similar training for the same category of workers does not change participants' perspective that attendance at similar, but required, training is non-effective use of their training hours. An example of dual demands on participants' time is the required refresher for workers at Resource Conservation and Recovery Act (RCRA) permitted treatment, storage, and disposal (TSD) facilities. Both the Occupational Safety and Health Administration (OSHA), through 29 CFR 1910.120 (p), and the Environmental Protection Agency (EPA), through 40 CFR 264.16 & 265.16, have imposed annual refresher training for TSD facility workers.

Another case of poor use of training hours is the delivery of identical training to workers performing totally different job duties. For example, 29 CFR 1910.120 (e)(8) and (p)(7) requires annual refresher training for both environmental restoration workers and TSD facility workers. These groups of workers have very dissimilar job assignments and should have refresher training that is specific to their job duties. However, many organizations use the same HAZWOPER refresher training session for both groups of workers.

This paper describes how Los Alamos National Laboratory's (LANL) ES&H Training Group working in conjunction with the Industrial Hygiene and

Safety Group have created separate annual refresher courses that focus on the unique needs of each of these groups of hazardous waste workers.

REGULATORY BACKGROUND

OSHA Requirements

OSHA has recognized that handlers of hazardous wastes face health and safety hazards that workers in other industries do not often encounter. To provide protection for these workers, OHSA issued 29 CFR 1910.120, the HAZWOPER standard in 1989. In this standard OSHA prescribes the length, topics, and frequency of training for three different groups of hazardouswaste workers. Paragraph (e) focuses primarily on general site and occasional site workers engaged in corrective actions or clean-up operations of hazardous waste sites; paragraph (p) prescribes the training requirements for TSD facility workers; and paragraph (q) addresses the training needs for emergency responders to hazardous material releases.

A quick comparison of these three sections reveals that the degree of training required depends upon the potential to be exposed over permissible exposure levels (PELs) to hazardous materials. Paragraph (e) mandates either 40 hours or 24 hours of initial training for restoration workers depending upon how well the hazards at the work site have been characterized. The greater the risks due to unknown conditions, the greater the number of training hours required. The language in paragraph (p) recognizes that the risks at TSD facilities are better characterized, so only 24 hours of initial training is required. The number of training hours required for Paragraph (q) workers depends on the emergency responders' assigned duties. The hours of required training increase as the workers' responsibilities increase. Although all three categories of workers need annual refresher training, the standard specifies that workers covered by paragraph (e) and paragraph (p) receive 8 hours of annual refresher training.

EPA Requirements

Through the passage of the RCRA in 1976, Congress empowered the EPA to regulate the handling, treatment, and disposal of hazardous wastes. Standards for operators of hazardous waste TSD facilities are listed in 40 CFR 264 and 265. Although the EPA does not require a specific number of training hours, sections 264.16 and 265.16 requires training for personnel working at TSD facilities. Paragraph (c) in each of these sections requires that TSD facility workers receive annual refresher training.

COMPARISON OF OVERLAPPING TRAINING REQUIREMENTS

Overlaps Within 29 CFR 1910.120

Paragraph (e)(8) of the HAZWOPER standard states that workers covered by paragraph (e) shall review the topics covered in their initial 40- or 24-hour training during an annual refresher. Thus, refresher training for these workers should review the following topics:

- the health and safety hazards on the site;
- the proper use of personal protection equipment (PPE);
- safe work practices for minimizing risks;
- engineering controls appropriate to the site;
- signs and symptoms of overexposure; and
- any site specific health and safety plans (HASP), including the following topics:
  - hazard analysis of the site,
  - required PPE at the site,
  - required air monitoring,
  - medical surveillance,

- site control measures, decontamination procedures,
- emergency response plan, confined space entry procedures, and
- spill containment program, environmental sampling procedures.

The language in paragraph (p)(7) is not as prescriptive as in paragraph (e)(8). Workers covered by paragraph (p) are required to have annual refresher training that enables them to perform their duties in a safe and healthful manner. An implication of this paragraph is that the annual refresher should review the required programs being implemented by the employer for safe operations at the TSD facility. Thus, refresher training sessions for TSD facility workers should review the following topics:

the health and safety program designed to evaluate and control health and safety hazards on site;
the hazard communication program;
the medical surveillance program;
any new technology currently being used on site;
decontamination procedures;
safe material handling procedures, including drum handling; and
the site emergency response program.

There is potential training overlap on hazard recognition, administrative and engineering controls, proper PPE, and decontamination procedures in these two refresher requirements. However, the less prescriptive requirements for TSD workers allow for very site-specific refresher training sessions. There is minimal overlap between the training required for paragraph (q) workers and paragraph (e) or (p) workers. The training requirements for emergency responders are covered by paragraph (q)(6) of the HAZWOPER standard. The number of required hours for these workers depends on the duties and functions each worker performs. This section of the regulation is very specific on which topics must be addressed, and this is the only section of the HAZWOPER standard that requires an individual worker to demonstrate competency before being certified.

OVERLAPS BETWEEN HAZWOPER AND RCRA

Both OSHA, through paragraph (p) of the HAZWOPER standard, and EPA, through subpart B of 40 CFR 264 and 265, require that TSD facility workers receive annual refresher training. Although OSHA requires 8 hours of refresher training, the EPA does not have specific time requirements. Also, the EPA is not as prescriptive as OSHA on topics that must be covered during the training. The RCRA training goals specify that workers are able to respond effectively to emergencies and that they know how to handle hazardous wastes. Thus, refresher training for TSD facility workers should review the following topics:

the site emergency procedures or contingency plan;
emergency equipment available;
communication and alarm systems;
monitoring equipment;
inspection procedures;
programs designed to control health and safety hazards on site, and
safe material-handling procedures, including drum handling.

There is significant overlap between the HAZWOPER and RCRA refresher training requirements. Both require that TSD facility workers receive training in four key areas:

site-specific emergency response,

administrative and engineering controls for reducing risks due to health and safety hazards,

proper use of PPE for the hazards on site, and

procedures for safe handling of hazardous wastes.

In contrast to OSHA the EPA does not require a specific number of hours, but it does require many of the same refresher training topics. Because of this similarity of required topics, it seems appropriate to fulfill the refresher training requirements for both of these regulatory agencies in one training session. The important issue is regardless of which training topics are required, the training should be appropriate to the participants' job duties.

HAZWOPER REFRESHER COURSES AT LANL

The content for each training course is determined through a needs analysis that includes customer interviews and surveys. Based on the analysis, the ES&H training staff determined that there was enough interest and demand for separate HAZWOPER refresher courses: one course for environmental restoration workers and another one for TSD facility workers. The ES&H training staff did not create any refresher training for emergency responders because LANL has a Hazardous Materials Response Team that is responsible for emergency response training. The Industrial Hygiene and Safety Group provided the subject-matter experts (SMEs) who reviewed the course materials for content accuracy and provided suggestions on appropriate activities.

Each of these courses uses a variety of interactive training activities such as small group discussions, review games, outdoor simulations, and dress-out exercises. The following sections of this paper describe the content of each of these courses and present the similarities and differences of these two courses.

HAZWOPER REFRESHER FOR ENVIRONMENTAL RESTORATION WORKERS

Environmental restoration projects at LANL range in complexity from collecting soil samples using hand augers, to drilling exploratory bore-holes, to total removal of contaminated soils and buildings. The refresher course focuses on the hazards associated with working on thesecharacterization and remediation projects. The five modules of this course are structured around the following objectives:

to identify potential hazards and describe safe work practices that can minimize the risks associated with the hazards;

to describe the impact that any new or modified federal regulations or LANL policies will have on the workers;

to review the required components of a site-specific health and safety plan (HASP);

to review the rights and responsibilities of both the worker and the employer involved in hazardous waste operations;

to have the participants demonstrate their ability to use the information provided in a HASP by completing a simulated field exercise; and

to have participants identify the correct action or response to a given health or safety problem.

In the first module, Risk Minimization and Regulatory Update, participants view several photographs of local work sites and identify any physical, chemical, radiological, or biological hazards at the site. The participants are asked to record in their workbook a description the hazards and to propose some safe work practices that might minimize the risks. This module also provides an update of federal and state

regulations and Laboratory policies. This risk-minimization module and the fourth module (Lessons Learned) are based on current environmental restoration sites or occurrence reports.

The second module, Health and Safety Plans, reviews the required components of a HASP. This module ends with the presentation of the new HASP written for all environmental restoration projects at LANL. Due to the inefficiency of rewriting the complete plan each time a change in the scope of work occurred, the environmental restoration project staff and the Industrial Hygiene and Safety Group developed a two-part HASP. The first part is a generic HASP that addresses legal and policy issues common to all projects, and the second part is a site specific (or SSHASP) that supplements the generic HASP. The site-specific document is the one that must be modified if conditions at the site change. The heavy emphasis on the HASP makes the content of this module more applicable to paragraph (e) workers than paragraph (p) workers.

In the third module, Task-Hazard Analysis and Field Exercise, the participants conduct a simulated core-sampling exercise. This module is the heart of the refresher training sessions. To complete this field exercise, the participants need to read the SSHASP, don the proper PPE, and coordinate as a team to collect the sample. Because the needs analysis had indicated that knowing how to obtain a radiological work permit (RWP) needed reinforcing, the simulation is designed to exceed the LANL developed radiation contamination action levels that require an RWP. This simulated event should cause the field-team leader or site-safety officer to stop work and initiate the process for an RWP.

The details of the simulation are given in their workbook and in the SSHASP written for the simulation. The training focus of this module is on finding and applying the information provided in the HASP to each task being performed at the work site. To accomplish finding the information in an efficient manner, the participants are assigned to small groups that are responsible for finding and presenting the following blocks of information to the whole group:

Following the presentations by the small groups, the whole class completes the field exercise. The exercise includes having teams of workers dress out in the proper PPE, establish the work zones, collecting the samples, and perform a decontamination of personnel and equipment. Following the decontamination and doffing procedures, the module closes with a debriefing session and review the teams' performance.

In the fourth module, Lessons Learned, participants read several case studies describing an occurrence or incident that resulted in personnel injuries. The participants are asked to describe what might be the cause of the accident and propose safe work practices that could reduce the risk or eliminate the hazard. Each of the case studies appeared in the Occurrence Reporting and Processing System (ORPS) within in the last sixteen months.

The last module, Review of Selected HAZWOPER Content, provides a rapid review of most of the topics covered in the participants' initial 40- or 24-hour training. The module is essentially a Jeopardy-like review game in which the participants respond to questions on topics such as toxicology, signs and symptoms of overexposure, proper container-handling techniques, air monitoring procedures, and medical surveillance programs.

HAZWOPER REFRESHER FOR TSD FACILITY WORKERS

Although LANL has a few treatment facilities and disposal sites, most of the participants in this refresher session work at hazardous waste

storage facilities. Thus, the annual refresher for these workers focuses on hazards associated with container handling, symptoms of exposure to the more common hazardous materials in storage, hazard identification at these facilities, and proper waste handling and segregation techniques. The seven modules of this course are structured around the following objectives:

to describe the impact that any new or modified federal regulations or LANL policies will have on the workers;

to describe the changes in LANL's waste management structure;

to identify the rights and responsibilities of both the worker and the employer involved in hazardous waste operations;

to identify recurring problems of noncompliance with RCRA.

to identify the response to a given situation based on information presented during initial HAZWOPER training; and

to demonstrate the ability to recognize and correct hazardous or noncompliance situations by completing a simulated field exercise.

The first two modules, LANL Reorganization and Regulatory Update, are similar to the first module of the environmental restoration refresher course. The difference is the extent of the coverage. This additional depth is needed because LANL has undergone extensive reorganization that has effected waste handling at the Laboratory. This information is much more significant to the TSD facility workers then to the environmental restoration site workers.

In the third module, Hazard Identification and RCRA Refresher, the participants receive a review of the inspection record form (IRF) used to record the daily or weekly inspection required at RCRA-permitted facilities. The IRF has two functions. First, it creates a paper trail that documents that the TSD workers have completed the required RCRA inspections. Second, it is an administrative control that helps assure that the TSD facility is operated in a safe manner. The IRF assures that any physical, chemical, or radiological hazards are identified and corrective actions are initiated to correct the deficiency. In the fifth module, the participants use the IRF and complete a field exercise that includes an inspection of a simulated TSD facility.

The fourth module, Actions Required, addresses both the physical and chemical hazards present at TSD facilities. Like in the environmental restoration refresher, the participants identify potential hazards illustrated in several photographs. However, the scenes are different. The photographs used for this refresher show activities at LANL TSD facilities. This module also spends more time on chemical storage and incompatibility problems. The participants are givena map of a simulated TSD facility and asked to designated were they would store specific chemical and hazardous wastes at the facility. This module ends with a video on hazardous chemicals and their properties.

Module five is the field exercise. This exercise requires the participants to use the IRF and document at least twelve of the sixteen deficiencies present in a simulated TSD facility. Following the inspection, the participants work in small groups and prepare a written description of two of the deficiencies, determine what steps are necessary to correct each deficiency, and propose what PPE is needed if someone were to follow through and correct the deficiency. Depending on the time, participants can go back out to the simulated site and correct several of the simpler problems or as a class discuss what needs to be done for each of the sixteen deficiencies.

In the sixth module, Lessons Learned, participants read several case studies that describe an occurrence or incident that resulted in personnel injuries or fines. This module is expanded, compared to the one in the environmental restoration refresher, to include both OSHA and RCRA occurrences. For the OSHA incidents, the participants are asked to describe what might be the cause of the accident and propose safe work practices that could reduce the risk or eliminate the hazard. For the RCRA occurrences, the participants are asked to identify the root cause for each violation and propose corrective actions to change the conditions that resulted in the non-compliance citation.

Just like the environmental restoration refresher, the last module provides a rapid review of most of the topics covered in the participants' initial 40- or 24-hour training. The same Jeopardy-like review game covers the topics of toxicology, signs and symptoms of overexposure, proper container-handling techniques, air monitoring procedures, and medical surveillance programs.

COMPARISON OF HAZWOPER REFRESHER COURSES

Based on the needs analysis, the refresher training was made more efficient for the customer by combining the required OSHA and EPA refresher training. The critical step was ensuring that the new refresher for TSD facility workers fulfills the intent of all of the legal drivers. Because both OSHA and EPA require that the refresher include emergency response, controls for reducing risks, use of PPE, and safe handling procedures for hazardous waste containers these topics were the first ones addressed in the course design.

To the same course sequence and major topics of the existing HAZWOPER refresher course, which met the OSHA drivers, was added additional TSD- and RCRA-specific material to meet the EPA drivers. Because the daily duties of TSD facility workers are different from environmental restoration workers, the activities and contents in our courses diverged on two key topics: safe work practices and administrative and engineering controls. Because workers at TSD facilities risk exposure to chemical hazards as their daily duties involve handling containers ofhazardous materials, their refresher focuses on work practices and controls that minimize this risk. Although environmental restoration workers can encounter unknown chemical hazards, the Laboratory's experience shows that physical hazards pose a greater risk and their refresher focuses on these hazards.

This difference in hazards resulted in creating several separate activities for the refresher courses. The following tables list the similarities and differences of these two courses:

Table I

Table II

For the safe work practices portion of the refresher training, the TSD facility participants complete an activity that reviews some chemical incompatibilities and safe storage practices for various chemicals. The environmental restoration workers spend time reviewing their sit-specific HASP. The chemical segregation activity allows the participants to be in charge of a "new" TSD facility and decide where each chemical will be stored. The participants are allowed to use the DOT shipping labels, guidebooks on hazardous materials, and their neighbor to make the necessary decisions. The HASP review requires the participants to search through the HASP to answer specific job-related questions concerning PPE, action levels, decontamination procedures, and site security.

The two administrative forms reviewed in these refresher courses are the RWP and the IRF. In addition to the HASP, the RWP is required for all "hot" jobs at LANL. This permit specifies the PPE required for the job and lists any additional health or safety concerns. The IRF is LANL's form used to document the inspections required by the EPA for all RCRA-permitted TSD facilities and serves as a hazard identification tool. These daily or weekly inspections help identify any site deficiencies that may result in worker overexposure to hazardous materials or create a hazardous situation. In the TSD refresher, the participants are asked to identify all of the deficiencies at the simulated TSD facility and then describe the appropriate actions required to correct the deficiencies or minimize the risk. The participants are asked to include a list of any tools or equipment needed and propose what PPE, if any, should be used during the corrective action. These two administrative forms are part of the LANL's administrative procedures that helps assure worker health and safety.

CONCLUSIONS

Both HAZWOPER and RCRA refresher training have the same four goals:

- to remind participants of safe work practices,
- to provide the participants with an opportunity to practice job-related safety procedures,
- to review emergency response programs or procedures, and
- to update participants on any policy or regulatory changes.

LANL's HAZWOPER refresher training for TSD facility workers is more efficient this year because the training reflects the design goals identified by in the needs analysis. The first goal was to reduce demands on participants' time, which was achieved by combining refresher training required by two different agencies into one training session. The second goal was to make the training session more appropriate by making the training more job-related, which was achieved by creating separate refresher training for TSD facility workers and for environmental restoration workers. Thus the creation of these two similar, yet job-specific, refresher courses meet the intent of the OSHA HAZWOPER standard (to protect workers' health and safety) and the EPA RCRA requirements (to protect both the worker and the environment) while satisfying the customers' demands for efficient and relevant training.

31-3

FORMALIZING THE INTERFACE OF OCCUPATIONAL MEDICINE AND INDUSTRIAL HYGIENE AND SAFETY: COORDINATION INSPIRES SYNERGY

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ABSTRACT
Los Alamos National Laboratory has had an ongoing "interactive relationship" between the departments of industrial hygiene and occupational medicine. This requirement was recently formalized in a University of California ES&H performance measure. A close working relationship between these two departments enhances the ability to direct resources to identify and follow employees who belong in special occupational medicine surveillance programs, and to get referrals from medical when unusual examination results are found. Los Alamos has formalized the relationship between the two departments through a recently developed "interface document" that clearly defines the responsibilities and required procedures for coordination. Specific areas of mandated coordination include quarterly meetings of senior staff, joint workplace evaluation and systematic prioritization, information exchange, continuous quality improvement, individual program elements, and measures of performance. The elements of the program will be presented and the first eighteen months of implementation reviewed.
INTRODUCTION
The interface between the Occupational Medicine Group (ESH-2) and the Industrial Hygiene and Safety Group (ESH-5) (6) is an integral component of the overall Occupational Health and Safety Program for the Laboratory. ESH-2 provides a comprehensive medical program which assures protection and promotes enhancement of worker's health. Protection of worker's health includes medical surveillance and job certification when required or prudent, periodic age-dependent medical evaluations, human reliability/fitness for duty assessments, managed care/rehabilitation of work related injuries and illnesses, emergency response capability, worksite evaluations, population studies for early detection of adverse health effects and a clinic which provides support to these programs. Enhancement of worker's health is offered through medical health promotion programs (e.g., cholesterol reduction, skin cancer awareness, flu immunization, etc.), The Employee Assistance Program (e.g., personal counseling, stress management, alcohol/drug rehabilitation, etc.), and the Wellness Center (fitness programs/equipment, aerobic instruction/classes and nutrition counseling).
ESH-5 (6) works closely with Laboratory line managers to help ensure the health and safety of personnel and to ensure that employee exposure to chemical, physical, biological, and ergonomic hazards is minimized. Primary support services include the full spectrum of anticipation, recognition, evaluation, and control of occupational health hazards: consultation and field support to operating divisions; design review of new or modified facilities and operations, maintenance of Material Safety Data Sheets; review of standard operating procedures and special work permits, implementation of training programs; recommendations for personal protective equipment; toxicology support services; and evaluation of air cleaning systems.
PURPOSE
This document describes the interactions between the Occupational Medicine Program and the Industrial Hygiene and Safety Program that are used to effectively exchange occupational health and safety information. Specifically, this document describes the exchange between ESH-2 and ESH-5 (6) programs designed to meet regulatory requirements such as DOE

Orders 5480.8A (1), 5480.10 (2), 5480.4, and 5483.1A (3); DOE referenced standards and documents; and the 29 CFR Parts 1904, 1910, and 1926 (8).

SCOPE

All exchange of occupational health and safety information between ESH-2 and ESH-5 (6) is subject to this document.

REQUIREMENTS

An active interface between ESH-2 and ESH-5 is a requirement of an effective Occupational Safety and Health Program. Requirements are established in DOE 5480.10 "Contractor Industrial Hygiene Program" (2) and DOE 5480.8A (1) "Contractor Occupational Medicine Program". Additionally, this interface is required by Contract No. W-7405-ENG-36 between the University of California and the Department of Energy (DOE). This contract includes the following provision.

Using a risk-based approach, the Laboratory will develop and implement a site-wide exposure assessment and monitoring plan to characterize employee exposure to hazardous chemicals, physical agents (except ionizing radiation) and biological agents. The plan will be developed by April 1, 1995. Monitoring data from implementing the plan will be provided to the medical staff who will utilize the data in the health evaluation of employees. Baseline data will be collected during FY95. Continuous quality improvement will be based on a peer review (See Appendix A) of a random sample of employee medical charts to evaluate the interaction between the Industrial Hygiene and Medical groups.

RESPONSIBILITIES

Occupational Medicine Group (ESH-2)

ESH-2 is responsible for the following actions to maintain an effective working relationship with ESH-5.

Participating on the Occupational Medicine/Industrial Hygiene Coordinating Committee.

Participating in joint workplace health surveys with ESH-5.

Referring significant events arising from day to day occupational medicine activities to ESH-5 where these activities may have an impact on the Industrial Hygiene Program. Examples include requests for a site review to evaluate potential exposure to toxic materials, high noise levels, or ergonomic stress factors.

Participating in occupational health related information exchange with ESH-5 through effective database management and internal awareness programs.

Teaming with ESH-5 assigned IH staff on programs of mutual interest including, but not limited to, injury and illness investigations, hearing conservation, reproductive hazards , carcinogen use, respiratory protection, ergonomics, medical surveillance, and bioassays.

Conducting medical surveillance for employees identified by ESH-5 for specialty programs to include OSHA regulated chemicals. Examples include lead, beryllium, asbestos, and cadmium.

Team with ESH-5 in the evaluation, on a case by case basis, employees with chemical related illness (sensitizers - eg. formaldehyde, MDI and TDI).

Industrial Hygiene and Safety Group (ESH-5)

ESH-5 is responsible for the following actions to maintain an effective working relationship with ESH-2.

Participating on the Occupational Medicine/Industrial Hygiene Coordinating Committee.

Participating in joint workplace health surveys with ESH-2.

Referring significant events arising from day to day industrial hygiene activities to ESH-2 where these activities may have an impact on the Occupational Medicine Program. Examples include referral of suspected occupational illnesses noted during workplace inspections, reevaluation on an individual's respirator use qualifications, notification of potential exposures to hazardous agent and reporting any worksite exposures measurements in excess of OSHA/DOE permissible exposure/limits, or those levels triggering medical surveillance.

Participating in occupational health related information exchange with ESH-2 through effective database management and internal training programs.

Teaming with ESH-2 assigned medical staff on programs of mutual interest including, but not limited to, injury and illness investigations, hearing conservation, reproductive hazards, carcinogen use, respiratory protection, ergonomics, medical surveillance, and bioassays.

Identifying employees for medical surveillance for specialty programs to include OSHA regulated chemicals. Examples include lead, beryllium, asbestos, and cadmium.

Team with ESH-2 in the evaluation, on a case by case basis, of employees with chemical related illness (sensitizers, eg. formaldehyde, MDI and TDI).

RESOURCE REQUIREMENTS

Maintenance of a strong interface between ESH-2 and ESH-5 requires adequate resources for implementation of program elements common to both groups. These resource requirements will be explained in detail in future documents.

IMPLEMENTATION

ESH-2 and ESH-5 coordinate efforts in virtually all aspects of occupational health and safety. This interaction occurs on both a regularly scheduled basis and on an ad-hoc basis as issues arise. The primary interactions between the two Groups are described below.

Occupational Medicine/Industrial Hygiene Coordination

The ESH-2 and ESH-5 management meet at least once per quarter and additionally as necessary to consider and acting upon important issues related to both group's responsibilities in Occupational Health at the Laboratory and to provide for continuous quality improvement in Occupational Health programs.

The committee tracks the progress made in implementing or improving Occupational Health programs, provides management support to resolving outstanding issues, and considers the impact of new Occupational Health regulations on Laboratory operations.

The committee documents each meeting and identifies action items for follow up.

Program Implementation and Referral Activities

Both ESH-2 and ESH-5 are involved in ongoing implementation of Occupational Safety and Health program elements. These elements include, but are not limited to, the emergency management program, respiratory protection program, hearing conservation program, occupational health training programs, reproductive hazards program, carcinogen program, medical surveillance, and illness and injury prevention.

Through implementation of these program elements, referrals are made from one group to the other. For example, when industrial hygienists are conducting a workplace evaluation and note complaints of dermatitis, ESH-

2 is notified for employee examination. Alternately, when a potential noise related hearing loss is documented by a physician, the event is referred to ESH-5 for an evaluation of potential high noise exposure in the workplace. These type of interactions occur on an almost daily basis, and ESH-2 and ESH-5 work together closely until the situation is resolved. Additional examples of this day to day interaction are shown in the program elements descriptions discussed in Section 7.6.

Workplace Evaluation and Systematic Prioritization

ESH-5 has taken the lead in conducting detailed workplace evaluations to identify and prioritize potential hazards through routine surveys and the Health Hazard Assessment (HHA). Through the HHA process, workplaces are systematically evaluated and can be ranked by the degree of potential hazard through a weighting of variables such as toxicity, amount of material used, frequency and duration of use, and if controls are in-place. A graded approach can then be employed to focus follow-up actions to high hazard areas.

ESH-2 and ESH-5 use this process to systematically select areas where joint evaluations are conducted by Occupational Medical and industrial hygiene specialists. In this manner, the expertise of both groups are applied to potential high hazard areas and an opportunity for cross-training and information exchange is achieved.

Information Exchange

ESH-2 and ESH-5 use several methods to facilitate information exchange in the occupational health area. The information exchange occurs primarily through the use of automated databases and internal cross-training of personnel.

There are automated databases for health hazard assessments, personal and area air sampling, location of specific chemical inventories, confined space locations, and nonionizing radiation sources. These databases are kept current through joint efforts of the two groups and are used in daily and periodic tasks performed by personnel in both groups.

Internal information sharing is conducted through seminars where professionals from either group are assigned a topic, and members of both groups are encouraged to attend. This provides for a greater understanding of the roles, capabilities, and interactions of the two groups in the overall occupational health program. Cross-training sessions address topics such as audiometric exams, HHA's, and database use. Employee rosters and agendas for cross-training classes are maintained.

Continuous Quality Improvement

ESH-2 and ESH-5 use the Laboratory's Continuous Quality Improvement Program to address specific topics both within and between the Groups. Continuous Quality Improvement practices are applied to each group's programs using available quality assurance and quality control professionals.

Continuous Quality Improvement Committees may also be established depending on the importance and need for improvement in a given area. Any employee of either group may bring up the potential need for a Continuous Quality Improvement Committee to Laboratory Management. ESH-2 and ESH-5 both support this process and have provided personnel to serve on Continuous Quality Improvement Committees such as the Respiratory Protection Team.

Individual Program Elements

The effective interface between ESH-2 and ESH-5 is evidenced by the activities of both groups in the following program elements.

Health Hazard Assessment (HHA) Program.

ESH-5 is primarily responsible for conducting and documenting the HHA (see Section 7.3) and managing the data in the HHA database. ESH-2 is a user of the information in the HHA database to assist with specific medical surveillance programs. The ESH-2 and ESH-5 interface occurs in the following areas.

Joint participation in workplace hazard assessments.

Joint use of HHA information to conduct prioritized workplace surveys

ESH-2 uses the HHA information when conducting pre-employment physical examinations, routine medical monitoring and surveillance, and injury and illness investigations

Biosafety Program

The Laboratory Biosafety Program is described in the Environment, Safety and Health Manual, Committee Charter "Biosafety Committee". The Biosafety Committee meets twice per year to provide peer review and approval of proposed Laboratory work involving the use of infectious biological agents and recombinant DNA. Both ESH-2 and ESH-5 participate in the Biosafety Committee to coordinate the industrial hygiene and medical aspects of biosafety.

ESH-2 and ESH-5 have worked together to implement exposure control plans for certain work areas and to develop training programs related to potential exposure to bloodborne pathogens (29 CFR 1910.1030)(7).

The primary interface between the groups in the biosafety area includes the following.

Coordination of Industrial Hygiene and Occupational Medicine biosafety concerns through joint participation in the Biosafety Committee

Development and implementation of bloodborne pathogen exposure control plans

Participation in the development and periodic review of bloodborne pathogen training programs

On a case by case basis, perform joint investigations of bloodborne pathogen incidents or processes

Respiratory Protection Program

The Laboratory Respiratory Protection Program is described in AR 12-2 and the LANL Draft Respiratory Protection Program Manual. ESH-5 is primarily responsible for assisting line management in the selection, use, and evaluation of respiratory protective equipment. ESH-2 is primarily responsible for medical qualification of respiratory protection users.

The primary interface between the two groups include:

Referral of new respirator users from ESH-5 to ESH-2 when identified during workplace surveys

Referral of new respirator users from ESH-2 to ESH-5 when identified during initial or periodic medical examinations

Joint annual evaluation of the effectiveness of the Respiratory Protection Program

Annual comparison of employee listings maintained by ESH-5 for fit-testing, and listings maintained by ESH-2 for medical qualification to ensure that an accurate record is maintained of personnel in the program

ESH-2 referral to ESH-5 of individual respirator user limitations or work restrictions

Hearing Conservation Program

The Laboratory Hearing Conservation Program is described in AR 8-2. ESH-5 performs noise level measurements, hazard investigations and design review for new noise sources. ESH-2 performs medical surveillance and audiometric exams for personnel included in the Hearing Conservation Program. The primary interface between the two groups include the following.

Periodic joint evaluation of the effectiveness of the Hearing Conservation Program

Referral of events where deterioration of hearing acuity is detected through audiometric testing from ESH-2 to ESH-5 for follow-up action. ESH-5 performs a site evaluation and provides the results to ESH-2 for completion of the evaluation. Upon determination of possible workplace noise associated loss, ESH-2 and ESH-5 collaborate on the application of engineering and/or administrative controls.

ESH-5 provides noise monitoring results to ESH-2 for inclusion into the employee medical record and for administration of the audiometric testing program

Development and joint review of the Hearing Conservation Program training materials and courses

Ergonomics

Both ESH-2 and ESH-5 participate in the Laboratory Ergonomics Committee. The Committee duties are described in the Ergonomics Committee Charter. ESH-5 is responsible for workplace evaluations and control measure recommendations concerning potential ergonomics hazards.

The ESH-2 and ESH-5 interface in the ergonomics area includes the following.

Joint participation in the Laboratory Ergonomics Committee to address potential industrial hygiene and occupational medicine concerns

ESH-2 referral to ESH-5 of potential ergonomics hazards in the workplace for evaluation

On a case by case basis, joint workplace evaluations of ergonomic related injuries or illnesses

Laser Safety and Non-Ionizing Radiation

The Laboratory has established a Laser Safety Committee as described in the Environment, Safety and Health Manual Committee Charter "Laser Safety Committee". The Laboratory Laser Safety Program is described in AR 5-2. ESH-5 is responsible for evaluating laser hazard precautions and for investigating injuries related to lasers. ESH-2 is responsible for performing eye examinations related to laser use.

The Laboratory Non-Ionizing Radiation Program is described in AR5-1. ESH-2 performs medical examinations on a case by case basis.

The two groups interface in referral, investigation, and follow-up of laser and non-ionizing radiation related injuries.

Hazardous Waste Operations and Emergency Response

The Laboratory program is described in the Draft Hazardous Waste Operations and Emergency Response Program Manual. ESH-5 is primarily responsible for supporting all Industrial Hygiene Program areas related to this work including identification of hazards, air sampling, exposure assessment, personal protective equipment specification, and control measure development. ESH-2 is responsible for development of medical surveillance programs and medical support services for this work.

The groups primarily interface in the following areas.

Joint occupational health workplace evaluations of high priority operations

Joint effort to determine the appropriate level of health and safety program participation and to categorize hazardous waste worker training and medical surveillance requirements. This effort is conducted through the use of the hazardous waste or emergency response worker surveillance questionnaire. This effort is to meet requirements in 29 CFR 1910.120 (7)

On a case by case basis, perform joint oversight of contractor health and safety programs for environmental restoration and waste management activities

ESH-2 and ESH-5 work together in preplanning and prearrangement of Occupational Health services during emergencies by participating in the Emergency Management Program.

ESH-5 provides support to ESH-2 during emergencies by providing industrial hygiene support to the medical facility, information on chemical toxicity, MSDSs, and respiratory protection.

Carcinogen Program

The Laboratory Carcinogen Program is described in both Laboratory Standard (LS) 106-01.0, Chemical Hygiene Plan, and LS 106-03.0, Carcinogen Use. ESH-5's responsibilities primarily include evaluating workplace conditions and work practices, conducting exposure monitoring and sampling, evaluating the adequacy of control measures, and notifying ESH-2 of affected employees for inclusion in medical surveillance. ESH-2 responsibilities in this program are primarily to determine and implement the medical surveillance programs for carcinogen users.

The groups primarily interface in the following areas.

Joint evaluation of high hazard carcinogen use areas for potential occupational health concerns

Joint review of Standard Operating Procedures for carcinogen use areas

Joint identification of personnel to be included in the carcinogen use program and periodic comparison of personnel listings

Hazardous Materials Use

Specialty programs for specific hazardous materials use including asbestos, beryllium, lead, and cadmium have been implemented. ESH-2's responsibilities under these programs are to design and implement medical surveillance programs. ESH-5's responsibilities primarily include evaluating workplace conditions and work practices, conducting exposure monitoring and sampling, evaluating the adequacy of control measures, and notifying ESH-2 of affected employees for inclusion in medical surveillance.

The two groups primarily interface in the following areas.

Providing occupational health input for implementation or revision to hazardous materials use programs

Conducting joint surveys of high hazard work areas involving use of specified hazardous materials

Conducting periodic review of hazardous materials use programs to ensure compliance with industrial hygiene and medical requirements

Coordination in the maintenance of lists of employees who are qualified to work with specified hazardous materials

Personal and Area Workplace Sampling Program

SH-5 conducts personal and area sampling to characterize individual employee exposure to chemical, physical, and biological agents. This information is maintained on a database and, along with the HHA, can be accessed by ESH-2 prior to performing medical examinations for individual employees.

ESH-2 and ESH-5 interface in the following areas.

ESH-2 and ESH-5 both use personal and area sampling data on-line to coordinate occupational health requirements

ESH-5 coordinates with ESH-2 to maintain personal and area sampling data in a format useable by both groups

ESH-5 provides copies of memos concerning personal sampling information to ESH-2 and provides interpretation of personal sampling data as requested by ESH-2. (Provisions for privacy protection are in place and followed)

Chemical Inventory and Toxicology

ESH-5 has implemented an extensive database management program for occupational health data concerning potentially hazardous chemicals and physical agents. ESH-2 maintains databases for medical surveillance programs and employee health status monitoring. Most data is either personal computer based or operates on the ESH Division Office (DO) VAX in the Oracle environment and can be used by both groups. Due to the confidential nature of most ESH-2 medical data, it is not accessible by ESH-5.

The databases jointly used and accessible by both groups include the following.

Chemical Inventory. The Automated Chemical Inventory System (ACIS) database tracks the location, using groups, and quantities of chemicals throughout the Laboratory. This database is available on-line to both groups through the ESH-DO VAX.

HHA. The HHA is available to both groups on the ESH-DO VAX for information concerning individual workplaces, including hazards present, employees, hazard control measures, and chemical use.

Sampling and Monitoring. A database of all air sampling, noise, etc. conducted by ESH-5 is maintained on the ESH-DO VAX. This information is used by ESH-2 physicians prior to conducting physical examinations.

Confined Space Locations. ESH-5 maintains a database of confined space locations at the Laboratory on the ESH-DO VAX. This information is available on-line to ESH-2 personnel.

Non-ionizing Radiation Sources. ESH-5 maintains a database of non-ionizing radiation source information by location on the ESH-DO VAX. This information is available to ESH-2 personnel.

Asbestos Inventory. ESH-5 maintains a database of asbestos-containing materials by location on personal computer. This information is not currently accessible by ESH-2 personnel on-line, however customized hard-copy reports can be generated upon request.

Pregnancy Consultation

Employee procedures for reporting pregnancies or receiving consultation regarding contemplated pregnancies are outlined in AR 2-1. ESH-2 is responsible for initial coordination with the employee and recommending temporary work restrictions if necessary. ESH-5 is responsible for conducting workplace evaluations for potential reproductive hazards.

The primary interface between ESH-2 and ESH-5 include the following.

ESH-2 referral of cases to ESH-5 for workplace evaluations

Joint ESH-2 and ESH-5 workplace evaluation of selected work areas for potential workplace reproductive hazards

Occupational Injury or Illness Reporting and Investigation

The Laboratory program is described in AR 1-1, Accident and Occurrence Reporting. For occupational illness or injury, ESH-2 is responsible for initial documentation and reporting. ESH-5 is responsible for determining the need for additional investigation and reporting.

The two groups primarily interface in the following areas.

Referral from ESH-2 to ESH-5 of all reported injuries and illnesses

Joint investigation of priority injuries or illnesses with medical and industrial hygiene implications

Joint implementation of preventive measures when lessons learned from an injury or illness affect both medical and industrial hygiene programs

Control Measures

The Laboratory has a program to implement adequate control measures for potentially hazardous work. Controls are selected that will eliminate or reduce the hazard to manageable levels using the following hierarchy.

Substitution

Engineering Controls

Safe Work Practices or Procedures

Administrative Controls

Personal Protective Equipment

ESH-5 maintains the lead role in recommending and evaluating the effectiveness of control measures for industrial hygiene hazards. ESH-5 interfaces with ESH-2 concerning implementation of control measures that may impact individual employee health status. This occurs most frequently when implementing the use of respiratory protection and for administrative controls such as job rotation.

MEASURES OF PERFORMANCE

The quality of the interface between the Occupational Medicine Group and the Industrial Hygiene and Safety Group in itself is a measure of performance of the overall Environment, Safety and Health Program at the Laboratory according to contract provisions. The following indicators may be used to measure the quality of this interaction.

Trend and analysis of occupational illnesses and injuries at the Laboratory

Workplace inspections conducted jointly by the Occupational Medicine and Industrial Hygiene and Safety Group

Results of external audits related to the interaction between the Occupational Medicine and Industrial Hygiene and Safety Groups

Interface activities for special medical surveillance programs such as hearing conservation and beryllium.

Focus on highest, more common accident types: strains/sprains; repeative trauma; and contusions/lacerations.

REFERENCES

1. DOE 5480.8A "Contractor Occupational Medicine Program"
2. DOE 5480.10 "Contractor Industrial Hygiene Program"
3. DOE 5483.1A "Occupational Safety and Health Program..."
4. Draft DOE 5483.XX "Occupational Safety and Health Program for Contractor Employees"
5. Los Alamos National Laboratory Environment, Safety and Health Manual
6. ESH-5 Operations Manual
7. 29 CFR 1910 & 1926
8. Supplemental Agreement to Contract No. W-7405-ENG-36, Appendix F, Objective Standards of Performance, Section A, Part 1, Environment, Safety, and Health

RESTORATION PROGRAM

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ABSTRACT

The Department of Energy (DOE)/Nevada Environmental Restoration Project is comprised of several subprojects which include many dissimilar activities ranging from remediating radioactively contaminated soils on Alaska's western coast, to decontamination and decommissioning activities at the Nevada Test Site. In order to include all these activities in the same health and safety program, the Environmental Restoration Division of the DOE Nevada Operations Office has developed a tiered approach to its health and safety plans. There is an "umbrella" health and safety plan that provides background details of the program and procedures, which is then integrated with site-specific health and safety plans that cover details such as site hazards, history, and personal protective equipment. Working together, these two plans provide the necessary framework to do work safely.

The site-specific health and safety plan contains all information required by 29 CFR 1910.120, "Hazardous Waste Operations and Emergency Response," in abbreviated form. This "user friendly" format is computerized and is completed by checking boxes, filling in blanks, and making brief statements. The site-specific plan is deliberately designed to exclude details that have no relevance to specific information about an individual location. The site-specific health and safety plan format is deliberately simple and is not easily adaptable to complex operations. Therefore, multiple, task-specific, site-specific health and safety plan forms may need to be compiled to address individual activities for complex operations. However, similar activities at different sites may be grouped under a single, site-specific health and safety plan if the hazards and protective measures are closely related.

This integrated approach has been applied to both large and small federal sites; treatment, storage, and disposal facilities; and commercial hazardous waste operations with equal success. The articulated health and safety plan is a cost-effective solution to the drivers of environmental restoration activities associated with multiple regulatory requirements, complex combinations of large- and small-scope operations, and diverse organizations.

INTRODUCTION

The DOE/Nevada Environmental Restoration Project's Health and Safety Program must integrate numerous regulatory and operational requirements to support a diverse environmental restoration program without reducing the efficiency of operations with overly burdensome requirements. Traditional approaches to health and safety plans were evaluated and subsequently rejected because of the enormity of a task involving a multitude of sites, located from Alaska to Nevada with more than five separate contractor organizations, and because of the urgency to increase efficiency through the streamlining of plan production, review, and dissemination. The product of this effort is the subject of this discussion.

OVERVIEW OF THE DOE/NEVADA ENVIRONMENTAL RESTORATION PROJECT

DOE/Nevada Environmental Restoration Project activities encompass characterization and remediation of inactive sites associated with DOE nuclear testing programs. Sites under this project typically have areas of low-level radioactive or chemical contamination (or both) and are found at the Nevada Test Site, Tonopah Test Range, and Nellis Air Force Range as well as remote locations in Alaska, northern and central Nevada, Colorado, New Mexico, and Mississippi. Many organizations are involved with the DOE/Nevada Environmental Restoration Project, including federal and state agencies in both operational and regulatory roles and multiple contractor organizations. These organizations must all work together with the DOE/Nevada Environmental Restoration Project to accomplish it's goals and objectives. Under these circumstances, it is frequently difficult to ensure efficient, effective communication between the various participants of the Project, and considerable effort has been expended simply to assure all entities, including workers, are kept informed of site hazards and controls.

Most of the activities conducted by the DOE/Nevada Environmental Restoration Project are hazardous waste activities as defined by the Occupational Safety and Health Administration (OSHA) in Title 29 of the Code of Federal Regulations, Part 1910, Section 120 (29CFR1910.120), "Hazardous Waste Operations and Emergency Response" (HAZWOPER), and have been adopted by the Department of Energy in DOE Order 5480.4, Environmental Protection, Safety, and Health Protection Standards. This section of 29 CFR 1910.120 (b) requires a safety and health program for hazardous waste operations, and paragraph 1910.120 (b)(4)(i) requires that a site-specific health and safety plan be produced as well. These requirements were reviewed by the Project in 1992, and the preferred approach was to produce two integrated documents: the DOE/Nevada Environmental Restoration Project Health and Safety Plan and site-specific health and safety plans.

Additional standards are also addressed by the use of this site-specific health and safety plan approach. Hazard communication (29 CFR 1910.1200), construction industry specifics (29CFR1926), and a variety of other standards and requirements such as specific contaminant controls (29 CFR 1910.1000, et al.) and elements of environmental protection (40 CFR) are built into each plan. The site-specific health and safety plans also draw on the requirements and procedures of the DOE Radiological Control Manual, and 10 CFR 835, "Radiological Protection."

The DOE/Nevada Environmental Restoration Project's varied project locations, objectives, contractors, and tasks create a challenge for effective health and safety planning and control. An additional complication arises when the mix of personnel includes those with backgrounds in the hazardous waste industry, radiological hazards only, environmental sciences, and government agencies because of the specialized terms and ideas that often cause confusion when addressed outside those particular fields. As with any communications difficulty, this confusion is assumed to affect planning effectiveness, document peer review efficiency, and possibly actual worker safety due to misunderstood instructions during project operations.

A decision was made early in the development of the DOE/Nevada Environmental Restoration Project Health and Safety Plan to use the document as a standard with a common set of definitions and concepts. To assure clarity, specific definitions of terms were provided. In some cases, such as Occupational Safety and Health Administration (OSHA)

construction standard terms in Title 29 of the Code of Federal Regulations, Part 1926 (29 CFR 1926), a "cross-index" was provided to equivalent terms used by the hazardous waste industry in its Hazardous Waste Operations and Emergency Response standard. Similarly, terms common to both the radiological health and hazardous waste fields, but having different meanings, are provided in a common dictionary.

ENVIRONMENTAL RESTORATION HEALTH AND SAFETY PLAN

Traditional health and safety plans are typically cumbersome documents designed to include all aspects of a health and safety program. The solution of this problem was to develop a brief plan in a standard format, focussed entirely on the particular job at hand.

DOCUMENT PRECEDENCE

The DOE/Nevada Environmental Restoration Project's Health and Safety Program's main objective is to protect workers conducting field operations. To accomplish this and encompass the Project's widely varied activities, a programmatic or "umbrella" health and safety plan is in place that prescribes the minimum procedures to be followed while doing work. Individual subprojects, sites, and/or tasks require site-specific health and safety plans to identify the particular features, hazards, communication methods, and protective measures to be employed at that location. Together, the programmatic health and safety plan and the site-specific health and safety plan compose the health and safety program for a subproject, site, or group of tasks. Figure 1, "DOE/Nevada Environmental Restoration Project Health and Safety Document Hierarchy," shows the relationship of the various components which document the DOE/Nevada Environmental Restoration Project's Health and Safety Program.

Fig. 1

Occasionally, a "full" health and safety plan, rather than the site-specific health and safety plan will be developed for an activity. This all-encompassing plan is the traditional health and safety plan approach mentioned earlier and contains extensive details of training, medical monitoring, and site control in much greater depth than is found in the "fill-in-the-blank" format usually used in the DOE/Nevada Environmental Restoration Project Health and Safety Plan. This traditional format may be required when extensive public scrutiny of the project might be involved or when it is requested by participating organizations, usually when such a plan will be more easily understood during plan review and approval.

CONTENTS OF THE DOE/NEVADA ENVIRONMENTAL RESTORATION PROJECT HEALTH AND SAFETY PLAN

The DOE/Nevada Environmental Restoration Project Health and Safety Plan is the governing document under which all operations are conducted. It describes the entire health and safety program, including such vital ingredients as health and safety responsibilities, hazards assessment, hazard control program, personal protective equipment, and site control. Each of these areas is addressed in the plan so that the requirements are clearly delineated and apply to all Project sites. The following sections are addressed in the plan:

Responsibilities - contains the requirements for each functional area from project manager to site supervisor to oversight organizations for the safety of site workers and describes the relationships between each of the roles. General responsibilities that apply to all personnel are also included.

Hazard Assessment - describes when a hazard assessment is to be done and what must be included/considered/addressed.

Hazard Control Program - includes discussions and descriptions of acceptable and/or required procedures related to general practices such as working with the "buddy system" and how to handle site visitors. This section also contains procedures related to common hazards/mitigation measures that may be encountered while performing work on the DOE/Nevada Environmental Restoration Project. These hazards include ionizing radiation, contracting heat and cold illness, hearing conservation, working in confined spaces, engineering and work practice controls, handling drums and containers, excavating, operating aircraft, and dealing with ordnance materials and other physical hazards as well as lockout/tagout procedures and sanitation, illumination, asbestos, and ergonomics issues.

Personal Protective Equipment - includes discussions on the selection and use of proper personal protective equipment, definitions of levels of protection, hazards assessment, and training.

Site Monitoring - involves topics such as airborne contaminant and personnel monitoring, records keeping, and notification.

Employee Training - discusses general training issues including tailgate safety briefings, material safety data sheets, and health and safety plans. Specific training requirements such as radiation safety, hazardous waste operations and emergency response, supervisor's course, first aid and CPR, requirements for site-specific training, and instructor qualifications are also listed.

Medical Surveillance - includes requirements for physical examinations, record keeping, and injury and illness treatment.

Emergency Procedures - includes general emergency procedures, i.e., responses to worker injury or fire, and notification and documentation requirements.

Site Characterization and Analysis - outlines the requirements for site health and safety evaluations, general site information to be gathered, initial site entry procedures, and hazard communication.

CONTENTS OF SITE-SPECIFIC PLANS

Each site-specific health and safety plan incorporates the applicable requirements from the programmatic health and safety plan and delineates how the requirements will be accomplished for that location's scope of work. General project information is included such as a brief site history, investigative objective/activity description, and a notation about whether the site is regulated under 29 CFR 1910.120. Other sections contained in the site-specific plans include:

Material characteristics

Facility descriptions

Confined space descriptions

Investigation-derived waste handling

Hazard analysis Site control procedures (personal protection requirements, surveillance equipment, monitoring requirements)

Site setup

Decontamination procedures

Potential exposure action levels

Emergency procedures and communications

Exposure symptoms and required actions

Address and specific directions to the nearest hospital (and an alternate, if available)

Personnel information
Accident/injury/near miss checklist, and emergency notification requirements
Contact and phone numbers

Also included in the plan is a section that requires the signature of each site worker acknowledging his/her understanding of the contents of the site-specific health and safety plan.

Site-specific health and safety plans are usually prepared by the organization with primary responsibility for completion of the work to be performed and are reviewed by all participating organizations, which assures complete concurrence and understanding of all participants. The time required for this process varies, but is typically a one- to four-week process, depending on the number of reviewing organizations and complexity of the tasks involved.

A computerized program has been developed to accelerate the production of site-specific health and safety plans. This approach has helped to standardize their format, thereby making them more useable to the employee in the field. After a brief adjustment period, field technicians have become accustomed to the site-specific health and safety plan format and can find and interpret information quickly in the printed document. This Microsoft Windows-based application is designed to fit any platform running Microsoft Windows with compatible functionality and design. As part of the Windows approach, on-line help screens are available on all form fields to help the user in completion of the site-specific health and safety plan. However, given the variety of contractor and government organizations expected to produce these plans under the DOE/Nevada Environmental Restoration Project, a word processor version was also developed. Either format is allowed by the DOE/Nevada Environmental Restoration Project Health and Safety Plan, and while the same basic information must be included, other formats are acceptable as well. This flexibility has allowed the health and safety program to focus on content, rather than form, with a concurrent increase in worker acceptance and understanding.

APPLICATION TO VARIOUS PROJECTS AND SITES

The DOE/Nevada Environmental Restoration Project Health and Safety Plan has been successfully applied to many different operations, i.e., the location of the Salmon test, an underground experiment performed under the Plowshares Program, a program evaluating the peaceful uses of nuclear explosives. Activities on this site included threatened and endangered species surveys; surface water, groundwater, sediment sampling; and biological sampling. Initial evaluation of the project seemed to dictate the use of a "full" health and safety plan, a traditional health and safety plan containing all background information in addition to site-specific information. However, by grouping tasks on location and type, it became apparent the site-specific health and safety plan approach would satisfy site requirements by using multiple "short form" style health and safety plans. Each task plan essentially became a chapter discussing the hazards and protective measures appropriate to that task and was then assembled into a complete site-specific health and safety plan for that location. This approach proved very effective given that each sampling team or personnel from each specific discipline activity could refer to a single, brief document rather than search through a large document for information specific to their tasks.

ASSOCIATED PROGRAMS

Several programs complement the health and safety plans. An integral part of the DOE/Nevada Environmental Restoration Project's Health and Safety Program is the Tailgate Safety Briefing. This briefing serves to reinforce information covered in the site-specific health and safety plan specific to that shift's activities and is conducted at the beginning of each shift or whenever new employees arrive at the job site once work commences. The briefing includes possible contaminants and exposure symptoms, personal protective equipment requirements, hazards, decontamination procedures, and emergency information. Each person entering the site, including visitors, must acknowledge attendance by signing a Tailgate Safety Briefing form.

Examples of other programs that complement the health and safety plans are the Radiological Work Permits and Permit Required Confined Space entry programs. These programs require specific training, documentation, and procedures which are not duplicated in the site-specific health and safety plan or the DOE/Nevada Environmental Restoration Project Health and Safety Plan. Each of these programs is referenced, and a brief description of the key features and requirements of the program is presented. This feature avoids the problems of document size and document control by incorporating these other programs by reference.

USE IN OTHER ENVIRONMENTS

The original basis of the "short form" approach to site-specific health and safety plans grew out of a need to produce such a plan to meet OSHA requirements for environmental emergency response operations. A large, unwieldy site-specific health and safety plan would be impractical to produce in the very short response time required, but the alternative, no plan at all, was both unwise and failed to meet OSHA requirements. The development of a concise, "check box" approach was the solution to this dilemma. The early versions of this approach were used in many rapid-response situations such as chemical spills on federal or state lands with great success.

An additional benefit of this approach was the reduced cost of preparation, which was appealing in the competitive commercial environment in which it was first developed and applied. Commercial hazardous waste operations typically regard the letter of the law as a necessary evil, therefore viewing such cost-savings as reduced health and safety plan production time as a boon to operations' profitability. Within this setting, however, is the possibility of an OSHA audit. A necessary component of completing an audit without incurring a citation is a health and safety plan which meets OSHA requirements. Meeting this requirement effectively dictates a health and safety plan and plan production process which is cost-effective in terms of production time and effort. The site-specific health and safety plan approach used by the DOE/Nevada Environmental Restoration Project provides this effectiveness.

SUMMARY

The format of the health and safety program outlined here provides efficiency in the process from project inception to close-out. The programmatic part of the health and safety plan had only to be developed once and revised annually. It reduces redundancies in documentation and does not repeat information found in other requirements documents, but serves to reference all the requirements in one place. This helps to ensure all requirements are being met in the most efficient manner. Each site-specific health and safety plan contains only the information needed to accomplish the scope of work covered by the document. The

format lends itself to quick production and review, and the workers on a site can easily read and understand the hazards and mitigating measures for the functions they are performing.
The DOE/Nevada Environmental Restoration Project articulated approach has been applied to large and small federal sites; treatment, storage, or disposal facilities; and commercial hazardous waste operations with equal success. The articulated health and safety plan approach is a cost-effective solution to the drivers of environmental restoration activities associated with multiple regulatory requirements, complex combinations of large- and small-scope operations, and diverse organizations.

31-5
PREVENTING ERGONOMICS INJURIES DURING REMEDIAL INVESTIGATIONS

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ABSTRACT
Health and Safety Plans for Remedial Investigations (RI) at hazardous waste sites often focus on chemical hazards and overlook physical hazards. The ergonomic hazards contributing to Carpal Tunnel Syndrome (CTS) and other musculoskeletal disorders (MSDs) are some of the most commonly overlooked physical hazards. A recent case of severe CTS developed while working on a remedial investigation site, was evaluated to determine which CTS risk factors were implicated. From that point, CTS risk factors routinely encountered on RI sites were identified through a series of RI site visits. CTS risk factors identified include the following: decontaminating sample spoons, lowering and raising pumps into wells, a variety of high strength gripping activities, surging wells with pipe and surge blocks, lowering and raising bailers into wells, using hand tools which force the hand into an unnatural position, subjecting the hand to cold, and wearing chemical protective gloves which hinder the motion of the wrist. RI activities examined include soil gas surveys, borehole soil sampling, well development, soil, water or sediment sampling and related RI activities. From this review of RI activities, it appears that CTS is a likely hazard on RI sites. Further work is required to examine the actual occurrence of CTS in RI workers, both reported and unreported cases.

HEALTH AND SAFETY HAZARDS FOR REMEDIAL INVESTIGATIONS

Remedial investigations (RI), as required by the Environmental Protection Agency in the Comprehensive Environmental Response, Compensation, and Liability Act, serve as mechanisms for collecting data for site and waste characterization and for conducting treatability testing as necessary. The field component of the RI may include a wide variety of activities, each presenting their own unique occupational safety hazards.

Field Activities During Remedial Investigations

The investigation team may implement any of a number of sampling methods during a remedial investigation. Site characterization efforts may include soil gas surveys, borehole drilling, well development, containerized materials sampling, and soil, water or sediment sampling.

Hazards of Activities During Remedial Investigations

Field activities during RI present chemical, radiological, biological and physical hazards. Field workers risk exposure to the chemical wastes they are investigating, as well as other chemicals used or stored onsite. Onsite safety staff use direct reading instruments and personnel air

sampling to gauge the degree of airborne chemical hazards. Local exhaust ventilation, remote sampling devices and other engineering controls are integrated with safe work practices and personal protective equipment to protect workers from chemical exposures. Similarly, controls are required by federal law and the locally-implemented Radiation Control Manuals to keep exposures to radiation and radiological contamination As Low As Reasonably Achievable (ALARA). Biohazards may be present in hazardous wastes such as hospital or research facility wastes. More commonly biohazards derive from the natural environment and may include disease-causing organisms (pathogens), and dangerous plants, insects, and other animals. Perhaps the most ubiquitous hazard at a hazardous waste site is presented by physical dangers. Safety issues such as electrical safety, lockout/tagout of hazardous energy sources, the use of fall protection, slipping and tripping hazards, working in confined spaces, use of heavy equipment, and a number of other physical safety issues manifest during an RI. While chemical, radiological and biological hazards are generally addressed in Health and Safety Plans and site-specific worker training, physical hazards are often addressed insufficiently. The ergonomic factors contributing to Carpal Tunnel Syndrome (CTS) and other repetitive strain injuries are some of the most commonly overlooked physical hazards.

CTS FACTORS-GENERAL

Repetitive movement, excessive wrist force or improper wrist position of the hand or fingers cause the tendons to slide against the walls of the carpal tunnel. The carpal tunnel is the channel in the wrist through which the median nerve and several finger flexor tendons run. As the tendons slide against the walls of the carpal tunnel, they may become irritated, causing then to become inflamed and swollen. With no room to expand, the swollen tendons press against the median nerve, causing pain. In OSHA's draft ergonomics standard, OSHA describes five signal risk factors, that is workplace risk factors whose presence is a signal that there is an increased likelihood of work-related MSD's. Those five risk signal risk factors include:

Performance of the same motion or pattern every few seconds for more than (2) hours at a time.

A fixed or awkward work posture (for example, overhead work, twisted or bent back, bent wrist, kneeling, stooping, or squatting) for more than a total of (2) hours.

Use of vibrating or impact tools or equipment for more than a total of (2) hours.

Forceful hand exertions for more than a total of (2) hours.

Unassisted frequent or forceful manual lifting.

Activities which require the hand to perform repetitive bending, twisting or pounding operations may create high risk of CTS. Subjecting the hand to cold, or using gloves which hinder motion or enforce an awkward wrist position may also increase risk. Risks are presented by objects which are difficult to grasp, or when gloves make it difficult to grasp an object. Additionally, powered hand tools may produce high levels of vibration. Poorly designed and bulky gloves can reduce hand strength up to 30%. Finger pinching is more stressful than hand gripping. It takes four to five times as much muscle strength and tendon force to pinch an object than it does to grip it.

REMEDIAL INVESTIGATION CTS RISK FACTORS

Table I summarizes the relationship between remedial investigation activities and CTS risk factors. CTS risk factors exist in most common RI activities including soil gas surveys, borehole soil sampling, well development and groundwater sampling, and sampling of containerized materials, soil, surface water or sediment. In some cases the stressing activity may be limited to a brief interval during the workshift. Some CTS risk factors appear in a variety of RI activities. For example, subjecting the hand to cold in outdoor work on many locations. Wearing chemical protective gloves may hinder the motion of the wrist and fingers, and increases the amount of force required for grasping. Inadequately designed hand tools, used for a number of different activities, may force the wrist into unnatural positions.

Table I

Soil Gas Surveys

Repetitive wrist twisting and excessive wrist force is indicated by the number of actions requiring unscrewing and screwing parts together during a soil gas survey. A single soil gas sample may involve screwing the tip adaptor to the soil gas rod, screwing the hammer cap to the soil gas rod, unscrewing then replacing the hammer cap to screw in a second rod once the first rod has been hammered down, unscrewing the hammer cap to screw on a sample port, then reversing the entire process. The entire process may take as little as 30 minutes, and 6-14 locations may be sampled in a single workshift.

Borehole Soil Sampling

Photograph One shows a commonly utilized sampling apparatus, the split spoon. Repetitive twisting and pounding, as well as excessive wrist force may be associated with use of this sampling apparatus to obtain subsurface soil samples. Subsurface soil sampling with a drill rig generally involves hand screwing the sampling apparatus to the drill rod either by hand or using a pipe wrench, drilling to the desired depth, unscrewing the sampling apparatus, and disassembling the sampling apparatus to obtain the soil sample. This process is repeated for each sample. Decontamination of sample spoons includes unscrewing the end cap or boot, cleaning and reassembling the spoon.

Well Development and Groundwater Sampling

Surging for well development often entails the use of pipe and a surge block, as shown in Photograph Two. When surging is performed by hand, repetitive wrist bending and significant wrist force may be involved. This approach involves the assembly of threaded lightweight piping such as polyvinyl chloride pipe. The pipe is often assembled in five foot sections with total lengths commonly ranging from 20 to 50 feet. A surge block is screwed onto the end of the pipe. The pipe is lowered by hand down the well and then raised and lowered along the screened interval of the well for 10 to 15 minutes at a time. After the surging is completed, the pipe is unscrewed and decontaminated.

Sampling of monitoring wells often involves lowering either a teflon bailer or a pump into the well. Equipment is lowered by line with a hand over hand motion requiring repetitive twisting and bending at the wrist. Photograph Three shows the use of a bailer for groundwater sampling.

Sampling: Containerized Materials, Soil, Surface Water or Sediment

Handling any sample generally involves screwing and unscrewing lids on sample jars, and labeling sample jars. Wearing chemical protective gloves can increase the amount of force required in the hand and fingers to perform these tasks.

While sampling containerized materials appears to involve minimal CTS risk. Removing and replacing bung tops on drums involves forceful finger pinching when a bung wrench is not used. A scenario such as sampling dozens of bung top drums in a workshift might present increased risk. Surface and near-surface soil sampling may include using a pick axe, shovel or scoop. Using a hand auger, as shown in Photograph Four requires repetitive twisting of the wrist. Posthole diggers may also be used. While surface water sampling does not appear to involve any significant CTS risk factors, sediment sampling may. Sediment sampling often involves lowering a sampling device by hand to the sediment to obtain a sample. Whether the sampler is lowered on a rigid pole or flexible line, repetitive wrist bending may be required.

CASE STUDY OF CARPAL TUNNEL SYNDROME (CTS) INJURED WORKER

"Joe" is the pseudonym used here for a worker who developed carpal tunnel syndrome (CTS) apparently while performing field work during an RI. This case is interesting in that many of the CTS risk factors inherent in RI work were all present in this case. Table I specifies that "Joe's" work activities would have subjected his hand and wrist to repetitive twisting, bending and pounding, excessive force, and improper positions. His long workshifts in a very cold environment also could contribute significantly to his injury. Furthermore, wearing both chemical protective gloves and gloves for warmth may have hindered the motion of his wrist and increased his required grasping torque in the performance of his tasks.

Medical History

Shortly after "Joe" was sent to a new field assignment, he noticed that his hands were becoming numb. This gradually worsened and began to keep him up at night with throbbing pain. He had noted much muscle tenderness in association with this discomfort. Upon completing his approximately one month field assignment, he returned home and saw a physician who felt that he had bilateral carpal tunnel syndrome. At this time, "Joe" complained of loss of grip strength and significant loss in coordination. He had difficulty writing, which resulted in severe tingling and burning of the hands. He was subsequently diagnosed with moderate to severe CTS which required surgery on both wrists.

Fig. 1

Fig. 2

Fig. 3

Fig. 4

Work History

"Joe" is a field technician who participates in a wide range of field labor activities during environmental investigations and remediations. This section details "Joe's" work activities during his one month work assignment. Shifts during "Joe's" field assignment were generally twelve to seventeen hours per day. Activities included decontamination of equipment, monitoring well development and groundwater sampling. Equipment decontamination included cleaning bailers, sample pumps and sample spoons. Each spoon assembly weighs approximately 20 pounds. "Joe" was involved in equipment decontamination for at least one to two hours each morning over the course of seven days.

For approximately seven hours each day for an estimated two weeks, "Joe" was involved with another team member in well development. This activity involves surging the well with pipe and surge blocks and then lowering a pump into the well. The pump, lowered into the well by hand, weighed

approximately 10 to 12 pounds, plus the weight of the line. Pumps were lowered from 15 to 50 feet into a well.
Finally, "Joe" participated in groundwater sampling. Sampling of monitoring wells involved lowering either a 1 3/4 " by 36" teflon bailer or a pump into the well.

CONCLUSIONS

From this review of RI activities, it appears that CTS is a likely hazard on RI sites. Further work is required to examine the actual occurrence of CTS in RI workers, both reported and unreported cases. As a corrective action, "Joe's" employer now includes a review of likely ergonomics hazards for pending field activities. Likely ergonomics hazards and controls are described in the site-specific health and safety plan, and discussed at the site orientation training.

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Session 32 -- HYDROLOGIC/HYDROGEOLOGIC CONSIDERATIONS IN RADIOACTIVE WASTE MANAGEMENT
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32-1

CONTROL OF WATER INFILTRATION INTO NEAR SURFACE LLW DISPOSAL UNITSPROGRESS REPORT ON FIELD EXPERIMENTS AT A HUMID REGION SITE, BELTSVILLE, MARYLAND

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ABSTRACT

This study's objective is to assess means for controlling water infiltration through waste disposal unit covers in humid regions. Experimental work is being performed in large-scale lysimeters 21.34 m x 13.72 m x 3.05 m (75 ft 45 ft 10 ft) at Beltsville, Maryland. Results of the assessment are applicable to disposal of low-level radioactive waste (LLW), uranium mill tailings, hazardous waste, and sanitary landfills.

Three kinds of waste disposal unit covers or barriers to water infiltration are being investigated: 1) resistive layer barrier, 2)conductive layer barrier, and 3) bioengineering management. The resistive layer barrier consists of compacted earthen material (e.g., clay). The conductive layer barrier consists of a conductive layer in conjunction with a capillary break. As long as unsaturated flow conditions are maintained, the conductive layer will wick water around the capillary break. Below-grade layered covers such as 1) and 2) will fail if there is appreciable subsidence of the cover, and remedial action for this kind of failure will be difficult. A surface cover, called bioengineering management, is meant to overcome this problem. The bioengineering management surface barrier is easily repairable if damaged by subsidence; therefore, it could be the system of choice under active subsidence conditions. The bioengineering management procedure also has been shown to be effective in dewatering saturated trenches and could be used for remedial action efforts. After cessation of subsidence, that procedure could be replaced by a resistive layer barrier or, perhaps even better, by a resistive layer barrier/conductive layer barrier system. The latter system would then give long-term effective protection against water entry into waste without institutional care.

As mentioned in the preceding paragraph, a bioengineering management cover might well be the cover of choice during the active subsidence phase of a waste disposal unit. Some maintenance is required during that period. Final closure, using geological materials, could follow cessation of subsidence. No further significant maintenance would then be required. If the geological material used is merely a clay barrier to water infiltration, the cover will be "sensitive" to imperfect construction or degradation by penetrating roots. The roots will die and decay, causing markedly increased permeability of the clay with the passage of time. A system using a conductive layer under the clay layer as a water-scavenging system will, in comparison, be "robust." Roots will still degrade the clay layer but will not degrade the scavenging layer. A roothole through the conductive layer will be analogous to a hole through a wick. It will do no significant damage. The combination of a resistive layer with a conductive (scavenging) layer underneath is thus less dependent on perfect construction techniques and will be resistant to damage by root invasion. In the absence of subsidence such a system should function effectively for millennia.

Another very useful application of the resistive layer barrier/conductive layer barrier system would be to protect an earth-mounded concrete bunker disposal unit. In that case, the barrier system would shield the concrete from exposure to flowing water. The resulting stagnant alkaline film of water would tend to protect the concrete from degradation over a long time period. Similarly, a resistive layer barrier/conductive layer barrier system could be used to protect high-level waste. If high-level waste were disposed of in fractured rock, this system could be used to divert possible fracture flow water around the waste.

INTRODUCTION

Infiltration of water into waste is the foremost problem associated with near-surface disposal of low-level radioactive waste (LLW). Up to this time, disposal unit covers have generally been constructed from soil materials. In humid areas, these soil or clay covers have generally proved less than satisfactory; often, the cover itself has served as the principal pathway for water entry into the waste (1). Water infiltrating

to buried wastes, contacting the wastes, then exiting the area can reasonably be expected to be the most important of radionuclide transport agents. Some radionuclides, such as tritium (present as tritiated water), and those present in anionic form or neutral complexes, will essentially move with the flow of water; others, present as cations, will move much more slowly, but all will move to a greater or lesser degree. Clearly then, it is advantageous to reduce water infiltration to buried waste to as low a level as reasonably achievable. It is the purpose of our work to examine and demonstrate various approaches for achieving that goal.

Three kinds of waste disposal unit covers or barriers are being investigated in this work:

1) Resistive Layer Barrier
2) Conductive Layer Barrier
3) Bioengineering Management

The resistive layer barrier is the well-known compacted clay layer and depends on compaction of permeable porous materials to obtain low flow rates. Flow through porous media is described by Darcy's law (2). Investigations on flow through such layers have gone on for over 100 years, so further progress in this area can be expected to be slow.

The conductive layer barrier (1) is a special case of the capillary barrier (3). Use is made of the capillary barrier phenomenon not only to increase the moisture content above an interface, but to divert water away from and around the waste. During such diversion, water is at all times at negative capillary potential or under tension.

This system consists of a porous medium underlaid by a capillary break (rock layer). Infiltration barriers such as a conductive layer barrier or a clay layer barrier (or a combination thereof) must fail if subjected to substantial shearing caused by waste subsidence. Reestablishment of a layered system after subsidence failure is a difficult undertaking and is exacerbated by the increasing complexity of the layered system. The failure potential of in-ground layered systems during the subsidence period argues for development of an easily repairable surface barrier for use during that period. To that end, a procedure called "bioengineering management" was developed (4). The bioengineering management technique utilizes a combination of engineered enhanced run-off and moisture-stressed vegetation growing in an overdraft condition to control deep water percolation through disposal unit covers. An artist's conceptual drawing is shown in Fig. 3, referenced in (5).

EXPERIMENTAL AND DEMONSTRATION

In this section we will discuss experiments being conducted in large-scale lysimeters at a humid region site in Beltsville, Maryland (see Fig. 4), referenced in (5).

Bioengineering Management

In bioengineering management the necessary run-off is provided by features installed at or above the soil surface rather than within the profile. The procedure, described by Schulz et al. (4), was designated bioengineering management. Its principal advantage is that subsidence can easily be managed by relatively simple, inexpensive maintenance of the above-ground features rather than by difficult reconstruction of below-ground layers. It should be noted that, after a length of time sufficient so that the organics have decayed and the waste containers have completely failed, subsidence will cease and a layered system could be then installed which could last over geological time periods.

In essence, the bioengineering management technique utilizes a combination of engineered, enhanced run-off and stressed vegetation in an overdraft condition to control deep water percolation through disposal unit covers. To describe it further: if a waste burial site is selected so that incoming subsurface flow is negligible, then precipitation is the sole source of input water. In a simplified model, that water has three possible fates: 1) evapotranspiration, 2) run-off, and 3) deep percolation. Evapotranspiration has a definite limit, governed by energy input. Ideally, deep percolation should be zero, leaving only the run-off component available for unlimited manipulation. Positive control of run-off becomes difficult with the use of compacted porous media trench caps as the sole barrier to water infiltration. The compacted material tends to become more permeable with the passage of time, due to fractures caused by waste subsidence and from the inexorable process of root growth, followed by death and decay of the roots, thus creating water channels. Evapotranspiration cannot then use all of the infiltrating water, and water percolates downward to the waste. As stated before, evapotranspiration has a theoretical maximum dictated by solar energy input to the system; only run-off remains available for nearly unlimited management. This run-off can be surface or subsurface, as long as it occurs before water reaches the waste.

Surface run-off can be managed to as high as 100% by means of a perfect, leak-proof roof, which is expensive and hard to guarantee. Alternatively, adequate but not total run-off can be engineered rather inexpensively by using an impermeable ground cover over part of the surface to achieve high and controlled levels of run-off. Vegetation planted between areas of impermeable cover will extend over the cover to intercept incoming solar energy to evaporate water. Roots will extend under the cover in all directions to obtain water.

Such a system can be visualized by imagining a supermarket parking lot, where trees are planted in islands, surrounded by concrete curbs, within an extensive paved area. In this case, the trees are maintained in a drought environment due to the small soil surface available for infiltration of precipitation. The paving, along with the curbing around the trees, causes run-off of most precipitation. Aboveground, the tree's branches and leaves extend over the parking lot and intercept incident solar energy. Beneath the surface, the roots, in a drought state, explore outward under the paving for any available water. Utilizing this concept, it should be possible, by combining engineered run-off with vegetation, to maintain the soil profile in a potential overdraft condition on a yearly basis.

Initial investigations of the bioengineering management technique were carried out in lysimeters at Maxey Flats, Kentucky. Results obtained in seasonal 1984-1985 and 1985-1986 were reported by O'Donnell et al. (6). In that work, a fescue grass crop was used with an engineered cover of stainless steel. Following seasonal 1985-1986 the grass cover was removed, a new stainless steel engineered cover was constructed, and Pfitzer junipers were planted in the lysimeters. After the junipers were established, percolation data were again collected in 1988 and reported by Schulz et al. (7). The woody junipers were excellent in preventing deep percolation of water in the lysimeter.

The encouraging initial results obtained in the Maxey Flats lysimeter experiment led to the establishment of a large-scale field demonstration at Beltsville, Maryland (Fig.4), referenced in (5). Figure 5, referenced

in (5), is a photograph of lysimeter 1, bioengineering management, taken in December, 1994, eight years after planting of the Pfitzer junipers. Alternating panels of aluminum and fiberglass were used as the hard cover. These plots, or lysimeters, are 21.3 m (70 ft) long by 12.7 m (45) ft wide, and the bottoms are 3.05 m (10 ft) below grade. Figure 6, referenced in (5), shows a side view of construction details of lysimeters 1 and 2 (bioengineering management). The only difference between the two was the initial water level in the lysimeters. The water level was 90 cm above the bottom of lysimeter 1 and 190 cm above the bottom of lysimeter 2. The water level in the lysimeters simulates the water table in a flooded disposal cell. In addition to the two bioengineered lysimeters, two reference lysimeters (3 and 4) were initially constructed. They were similar to the former, except that they were merely planted with fescue grass. No hard cover was present, but surface slopes were similar to the two bioengineered lysimeters (i.e., a slope of 1:5). Performance data for the reference lysimeters are given in Fig. 7, referenced in (5).

The water level in the two reference plots or trenches (lysimeters 3 and 4) rose until it was near the surface. At that time, water was pumped from the lysimeters to keep them from running over. The graphs of the water tables (i.e., water levels) in the bioengineered plots (lysimeters 1 and 2) show an entirely different story, as evidenced in Fig. 8, referenced in (5). In both cases, the water table was eliminated. It appears that the bioengineering approach could prevent water infiltration to a disposal unit. It also could be used for a remedial action in dewatering existing problem sites such as Maxey Flats.

On February 4, 1988, lysimeter 4 was pumped out to prevent overflow. It was then discontinued as a reference lysimeter and converted to a rock-surfaced, resistive-layer barrier plot. Lysimeters 1 and 2 (bioengineered) and lysimeter 3 have been continued. A summary of run-off, evapotranspiration, and pumping from those three lysimeters is given in Fig. 1.

Fig. 1

Figure 1 shows that there was very little run-off from the grass-covered plot. Most of the precipitation was disposed of, via evapotranspiration, by the fescue crop, but this was not adequate to prevent the rise of the water table. Table I gives the run-off, evapotranspiration, and deep percolation in the bioengineered plots during the past eight years. There was no deep percolation during this period. Until seasonal 1993-1994 the evapotranspiration had been rising annually, probably as a result of the greater vegetative canopy intercepting a greater percentage of the precipitation. In 1988, 1989, 1990, 1991, and 1992 the run-off percentages were 80, 74, 70, 67 and 63, respectively. In 1993, the run-off decreased to 61% of the precipitation. In 1994 the run-off remained the same as in 1993 (8). In 1995 the run-off decreased to 58% of the precipitation. During 1989, the water table was completely eliminated in both plots (Fig. 8), referenced in (5).

Table I

In addition to rainfall, run-off, and evapotranspiration measurements discussed above, neutron-probe soil-moisture measurements have been made continuously to monitor soil moisture changes in all six lysimeters depicted in Fig. 4, referenced in (5). The neutron probe measurements will indicate whether there is a gain or loss of moisture from the soil profile or, perhaps, steady-state situation, where there is little or no

net gain or loss of soil moisture during a year. A steady-state situation with relatively constant-moisture "dry" soil above waste would be highly desirable with a bioengineered cover. There would then be a large safety margin to protect the waste from infiltrating water.

Neutron probe apparatus, as supplied by the manufacturer, is calibrated against moisture measurements in sand. Such calibration is of unknown accuracy when applied to soil measurements. For this reason, the probe was calibrated using the same soil as in the lysimeters. Six hundred and twenty-eight kilograms (1400 lbs) of soil were placed in a weighing lysimeter, and measurements were made over a seven year period. Calibration data obtained using the weighing lysimeter are given in Table II, referenced in (5). The resulting curves, depicting the factory calibration and the weighing lysimeter calibration, are given in Fig. 10, referenced in (5). It is evident that use of the factory calibration on sand would result in a very large error in soil moisture determination.

Results of some neutron probe measurements are shown in Fig. 2 for bioengineered lysimeters 1 and 2. The data are plotted as volumetric moisture content, as a function of soil depth, on specific dates. Only ten widely spaced measurement dates are shown, for clarity. From inspection of the figure it is seen that, at the start of the experiment in July, 1987, the moisture content of the soil increased with depth until the water table was reached, then became constant. By July, 1989, the water table had been eliminated from both lysimeters, and the soil profiles were drying out. However, the soil moisture content, although much lower in the soil profile than in July, 1987, still increased with depth. This same relationship was still evident in October, 1995, although the soil profile had become still drier.

Fig. 2

Figure 12, referenced in (5), shows the moisture content of the soil profiles in lysimeters 1 and 2 at the end of each seasonal year. Following the complete removal of the water tables during the 1987-1989 period, the soil profiles were dried out further during the ensuing years. However, an unanticipated result turned up in lysimeter 1 at the end of seasonal 1993-1994. The moisture content of the soil profile increased slightly. To shed light on that result, the moisture content in the soil profiles at four depths were plotted monthly along with monthly rainfall data (Fig. 3). Here we see seasonal cyclical variations in moisture content in the soil profiles, with peak moisture concentrations occurring in the early spring, following periods of significant rainfall and minimal evapotranspiration. That cycling is both obvious and expected. What was totally unanticipated was the increase in the moisture peaks in lysimter 1 in each of the 3 seasonal years leading to 1994-1995. The increasing amplitudes of the moisture curves did not appear to be a result of rainfall variations, nor were they present in lysimeter 2. The aforementioned trend did not continue during the past year, seasonal 1994-1995.

The results to date indicate that bioengineered closure, as described in this experiment, would maintain the cover over buried waste in a "dry" steady-state condition. This would not only prevent water from percolating down to the waste, but would do so with a large safety factor.

Resistive Layer Barrier

As previously mentioned, on February 4, 1988, lysimeter 4 was pumped out, discontinued as a reference lysimeter, and converted to a rock-surfaced,

resistive-layer barrier plot. The primary reason for constructing that particular cover is the likelihood of such covers being used for uranium mill tailings. An end view of that plot or lysimeter is shown in Fig. 14, referenced in (5). This lysimeter was completed in the fall of 1988, and data collection (measuring performance) is underway. The most important information to be gained here will be the relative weighing of the advantages and disadvantages of rock surface vs. a vegetated surface.
In addition to the UMTRA or rock-surfaced, resistive-layer barrier plot, a vegetated resistive layer barrier plot was constructed. The primary purpose of this plot is for comparative measurements. Essentially, this plot is similar to the rock-surfaced plot except that topsoil replaces the rock layer, and the plot is planted with fescue grass. A diagram of this plot is given in Fig. 15, referenced in (5).
Fig. 3
In Fig. 1, the fate of precipitation in the UMTRA and grass-covered, clay-layer lysimeters is given. There was more than twice as much run-off from the rock-covered plot was as from the grass-covered plot. The data show no deep percolation through the clay layers through seasonal 1993-1994 in either lysimeter and there is little indication as to how much safety margin has been offered. In seasonal 1994-1995, 0.10 cm (0.04 in) passed through the cover in Lysimter 4. Nor is it known how consistently such near-perfect clay barriers would be installed in a routine operation. That remains a problem for future consideration.
Another concern is the possible drying out of clay barriers. If this were to happen, the clay layer would not be as efficient a barrier for preventing radon escape as planned in the UMTRA application. In addition, drying out of the clay layer could lead to cracking, leading to subsequent leakage prior to resealing by wetting. Figure 4 gives the volumetric moisture content of clay in the rock-covered (lysimeter 4) and the grass-covered (lysimeter 6) plots. In no case did the clay layer dry out significantly. On the contrary, in the UMTRA or rock-covered plot, which was devoid of vegetation, there was a slight increase in moisture content with time, suggesting that some leakage of water through the clay layer would occur in the future. That first leakage occurred during the past year, seasonal 1994-1995. Lysimeter 6 has a clay layer and a grass cover. In this case, no increase in moisture content has been observed. On the contrary, to date the moisture content of the clay layer seems to be in a rather steady state, taken over the 7 year period of measurement.
Conductive Layer Barrier
If we consider the case of water flowing downhill in an unsaturated porous medium, we have the case shown in Fig. 17, referenced in (5). The "holes" shown in the diagram could be a rock layer, affording a capillary break or capillary discontinuity (Fig. 18), referenced in (5). Under appropriate conditions, water everywhere in these cross-sections will be under tension, and there will be no leakage. This might then serve as an excellent means of protecting waste by conducting water around the waste. Figure 17, referenced in (5), simulates a conducting porous medium, such as a fine sandy loam soil, lying smoothly on top of a rock layer.
Problems with water flow under saturated conditions could certainly arise where a less than smooth surface ends up being constructed as depicted in Fig. 19, referenced in (5). That is, what happens if imperfections are constructed so that "pockets" of soil extend down into the rock layer? Figure 19 represents that case. Again, there will be no leakage, provided

conditions are such that the water in all parts of the conductive layer remains under tension.

Fig. 4

The big question is, can conditions required to maintain the necessary soil water tension be practically maintained while using this procedure to effectively protect waste disposal units? To answer this question the apparatus schematically depicted in Fig. 20, referenced in (5), was constructed, i.e., a "soil beam." Several miniature soil beams (Fig. 21), referenced in (5), were constructed for use in the laboratory so that a variety of candidate conductive-layer materials could be quickly evaluated.

A number of materials were evaluated using the miniature soil beams. It was quickly established that it would be necessary to construct a resistive layer barrier above the conductive layer barrier to have a practical system. The standard was set that the resistive layer barrier have an easily achievable conductivity of not greater than 10-6 cm/sec. On this basis it was found that material such as fine sandy loam could provide an effective conductive layer barrier, that is, conduct around the waste 100% of water percolating through the resistive layer. However, the measurements showed that such materials would not provide the desired (factor of 10) safety margin.

Further investigations turned up a material, diatomaceous earth, that would fit these requirements. Measurements of tension vs. distance of flow are shown in Fig. 22, referenced in (5).

The results of this experiment in the 137 cm (4.5 ft) long beam suggest that, as long as the flow rate is no greater than 4.2 10-4 cm/sec, the soil water will remain under tension regardless of the soil beam length. These results show that with the use of diatomaceous earth for the conductive layer and following the easily achievable standard set above for the resistive layer, it should be possible to construct a barrier that would allow no water leakage to a waste disposal unit. However, before final selection of the diatomaceous earth as the conductive layer material, we believed it to be prudent to conduct tests in a large-scale soil beam. The large beam, shown in Fig. 23, referenced in (5), has a soil beam length of 6.4 m (21 ft). As shown in Fig. 24, referenced in (5), a matric potential of about -15 to -20 cm of water is maintained over the entire 6.4 m length of the beam when the flow rate does not exceed 3.1 10-4 cm/sec.

The studies carried out in the large soil beam closely confirmed the data obtained in the miniature beam. Accordingly, diatomaceous earth was used as the conductive layer material in the demonstration lysimeter (lysimeter 5). It has been estimated that purchasing and shipping the diatomaceous earth to a job site any place in the United States will add about $0.50 per ft3 of disposed waste. This is over the cost of using locally obtained soil, and based on waste being 3.05 m (10 ft) deep.

After the time-consuming task of selecting the conductive layer material was accomplished, a resistive layer barrier over a conductive layer barrier was constructed in lysimeter 5. It was completed in January, 1990. A local clay from Beltsville, Maryland, the Christiana Clay, was selected as the resistive layer barrier. Testing has shown this material more than meets specifications. A cross-section of the cover system is shown in Fig. 25, referenced in (5).

Performance of this cover is shown in Figs. 1 and 4 (lysimeter 5). Until seasonal 1993-1994 the cover system was 100% effective in preventing

water movement downward through the cover. In seasonal 1993-1994, 0.13 cm (0.05 in) of water passed through the cover to the pan shown in Fig. 25, referenced in (5). In seasonal 1994-1995, 0.18 cm (0.07 in) passed through the cover. Although that amount is an extremely small percentage of the total rainfall, in theory no water should have percolated through the cover to pan. It is possible that the cover system was compromised by the instrumentation installed to measure performance.

Further Studies of Conductive-Layer Materials

For the Beltsville study, diatomaceous earth was selected for the conductive-layer material, based both on performance and cost considerations. Based on these two considerations only, diatomaceous earth would still be the material of choice, particularly since it has a much lower bulk density than sand and is therefore less expensive to ship. However, the engineering properties of sand are better known, thus sand may be more attractive to some installers. Therefore, we have been conducting further studies with various sands. Results of studies of the unsaturated flow characteristics of four different sands are given in Fig. 26, referenced in (5). All these sands exhibit unsaturated flow rates that are about twice that of the diatomaceous earth at any given negative matric potential. The particle size distribution of the four sands is given in Table III, referenced in (5). The mortar sand, for example, had the narrowest particle size range, and the foundry sand had the widest particle size distribution, although the particle size distribution did not have an important effect on the flow rates reported in Fig. 26. The Nevada dune sand and the Kelso dune sands are from large eolian deposits in the Nevada and California deserts, respectively. The Kelso deposit has been mined commercially. This work on the unsaturated flow characteristics of various materials is ongoing. Hydraulic properties will be studied over a larger range of matric potentials, further deposits will be located and investigated, and these investigations will be described in a future report.

APPLICATION

The three procedures described in the Introduction may be used singularly or in combination to protect disposal units from percolating water. The principles apply equally to above-ground or below-ground disposal. For example, a combination of covers 1. and 2., described in the Introduction, could be ideal for a stabilized, shallow land burial facility, whether it is above or below ground; e.g., the subsurface disposal could be in below-ground vaults, and the above-ground disposal units could be earth-mounded concrete bunkers. A combination of a resistive layer over a conductive layer in a concrete bunker or above-ground application is shown in Fig. 27, referenced in (5). The resistive (clay) layer is the primary barrier. The small amount of water passing through the clay layer will be diverted around the concrete bunker by the conductive layer. This cover over the concrete bunker can, in theory, be 100% effective, shielding the bunker from exposure to flowing water. This would result in a film of stagnant alkaline water at the gravel/concrete interface. The presence of this high pH, stagnant water would tend to protect the concrete from degradation over a long period.

The bioengineering concept could be advantageous for either a tumulus or shallow land burial unit that would be likely to exhibit subsidence. If desired, and after subsidence has ceased, a combination of covers 1. and 2. could be constructed with geological materials to give extremely long-

term isolation without further maintenance. Another possible application of a combination of covers 1. and 2. described in the Introduction is shown in Fig. 28, referenced in (5). Here, high-level waste is emplaced in a tunnel excavated in rock. If a fracture were present in the rock, and fracture flow occurred, the combination of a resistive layer and a conductive layer could provide excellent isolation of the waste from flowing water. Figure 29, referenced in (5), depicts an application where only very low flow rates need be protected against (essentially, dropwise fracture flow). Here, the system could be simplified so that only a conductive layer with a capillary break is necessary.

ACKNOWLEDGMENTS

Funding for this project was provided by the Waste Management Branch, Division of Regulatory Applications, Office of Nuclear Regulatory Research. Dr. Edward O'Donnell is the Project Manager, Dr. Sher Bahadur is Chief of the Waste Management Branch, and Dr. Bill M. Morris is Director of the Division of Regulatory Applications.

The authors wish to express their thanks to Lester M. Fujii for his contribution to this study. Furthermore, we are indebted to the support provided this project by the University of Maryland's Experiment Station, Beltsville Unit staff, notably Peter Godwin.

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HYDROGEOLOGIC ANALYSES IN SUPPORT OF THE CONCEPTUAL MODEL FOR THE LANL AREA G LLRW PERFORMANCE ASSESSMENT

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ABSTRACT

The Los Alamos National Laboratory (LANL) low level radioactive waste (LLRW) disposal facility at Area G is currently completing a draft of the site Performance Assessment (PA) as required per DOE orders. The site is located on a narrow mesa top of volcanic tuff where possible mechanisms of subsurface transport include vapor movement with vapor-liquid phase coupling, transport in fractures, and other as yet unresolved subsurface dynamics. The large depth to the water table (~250m) combined with uncertain hydrogeologic data below 100m and the complex stratigraphy of layered volcanic flows and ash provide for a challenging analysis. Results from previous field studies have estimated a range in recharge rate up to 1 cm/yr. Recent estimates of unsaturated hydraulic conductivity for each stratigraphic layer under a unit gradient assumption show a wide range in recharge rate of 10-4 to 1 cm/yr depending upon location. Numerical computations show that a single net infiltration rate at the mesa surface does not match the moisture profile in each stratigraphic layer simultaneously, suggesting local source or sink terms possibly due to surface connected porous regions. The best fit to field data at deeper stratigraphic layers occurs for a net infiltration of about 0.1 cm/yr. Surface moisture data and vertical moisture profiles suggest a wide range in infiltration at the near surface. Transients are hypothesized to play an important role in fracture infiltration and evaporation, but are seen in analyses to dampen rapidly within the mesa tuff matrix to a steady state condition.

A recent detailed analysis evaluated liquid phase vertical moisture flux, based on moisture profiles in several boreholes and van Genuchten fits to the hydraulic properties for each of the stratigraphic units. Results show a near surface infiltration region averages 8m deep, below which is a dry, low moisture content, and low flux region, where liquid phase recharge averages to zero. Analysis shows this low flux region is dominated by vapor movement. Field data from tritium diffusion studies, from pressure fluctuation attenuation studies, and from comparisons of in-situ and core sample permeabilities indicate that the vapor diffusion is enhanced above that expected in the matrix and is presumably due to enhanced flow through the fractures. Below this dry region within the mesa, near the interface of the vitrified and devitrified tuff units and also coincident with the elevation of the adjacent canyon floor, is a moisture spike which analyses show corresponds to a moisture source. This may indicate a narrow region with unique hydrologic properties, or horizontal moisture transport from the canyons. However, the likely physical explanation is seasonal transient infiltration through surface-connected fractures. This anomalous region is being investigated in current field studies, because it is critical in understanding the moisture flux which continues to deeper regions through the unsaturated zone.

INTRODUCTION

The Los Alamos National Laboratory (LANL) low-level radioactive waste (LLRW) disposal facility at Area G has completed a preliminary draft of the site Performance Assessment (PA) (1) as required per USDOE orders.

The final draft PA becomes the technical basis for authorization and management of disposal operations at the active disposal site.

The Area G disposal facility is located on the top of a narrow finger-like mesa composed of volcanic tuff (Bandelier Tuff), deposited in stratigraphic layers of ash and solidified volcanic flows. Waste disposed at Area G is placed into pits excavated in the volcanic tuff, crushed in place, and backfilled with the native crushed tuff to about 30% waste package and 70% tuff by volume. These disposal operations are evolving to minimize future disposal volume, and to assure stability of emplaced waste.

The unresolved subsurface dynamics of the mesa top site location and the adjacent canyons with the large depth to the water table (~250m) provide for a challenging analysis. Uncertain hydrogeologic data below 100m have led to preliminary analyses focusing on the near surface hydrology in the mesa-canyon system. Preliminary review by the USDOE Peer Review Panel indicated that the proposed subsurface transport conceptual model (2) was not fully integrated with the numerical models (3) used to evaluate the site, especially in defining the transition from significant transient events to the steady state analysis and in defining the possible role of vapor transport. This report provides additional data review and analyses to address these issues.

DATA REVIEW

Area G site hydrology was summarized (4) as of 1987, based primarily on field studies by Bendix (5). Several boreholes and core samples were used to characterize moisture verses depth and detailed hydrologic parameters. Moisture profiles by in-situ neutron probe measurements were seen to be independent of time below depths of about 2m. This was confirmed in recent measurements in a borehole into the crushed tuff backfill of an active Area G disposal unit (6). These results suggest transients in moisture content play a negligible role at depth.

A recent review of the site geohydrologic data (7) including recent permeability work has been completed as part of the Area G PA work. The layered stratigraphy within and beneath the disposal site mesa is illustrated in Fig. 1. Interpretation of the site geohydrologic data (2) identified several field observations which may be important in subsurface transport and postulated that transient infiltration into and long-term dryingfrom fractures play a significant role.

Fig. 1

In-situ air permeability measured by a borehole packer-isolation method was seen to be 4 to 20 times greater than the permeability of recovered core samples (4,5). Similar results were obtained recently comparing core sample air permeability in borehole (G-5) (8) to in-situ permeability measured in the same borehole (9). Attenuation with depth of atmospheric pressure oscillations (10) is consistent with permeabilities much larger than that measured on intact core samples. Subsurface diffusion of a tritium plume at Area G has been compared with numerical results to determine an in-situ diffusion coefficient (11).

Attributing the migration to vapor diffusion (liquid phase transport is negligible under the field conditions as will be discussed) gives a vapor diffusion coefficient which is more than one order of magnitude larger than that expected in the volcanic tuff matrix. These results indicate that fractures in the mesa top stratigraphic layers play a major role in determining the air permeability and effective vapor diffusion in the mesa.

ANALYSES

Unsaturated hydraulic conductivity and water characteristic curves were determined from van Genuchten fits (12) to matric potential data and from saturated conductivity data for several core samples in each of the near surface stratigraphic units at Area G disposal facility (13,14). Detailed comparisons of unsaturated conductivities predicted in this manner with values measured by the unsaturated flow apparatus (UFA) centrifuge method (15) show good agreement in only about half the cases and indicate a large variability and uncertainty in the transport characterization (16,17).

The stratigraphic unit averages of van Genuchten fits were summarized for the Area G PA work (7) (Fig. 2) over a range in moisture content up to 25% where the fits are accurately approximated by a straight line on a log scale. Assuming unit hydraulic gradients in the field, these curves correspond to recharge rates of 10-4 to 1. cm/yr in the range of in-situ moisture content from about 1% to about 10%. If there are no local source or sink terms, then the recharge or flux through the strata must be constant, which would result in higher moisture contents in Units 1b and in the Cerro Toledo (see Fig. 1, results labeled T-CT in Fig. 2, and labeled C in Fig. 3). This trend is consistent with field observations, but an accurate match of moisture content to field observations in each unit simultaneously is not possible as shown in the following.

Fig. 2

Fig. 3

This issue is examined in a comparison between numerical results and field data shown in Fig. 3. Steady state moisture contents are computed for several values of net infiltration at the mesa top, shown along a vertical line through the center of the mesa in a 2-D model with accurate stratigraphic cross-sections. The range of field data is shown as the darkened area. The canyon floors adjacent to the mesa coincide closely with the interface between Units 1b and 1a at Area G. The best agreement with field data occurs for downward vertical flux values of 0 to 0.001 cm/yr in the top two units, about 0.1 cm/yr through Units 1b and 1a, and as much as 1 cm/yr in the Otowi. Recent simulations show the higher moisture contents in the Otowi can be matched with lower flux values (0.1 cm/yr) when hydraulic properties of deeper strata (subject to large uncertainty) are included in the numerical model.

The steady saturation profile calculated for any single infiltration rate does not match all the in-situ saturation data gathered at the site. Saturations within the mesa top (Units 2b and 2a) are extremely low in field data. Saturations below the base of the mesa (Unit 1b and below) are higher. Several factors which may contribute to this discrepancy include: 1) infiltration from canyons or from deep surface-connected fractures resulting in higher saturations below the base of the mesa, 2) evaporation from the mesa sides and from fractures resulting in very low saturations within the mesa top, 3) uncertainty and heterogeneity in the hydrologic transport parameters.

For preliminary PA modeling work through all of the stratigraphic units, a rate of 0.1 cm/yr was chosen as the best 'fit' because there is no detailed knowledge of local source or sink terms to supply to the numerical effort. However, the comparison summarized in Fig. 3 suggests a complex situation with local (elevation dependent) source and sink terms to the recharge rate.

The source and sink terms in the underlying strata were examined in detail in a recent study (18). The vertical flux and the vertical component of its divergence, interpreted as a local source or sink term, were evaluated directly from vertical moisture profiles using the stratigraphic unit-averaged van Genuchten fits to determine the hydraulic transport parameters for matric potential and for unsaturated hydraulic conductivity, and thus the vertical flux as a function of local moisture content. Results are discussed in detail for data from a dozen boreholes at Area G (18), for example, as shown for one 'typical' hole in Fig. 4.
Fig. 4
Figure 4A (top) shows the moisture profile in a borehole located near the center of the mesa top disposal facility. The inferred vertical flux and an effective source term to the local recharge rate labeled 'source' in Fig. 4B (middle), evaluated as the difference in vertical flux between adjacent points, show nearly zero flux or source throughout the range in depth from 7m to 25m. Near the surface there is a net downward flux (negative value of flux) of about 1 cm/yr, but there is also a net sink (negative source term) in this region, presumably corresponding to an evaporative loss to the surface or surface-connected fractures. At ~30m depth there is an apparent source of moisture with vertical moisture movement and evaporation (net sink) away from that source location. Similar results are seen in all the boreholes at this depth, however, quantitatively, there is a large variability with location (18).
The near surface infiltration region also has a large variability with location dependent upon the local disposal operations (18). Modeling of deep percolation from the disposal pits is sensitive to assumed rooting depths, leaf area index, and other parameters which govern the detailed surface water balance (19). The large variability in surface flux is consistent with the broad range measured in surface soil moisture content (20).
The source term indicated by the vertical difference in moisture flux can also be normalized per unit depth which allows the source to be expressed as an inverse time scale. This time scale can be interpreted as a characteristic time over which the moisture flux (labeled St-inv in Fig. 4C) is changing locally due to liquid phase movement, or, the time scale over which the moisture content (labeled St-inv*vol%) is changing if the entire vertical component of the divergence of flux is interpreted as a time dependent effect.
In Fig. 4C (bottom) the source term time scale for the moisture content (St-inv*vol%) is seen to be at least one year at all locations, and 100-1000 years over most of the 'low fluxregion' identified from Fig. 4B. In the near surface infiltration region, the observed characteristic time scale of 6-10 years implies deep penetration only once every several years, consistent with predictions by detailed surface water balance modeling studies (19). Interestingly, the apparent source at the ~30m horizon has the minimum characteristic source time scale of about 1-3 years, consistent with significant infiltration once every few years. The short times near depth 30m may be an artifact of the unresolved stratigraphic hydrologic properties near the 1b-1a interface or may indicate a significant local source. Field studies are currently underway to resolve the matric properties within this region, which should allow a determination as to whether the apparent source term is real.
If the apparent source of moisture proves to be real, it may indicate moisture influx by horizontal transport from the adjacent canyons or from

another fast flow path. A possible explanation is that this elevation is the bottom of a network of surface-connected fractures which allow transient infiltration during intense and infrequent storm events. The magnitude of the source term as a characteristic time scale of about 1-3 years is consistent with significant recharge once every few years.

DISCUSSION

Transients

Field studies (4,5) and analyses (18) indicate that transients in moisture content within the Bandelier Tuff matrix are dampened over short distances of 1-2m. Transients are expected to play a negligible role in moisture flux through the mesa tuff matrix compared to steady state flux. Transients can be significant if fast paths exist, e.g., near elements of surface-connected macro-porosity which can include surface-connected fractures, horizontal strata of high permeability (as proposed may occur at the surge beds between Units 2a and 2b) or other possible 'macro-pores'. An example of this may be the moist horizon observed near the vitrified-devitrified interface at ~30m depth. These transients average to an effective steady state moisture flux and moisture source term for long time scale numerical simulations.

Vapor Phase Flow

Vapor phase movement may be enhanced by diffusion due to barometric pumping or temperature fluctuations especially in strata of high permeability, however, the net convective movement of air through the tuff matrix is negligible. Evidence suggests convective movement through surface-connected features is significant and may influence the matrix flow. As such, diffusion of vapor phase contaminants is related to empirically determined effective diffusion coefficients which include possible barometric pumping effects.

Moisture profiles are expected to be driven by vapor diffusive movement when the moisture content falls below a specific value, which is about 2-5% for the hydrologic properties of the Bandelier Tuff under Area G. Here, in the low flux, dry region of the mesa interior, vapor phase loss through evaporation to surface connected fractures may contribute as a local sink term to drying regions below the moisture content expected to support liquid phase movement. It may be possible to model this as a sink term in the liquid phase matrix transport equations. To model transient dynamics between the fractures and the matrix, more sophisticated source/sink terms need to be considered such as that afforded by a dual porosity, dual permeability numerical model which can account for rapid fracture flow following infiltration events and evaporation along fractures during dry periods.

Upward and Horizontal Migration

Analysis of upward vertical moisture or contaminant flux from the disposal units to the surface is on-going. Preliminary results suggest near surface moisture profiles relax rapidly enough that long-term upward contaminant transport is negligible, leading to less surface contamination than the small amount attributed to translocation by biotic species as analyzed previously (1).

A preliminary parametric evaluation of horizontal moisture flux from the disposal unit to the mesa edges shows small contaminant concentrations can reach the mesa edge on the time scale of a thousand years. Numerical simulations currently in progress in accurate 2-D mesa geometry yield similar results and show radionuclide concentrations in the adjacent canyons comparable to that eventually reaching the ground water

compliance point. Contamination at the mesa edge is assumed to be carried to the adjacent canyon floor by run-off and will be input to the off-site receptor dosimetry model and to a surface contamination source term for subsequent unsaturated transport to the deep aquifer under the saturation conditions existing beneath the canyon. The net contribution to canyon receptor dose and to deep ground water contamination via this canyon contamination route is expected to be demonstrated to be small compared to the direct path beneath the mesa site.

Canyon Recharge

Recharge under the canyons adjacent to the mesa top disposal facility is likely to be much larger than that predicted within the mesa. Borehole data at nearby canyon locations shows volumetric moisture content values greater than 10% and profiles which do not vary greatly with depth in the Otowi layer. Assuming unit gradient conditions, valid under the reasonable assumptions that $Q/z \sim 0$, and $hm/z \sim 0$ (where Q is volumetric water content and hm is matric potential expressed as a head) then the recharge rate is evaluated from the Kunsat curve (Fig. 2) and corresponds with (1 to ~30) cm/yr over the observed moisture range in the Otowi from 10% to 22%. This large recharge rate may mix with the low recharge rate under the mesa at some depth, dependent upon the complex 3-D stratigraphy and stratigraphic properties. This is the subject of on-going investigation.

Toward a Conceptual Model

In this iteration of continuing analyses, the conceptual model which emerges considers first the unperturbed system and then the result of disposal operations. Transient events and vapor-liquid phase coupling can be modeled using local sources (increased local infiltration) and sinks (net evaporation from fractures or surge beds) or using more sophisticated dual porosity, dual permeability models.

Vertical contaminant flux through the unsaturated zone may be complicated by moisture source and sink terms for elevations equal to and above the adjacent canyon floors. The physical nature of these terms is uncertain, however, a picture consistent with data in the upper most strata assumes that infiltration and drying from the fractures in the mesa contribute to near surface (0-10m) moisture profiles and deeper (10-25m) drying relative to that expected for a constant vertical moisture flux. At the elevation of the adjacent canyons there is evidence of a moisture source which implies moisture migration from canyon alluvial aquifers with possible 2-D and 3-D effects, or deep infiltration through open fractures. The evidence is ambiguous and intended to be resolved in on-going field studies.

Disposal operations have been projected to increase the net moisture flux to the aquifer by over 200% of the unperturbed flux (1,3), through effectively increasing the moisture content and thus the hydraulic conductivity of the region beneath the disposal units and thus beneath the mesa as a whole. This perturbation is very sensitive to the assumed ground cover and rooting depth on the disposal unit post-closure covers (19). The effect of disposal operations on moisture flux and therefore contaminant migration from the site is therefore likely to be significant, emphasizing that operations should be conducted to minimize the perturbations to a geologic system which is naturally well suited to minimize contaminant migration and maximize waste isolation.

Unsaturated transport from the disposal site through each of the strata to the main aquifer is being approximated using conservative estimates

for the unknown transport properties in the deeper strata. The net result in concentrations and transit time to the aquifer is being revised from that determined in the original steady state analysis (1,3) to include transport through the deeper basalt layers and 3-D geometry effects.

CONCLUSIONS

As more data and analyses accumulate, the best estimate of the mean recharge rate in the upper stratigraphic layers within the mesa decreases while the variation between locations remains large. Our present best estimate is that the mesa interior is dominated by vapor phase movement and supports negligible liquid phase movement at least down to an elevation near the adjacent canyon floors. This holds under undisturbed conditions but is uncertain under the disturbance of disposal operations. A moisture source is apparent in analyses of the moisture profile peak observed near the depth of the interface between the vitrified and the devitrified tuff, also near the elevation of the adjacent canyon floors. Surface connected fractures to this depth could explain this source although the analysis method is inconclusive, subject to data uncertainties requiring additional field work for verification. At greater depths below the mesa the recharge rate is less well determined but results are consistent with a downward flux of 1 mm/yr or less directly beneath the mesa. Beneath the adjacent canyons the recharge rate is higher and remains to be quantified in near future studies. Preliminary results showed little mixing of infiltration originating beneath the canyon and mesa in the case of horizontal stratigraphy. Recent model results show different behavior with mixing sensitive to the imposed dip of some units. The extent of mixing between canyon and mesa infiltration in more realistic 3-D geometry is currently under evaluation. The implications for PA modeling is the subject of on-going work.

ACKNOWLEDGMENT

This work is sponsored by the U.S. Department of Energy, Waste Management Program Office.

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32-3
U.S. NUCLEAR REGULATORY COMMISSION STAFF TECHNICAL POSITION ON EFFLUENT DISPOSAL AT LICENSED URANIUM RECOVERY FACILITIES
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ABSTRACT

NRC-licensed uranium recovery operations, including conventional mills and in situ leach (ISL) facilities, generate liquid waste (i.e., effluent) that requires proper disposal. This paper discusses a Staff Technical Position (STP) that was issued by NRC's Division of Waste Management in April 1995, which provides the technical and regulatory basis for review and evaluation of proposals for effluent disposal at NRC-licensed uranium recovery sites. The STP is primarily intended to guide NRC staff reviews of site-specific proposals for disposal of liquid waste at conventional mills and ISL facilities; but the STP can also be used by licensees for the preparation of such proposals.

According to the STP, disposal of liquid waste at licensed uranium recovery facilities should comply with the design standards and other provisions for ground water protection in Appendix A to 10 CFR Part 40; the dose limits and other regulations in Subparts K and D, 10 CFR Part 20, as applicable depending on the proposed disposal procedure; and NRC's decommissioning requirements. In addition, the STP points out that in accordance with the provisions of Part 20, compliance with the dose limits in Part 20 does not relieve licensees from complying with the other applicable Federal, State, and local environmental and health protection regulations governing any other toxic or hazardous properties of materials that may be disposed under Part 20, Subpart K.

Specific regulatory requirements and guidance on compliance with the NRC regulations at particular sites are outlined for four known effluent disposal methods practiced at uranium recovery facilities. These are: on-site evaporation, on-site land application including on-site irrigation, release in surface waters, and injection in deep wells.

BACKGROUND

NRC-licensed uranium recovery facilities, including conventional mills and in situ leach (ISL) facilities, generate liquid waste (i.e., effluent) that requires proper disposal. At conventional mills, effluent may include liquid wastes discharged from mills, tailings liquor, and contaminated water recovered from ground-water corrective action programs and tailings dewatering activities.

At ISL facilities, effluent is generated from four liquid waste streams: Two involving the host aquifer and the other two originating from above ground operations. Liquid waste streams involving the host aquifer include production bleed and ground-water sweep. Production bleed is ground water extracted from the aquifer during the uranium recovery operation, in excess of injected water, in order to maintain a net ground-water inflow into the recovery zone and minimize or eliminate the migration of lixiviant and dissolved uranium outside the recovery zone. Ground-water sweep is ground water extracted primarily to restore ground-water quality in the recovery zone, after the ore extraction is stopped. Liquid waste streams originating from above ground operations include wastewater from yellowcake processing and reject brine from reverse osmosis treatment of contaminated water.

Evaporation has generally been used for management of liquid waste at licensed conventional mills and mill tailings disposal sites. This

practice involves discharging liquid waste in one or more on-site lined evaporation ponds where the water is lost to the atmosphere by surface evaporation and other evaporation enhancement systems, and the remaining sludge is placed in a licensed tailings disposal facility. At ISL facilities, management of liquid waste has generally involved such disposal practices as release in surface waters, on-site land applications including on-site irrigation, and injection in deep wells.

PURPOSE AND APPLICABILITY

The NRC Staff Technical Position (STP) on effluent disposal provides guidance and discusses the technical and regulatory basis for review and evaluation of proposals for disposal of liquid waste at licensed uranium recovery facilities.

The STP is primarily intended to guide NRC staff reviews of proposals for effluent disposal at conventional mill and ISL sites. The STP can also be used by NRC licensees and applicants for preparation of proposals for effluent disposal at particular sites.

The STP is applicable to both licensed and new facilities.

APPLICABLE REGULATIONS AND STANDARDS

In general, applications and proposals for effluent disposal at licensed uranium recovery facilities must comply with the regulations in Appendix A to 10 CFR Part 40; Subparts K and D, 10 CFR Part 20, as applicable depending on the proposed disposal procedure; and the applicable decommissioning requirements.

Applicable Regulations in Appendix A to 10 CFR Part 40

The applicable regulations in Appendix A to 10 CFR Part 40 (Appendix A) a, mainly include design standards for construction, maintenance, and operation of surface impoundments that are used for disposal of liquid waste or waste containing free liquids (Criteria 5A(1) through 5A(5)); installation of liners (Criterion 5E); and seepage control (Criterion 5F). Appendix A also includes other generally applicable provisions for ground-water protection, including site-specific standards for radioactive and non-radioactive hazardous constituents (Criteria 5B, 5C, and 13); corrective action programs (Criterion 5D); monitoring requirements (Criterion 7); and closure requirements (Criterion 6). Furthermore, Criterion 8 of Appendix A requires that byproduct materials be managed so as to conform to the applicable provisions under the U.S. Environmental Protection Agency's (EPA) National Pollutant Discharge Elimination System (NPDES) regulations, in 40 CFR Part 440, "Ore Mining and Dressing Point Source Category: Effluent Limitations Guidelines and New Source Performance Standards, Subpart C, Uranium, Radium, and Vanadium Ores Subcategory," as codified on January 1, 1983. A detailed discussion of the EPA NPDES regulations is outside the scope of this paper; but a summary of those NPDES regulations that are applicable to effluent disposal from licensed NRC facilities, including those pertaining to conventional mills and ISL facilities, is provided in an appendix to this paper.

Applicable Regulations in 10 CFR Part 20

Byproduct material disposal under Part 20b requires compliance with the applicable regulations in 10 CFR Part 20, Subpart K (Waste Disposal) and Subpart D (Radiation Dose Limits for Individual Members of the Public). Subpart K offers provisions for byproduct material disposal by "release in effluents"(20.2001), or other disposal methods proposed by the licensee (20.2002). Among other requirements, the provisions in 20.2001 and 20.2002 require compliance with the radiation dose limits for

individual members of the public in 20.1301, and a demonstration of compliance with these limits as provided in 20.1302.

The dose limits in 20.1301 include the total effective dose equivalent to individual members of the public (1 mSv/year), as well as the dose in any unrestricted area from external sources in any one hour (0.02 mSv in any one hour) (20.1301 (a) and (b)). In addition, the regulations allow a licensee to apply for Commission authorization in advance to operate up to an annual dose limit for an individual member of the public (5 mSv), which the Commission may generally authorize on a temporary basis or under special circumstances involving existing facilities (those designed prior to January, 1994), subject to the requirements in 20.1301 (1), (2), and (3). The regulations also require (in 20.1301 (d)) that licensees who are subject to the provisions of EPA's generally applicable environmental standards in 40 CFR Part 190 shall comply with these standards. In some cases, the Commission may impose additional restrictions on radiation levels and on the total quantity of radionuclides that may be released in effluents in order to restrict the collective dose at a particular site (20.1301 (e)).

In order to demonstrate compliance with the dose limits for individual members of the public in 20.1301, licensees and applicants must do so according to the provisions of 20.1302, which require that licensees:

a) demonstrate compliance with the dose limits for individual members of the public by conducting surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas; and,

b) show compliance with the annual dose limit by demonstrating, by measurement or calculation, that the total effective dose equivalent to the individual likely to receive the highest dose from the licensed operation does not exceed the annual dose limit; OR, by demonstrating that the annual average concentrations of released radioactive materials do not exceed the effluent concentration values (for water) provided in Table II of Appendix B to 20.1001-20.2401 and that the dose from external sources to a continuously exposed individual would not exceed the established standard (0.02 mSv/hour and 0.5 mSv in a year).

The provisions of 20.1302 also allow licensees, upon approval by the Commission, to adjust the effluent concentration values in Table II of Appendix B to 20.1001-20.2401 for members of the public to take account of the actual characteristics of effluent that will be released (20.1302 (c)).

The provisions of 20.2007 explicitly state that compliance with the dose limits in NRC regulations does not relieve licensees from complying with the other applicable Federal, State, and local environmental and health protection regulations governing any other toxic or hazardous properties of materials that may be disposed of under Subpart K.

In addition to the above requirements, licensees and applicants considering disposal of licensed materials under the provisions of either 20.2001 or 20.2002 are further required to comply with NRC's regulatory provisions for decommissioning of licensed facilities, prior to facility closure and license termination. These provisions include the interim cleanup criteria presently in use, and those specified in the final rule when the final rule is promulgated (the proposed radiological criteria for decommissioning are provided in the proposed rule in 10 CFR Part 20, Subpart E: 20.1401 through 20.1405, FR Vol 59, No. 161, page 43228, dated August 22, 1994).

APPLICABLE REGULATIONS BY DISPOSAL METHOD AND PROPOSAL REVIEW AND EVALUATION CRITERIA

In general, licensees of uranium recovery facilities are required to submit proposals for disposal of liquid waste, and obtain NRC's approval of the proposed procedures. Proposals will be approved on a site-specific basis by NRC staff based on demonstrated compliance with all of the applicable regulations.

The applicable regulations and proposal review and evaluation criteria are discussed in the following paragraphs for four disposal procedures that have been in practice at licensed uranium recovery facilities. These include: on-site evaporation; release in surface waters; on-site land applications; and injection in deep wells.

On-Site Evaporation

In accordance with the provisions of Appendix A, proposals for on-site evaporation systems must demonstrate that the proposed disposal facility is designed, operated, and closed in a manner that prevents migration of waste from the evaporation systems to a subsurface soil, ground water, or surface water. In addition, applicants must demonstrate that site-specific ground-water protection standards and monitoring requirements are adequately established to detect any migration of contaminants to the ground water and to implement corrective action to restore ground-water quality if and when necessary as required by the regulations.

Specifically, evaporation pond systems will be approved if they comply with the regulatory requirements in Appendix A. These mainly include the design provisions for surface impoundments (Criteria 5A(1) through 5A(5)); installation of liners (Criterion 5E); and seepage control (Criterion 5F). In addition, evaporation ponds must also meet other generally applicable provisions for ground-water protection, including site-specific standards for radioactive and non-radioactive hazardous constituents (Criteria 5B, 5C, and 13); corrective action programs (Criterion 5D); monitoring requirements (Criterion 7); and closure requirements (Criterion 6). The closure requirements in Criterion 6 would not apply to evaporation pond sites if the evaporation impoundments are dismantled after evaporation is stopped, and the remaining sludge and other contaminated materials are transferred to a licensed tailings impoundment for long-term disposal. However, compliance with the applicable decommissioning regulations would still be required.

Release in Surface Waters

Proposals for effluent release in surface waters must demonstrate compliance with the provisions of 20.2001 and 20.2007, as well as the effluent limitations in 40 CFR Part 440, as applicable based on site-specific conditions, pursuant to Criterion 8 of Appendix A to 10 CFR Part 40.

Specifically, release in surface waters must meet the regulatory provisions in 20.2001 (a)(3), which require compliance with the dose limits for individual members of the public in 20.1301. Licensees and applicants will need to demonstrate compliance with these limits in accordance with the provisions of 20.1302. The provisions in both 20.1301 and 20.1302 have already been discussed (see Applicable Regulations above). It should be noted that pursuant to the provisions of 20.2007, compliance with the dose limits under 20.1301 and 20.1302 will not relieve the licensees and applicants from complying with other applicable Federal, State, and local environmental and health protection regulations

governing any other toxic or hazardous properties of materials disposed of under Subpart K.

Compliance with Criterion 8 of Appendix A to 10 CFR Part 40 requires conformance to the EPA's NPDES provisions in 40 CFR Part 440, as applicable (see appendix). Licensees must obtain NPDES permits issued by the EPA or a permitting state, and demonstrate compliance with the effluent limitations in 40 CFR Part 440, Subpart C, as they apply to their particular sites.

The effluent limitations in 40 CFR 440 that are applicable to existing NRC-licensed facilities are provided in Tables I and II (in the appendix). The concentrations of pollutants discharged from existing mills, including conventional mine-mill facilities and process wastewater from existing ISL facilities, shall not exceed the BPT-based effluent limitations for "mill discharge" in Table I. At ISL facilities, process wastewater that must comply with the effluent limitations in Table I includes liquid waste generated from yellowcake processing, reject brine from reverse osmosis treatment of wastewater, and production bleed from the host aquifer.

Table I

The concentrations of pollutants discharged in mine drainage, from conventional mines and mines using ISL methods, shall not exceed the BAT-based effluent limitations for "mine drainage" in Table II. At ISL facilities, mine drainage that must comply with the effluent limitations in Table II includes groundwater sweep, or ground water extracted to restore water quality in the host aquifer after uranium and pregnant liquor extraction is stopped.

Table II

It is noted that the effluent limitations in Tables I and II are identical for Zn, Ra226 (dissolved and total concentration). However, the effluent limitations in Tables I and II are different in the following respects: 1) Table I provides effluent limitations for TSS, As, NH3, and pH for which there is no corresponding values in Table II; 2) the effluent limitation for COD in Table I is different from that in Table II; and 3) only Table II provides an effluent limitation for uranium. Therefore, ISL licensees proposing to dispose byproduct material by release in surface waters under 40 CFR Part 440 may need to satisfy different effluent limitations, depending on whether the effluent discharged is characterized as a "mill discharge" or a "mine drainage". Consequently, licensed ISL facilities that involve commingling of process wastewater and groundwater sweep wastewater in an interim common storage facility (i.e., storage reservoir) before the wastewater is released in surface waters have two alternative options to satisfy the regulations. Under the first option, a licensee would monitor the incoming wastewater by source and meet the corresponding effluent limitations separately for "process wastewater" and "mine drainage" at their respective points of discharge into the interim storage facility. If both input streams are within the appropriate effluent release limits, the licensee would be free to release the wastewater from the storage facility. In the second option, a licensee would not monitor the input streams, and would release the liquid waste under 10 CFR Part 20, Subpart K and meet the dose limits and other applicable requirements in 20.2001, before releasing the commingled wastewater in surface waters.

Licensees and applicants disposing effluent by release in surface waters are further required to comply with NRC's regulatory provisions for

decommissioning, prior to facility closure and license termination (decommissioning requirements have already been discussed under Applicable Regulations and Standards).

Land Applications

Proposals for disposal of liquid waste by on-site land applications, including irrigation, will be approved under the provisions of 20.2002. Licensees must in this case provide a description of the waste, including its physical and chemical properties that are important to risk evaluation; the proposed manner and conditions of waste disposal; an analysis and evaluation of pertinent information on the nature of the environment; information on the nature and location of other potentially affected facilities; and analyses and procedures to ensure that doses are maintained As Low As Reasonably Achievable (ALARA) and within the dose limits in Part 20 (i.e., 20.1301).

Proposals must analyze and assess projected concentrations of radioactive contaminants in the soil; projected impacts on ground-water and surface water quality, and on land uses including particularly crops and vegetation; and projected exposures and health risks that may be associated with radioactive constituents reaching the food chain to verify that the projected doses and risks conforming to the risk levels permitted under Part 20. It is expected that proposals include provisions for periodic soil surveys that include contaminant monitoring to verify that the contaminant levels in the soil do not exceed those projected, and a remediation plan that can be implemented in the event that the projected levels are exceeded.

In addition to the radiation dose, it may also be necessary in some cases to conduct analyses to assess the chemical toxicity of radioactive and non-radioactive constituents in order to evaluate the health risks associated with land applications involving irrigation at particular sites, in compliance with other applicable Federal, State, and local environmental and health protection regulations that must also be satisfied pursuant to 20.2007. Staff will work with appropriate Federal and State agencies if necessary to review site-specific chemical toxicity evaluations, and to verify that any necessary permits for this purpose are secured as warranted by the applicable regulations.

In the absence of compliance monitoring wells in the uppermost aquifer in the area used for effluent disposal or for installation of land application systems including temporary surface storage facilities, proposals must demonstrate that contaminants will not be returned to the ground water and cause exceedence of any site-specific ground-water protection standards that are established pursuant to Appendix A of 10 CFR Part 40.

Licensees and applicants disposing effluent by on-site land applications are further required to comply with NRC's regulatory provisions for decommissioning, prior to facility closure and license termination (decommissioning requirements have already been discussed under Applicable Regulations and Standards).

Deep-Well Injection

Proposals for disposal of liquid waste by injection in deep wells must meet the regulatory provisions in 20.2002. Specifically, proposals must in this case include a description of the waste, including its physical and chemical properties that are important to risk evaluation; the proposed manner and conditions of waste disposal; an analysis and evaluation of pertinent information on the nature of the environment;

information on the nature and location of other potentially affected facilities; and analyses and procedures to ensure that doses are ALARA, and within the dose limits in Part 20 (i.e., 20.1301).

Proposals must also demonstrate that the injection zone is confined, that it is not a drinking water source, and that the injected contaminants will not cause exceedence of any established site-specific ground-water protection standards in the uppermost aquifer or result in any cross contamination that would adversely impact another zone that is a source of drinking water. If necessary and warranted by site conditions, proposals may include provisions for periodic ground-water monitoring in the vicinity of the injection well to verify that drinking water zones are free from cross contamination, and a remediation plan that can be implemented in the event that unacceptable levels of contamination are detected.

In addition, pursuant to the provisions of 20.2007, proposals for disposal by injection in deep wells must also meet any other applicable Federal, State, and local government regulations pertaining to deep well injection, and obtain any necessary permits for this purpose. In particular, proposals must satisfy the EPA's regulatory provisions relevant to the injection of radioactive waste under the Underground Injection Control (UIC) Program (i.e., 40 CFR Parts 144, 145 and 146), and obtain necessary permits from the EPA and/or States authorized by EPA to enforce these provisions. In general, proposals that satisfy the EPA regulations under the UIC program will be approved by NRC staff.

Licensees and applicants disposing effluent by injection in deep wells are further required to comply with NRC's regulatory provisions for decommissioning, prior to facility closure and license termination (decommissioning requirements have already been discussed under Applicable Regulations and Standards).

ISSUES

The NRC staff has received many comments on the STP from the uranium industry. The primary issues raised by the industry to date are:

1) The soil cleanup requirement in the proposed rule for decommissioning of licensed facilities is too stringent and cannot possibly be met by the licensees disposing effluent by land application/irrigation.

2) According to the industry, the sludge in the radium settling ponds may no longer be considered as an 11e.(2) byproduct material if the groundwater sweep is considered as a mine drainage and not an 11e.(2) byproduct material; therefore, licensees may not be able to continue the practice of sludge disposal in tailings impoundments.

These issues are not considered germane to the STP because the STP only interprets and provides guidance on implementation of the applicable regulations, and is not a rulemaking instrument. Nevertheless, these are important issues and the staff will make necessary revisions to the STP as warranted after the issues have been satisfactorily resolved.

REFERENCES

1. Division of Waste Management, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, "Staff Technical Position on Effluent Disposal at Licensed Uranium Recovery Facilities, Revision 0," (April 1995).

APPENDIX

A Summary of Effluent Limitations and Standards Applicable to NRC Licensed Facilities in 40 CFR Part 440: "Ore Mining and Dressing Point Source Category,Subpart C, Uranium, Radium and Vanadium Ores Subcategory"

The U.S. Environmental Protection Agency's (EPA) regulations in 40 CFR 440 that are applicable to NRC licensed uranium recovery facilities are provided in 40 CFR 440.30, 440.32, and 440.33. Since the NRC does not regulate conventional mining, the effluent limitations in 40 CFR 440 pertaining exclusively to conventional mines, including the new source performance standards, in 440.34, are not applicable to NRC licensed facilities and will not be discussed in this summary.

Effluent limitations in 40 CFR Part 440 that are applicable to NRC-licensed facilities are reproduced in Tables I and II. The effluent limitations in Table I, representing the degree of effluent reduction attainable by the application of the best practicable control technology available (BPT), are applicable to the concentrations of pollutants discharged from existing mills, including conventional mill-mine facilities and mines using in situ leach (ISL) methods. The effluent limitations in Table II, representing the degree of effluent reduction attainable by the application of the best available technology technologically achievable (BAT), are applicable to pollutants in mine drainage from existing mines, including conventional mines and mines using ISL methods.

Treatment technology for specific sites will be approved by the EPA based on the regulatory provisions in 40 CFR Part 125: Criteria and Standards for the National Pollutant Discharge Elimination System; Subpart A: Criteria and Standards for Imposing Technology Based Treatment Requirements Under Sections 301 (b) and 402 of the Act (i.e. Clean Water Act) (40 CFR Part 125, 125.1 through 125.3).

The definition of ISL methods in 40 CFR Part 440 indicates that in situ mine and mill process wastewater does not include discharges from wells from within or surrounding the in situ mines used to restore aquifers after all actual mining activity (i.e., extraction of ore or pregnant liquor from the in situ process) has been completed (40 CFR Part 440, Subpart L, and 47 FR 54604). NRC's Office of the General Counsel has interpreted the EPA regulations such that the effluent limitations in Table I would apply to wastewater discharge from conventional mills and process wastewater discharge from ISL facilities; and that the limitations in Table II would apply to mine drainage from conventional mines (unregulated by NRC) and to groundwater sweep discharged from ISL facilities as a result of aquifer cleanup activities after ore and pregnant liquor extraction is stopped.

Therefore, effluent discharge from existing mills, including process wastewater discharge from existing ISL facilities, shall not exceed the BPT-based effluent limitations provided for "mill discharge" in Table I. Groundwater sweep from existing ISL facilities, shall not exceed the BAT-based effluent limitations for "mine drainage" provided in Table II.

32-4

HLW DISPOSAL IN DEEP SULFUR-FILLED BOREHOLES: START OF R&D*

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ABSTRACT

According to a new concept for high level nuclear waste (HLW) disposal which was recently suggested 1) and discussed 2), great number of small (of about 0.2---0.3 m in diameter) capsules with HLW are loaded in

sequences into the mouth of deep (about 4 km), narrow (less than 1 m in diameter) sulfur-filled borehole. The capsules sink downward melting sulfur by their rad-heat and after about a year will reach the borehole bottom where they accumulate - initially as a column of adjusted capsules. Being relatively cold (120C) during the loading process the capsules near the borehole bottom will undergo slow heating up to temperatures about 400C, causing thermal and chemical destruction of the borehole walls along the column of accumulation. At these temperatures sulfur reacts mainly with oxidized iron Fe(II) of rocks forming stable pyrite and troilite. Then capsules should form an ensemble (hot magma droplet) of diameter about a few meters and of temperature exceeding the rocks melting temperature (1600C). Due to its higher than rock average density such ensemble will sink down up to the moment (30-100 years) when its decreasing radioactivity becomes insufficient to melt rocks thus forming a deposition at depth 5 to 8 km.

The experimental and computer research of related processes was going in the directions:

1) Study of sulfur-granite interactions in the temperature range 120 to 550C (Natalia A. Kosyakova, Institute of Experimental Mineralogy, Russian Academy of Sciences [RAS], Chernogolovka). First results are: liquid sulfur penetrates through granite micro-crackings and slowly interacts with surface of grains. At moderate temperatures water-permeability of sulfurized granite decreases. Durability of obtained materials is under further investigation.

2) Study of dynamic melting around a single model capsule (Barrikad V. Zamyshliayev, Vladimir M. Chernyshev, Viktor S. Osolovsky, Central physical-technical institute, Sergiev Posad) and around groups of capsules heated by microwaves (Andrei V. Vityazev A.V., Julius I. Zetzer, Igor B. Monastyrsky Georgy A. Ovsyannikov, Institute for Dynamics of Geospheres RAS, Moscow). The nontrivial equilibrium form of heat evolving and simultaneously sinking ensemble was found in experiments with microwave heating. Computer modelling is intended to confirm this dynamic equilibrium and to find its possible modifications in real media.

3) Study of thermally induced stresses developing during rocks heating and then cooling after solidification of rock melt. (Yuri A. Paveshenko, Institute of Applied Mathematics RAS, Moscow). In a long-term run these stresses can in principle cause multiple cracks in fresh rocks which could lead to water convection driven by the radioactive heat source.

THE "NEWDEEDS" CONCEPT AND RELATED PROBLEMS

The idea of the "NEWDEEDS" (NuclEar Waste DEEp Disposal in Sulfur filled boreholes) concept (1,2) is to transport high-active nuclear waste to great depths through a narrow channel bored in rocks and filled with sulfur. It is a development of a known (3,4) idea of HLW self-burial but starting from initial deep horizon reached by borehole thus providing higher safety (patents (5,6), the first variant reported in 1991 (7)). The theoretical results show a principal possibility for its practical implementation. The disposal depth leads to high isolation from the environment. The estimates show reasonable economic efficiency of the proposed method due to a possibility of multiple use of the borehole. The diameter of the borehole in rocks (granites) can be made as small as 0.4---0.5 m in its main cross-section and near the bottom. The next operation is a filling of the borehole with natural sulfur. This operation is a problem for technology, but no principal difficulties can be seen on the way of its realization. Then loading of nuclear waste

starts: a consequence of hermetically-sealed capsules with HLW sink down in sulfur filled borehole, melting sulfur by the heat release of waste, it is possible by a series of rather small capsules with the diameter less than 0.2 m. If the time interval between consequent capsules is less than 1 day, then each capsule is moving in a thermal channel of previous capsules, the theory of such consequent loading was given in (1). The peculiarity of this loading process is based on a fact that the sulfur heat conductivity is more than 10 times lower than that of rocks. The temperature of the hole walls remains low during loading, it never exceeds the sulfur melting temperature 113---120C (depending on pressure).

Capsules accumulate near the borehole bottom causing a local temperature increase, which in its turn causes the thermal and mineral instability of the surrounding rocks. The criterion of the borehole thermal instability obtained in (1), shows that it occurs about a year after the start of accumulation when temperature near the bottom increases up to 300-400C. Somewhere in the same temperature region the mineralogical interactions of sulfur with granite also leads to sharp decrease of the rock durability.

The loss of stability due to the thermal expansion and sulfur erosion of the walls leads to deformation of the hole. Then the whole heavy column of the accumulated waste (a several hundred meters) subsides to the bottom of the hole. There the ensemble of capsules is formed (a hot droplet or heavy magma chamber). Its temperature will be growing to those of softening of granites (about 1600C). By that moment the whole sulfur in this droplet is to be spent already for chemical reactions, the basic products of which are solid, water insoluble pyrite and pyrrhotite. Since the densities of both pyrite and encapsulated waste are higher than that of granite, a hot droplet will separate from the sulfur column and start melting rocks in a self-burial regime. The theory of sinking of a hot sphere was given in (8), however the equilibrium form of a liquid hot moving body was unclear.

After some cooling of the droplet due to decay of the most active isotopes with the half-time 30 years a solid water-non-soluble matrix will form around the disposal, it should preserve a leakage of radionuclides into underground waters. However some cracking of this solid matrix is possible during its cooling, this process was unclear in details.

RESEARCH DIRECTIONS

On account of great responsibility for radioactive and ecology safety it appears to be necessary to carry out some laboratory simulations of main stages of the suggested method of HLW disposal without actual use of radioactive isotopes. Four intercrossing directions were suggested for further theoretical and experimental research:

1) calculations of thermodynamical equilibrium in the sulfur-rock system and comparing its results with the experimental rock samples undergoing thermal contact with sulfur;
2) a study of electrically heated capsules sinking in sulfur;
3) a study of collective motion simulated by microwave heating of liquid droplet in a microwave transparent media;
4) a study of solidification in the trace of a hot droplet.

Here some results of these studies will be briefly reported.

MINERALOGICAL STUDIES

We found no scientific literature on mineralogical experiments on granite rock interactions with sulfur. Such a contact is also not observed in natural environment. Preliminary theoretical results for thermodynamical equilibrium in the granite rock-sulfur system are included in the book (1). The intermediate mineral compositions are determined by temperature and the overall sulfur concentration among a vast variety of possible minerals and chemical compounds in the system of sulfur, water and iron oxides. Other natural minerals (oxides that constitute granite rocks) apart from iron II and III oxides and water were found not to interact with sulfur, they produce only secondary influence on the final equilibrium phases. Actually in the temperature region 100-600C 12 stable regions on the phase diagram for the system FeO+S and 24 stable regions on the phase diagram for the element system Fe-S-H-O were found theoretically.

The mineral distribution in the cross-section of the initial mineral sample (granite) after a fixed exposition with hot sulfur is determined by simultaneous processes of heat conductivity and diffusion along natural cracks. It corresponds to a cross-section of the phase diagram known from thermodynamic calculations, thus it could give a principal possibility to compare experimental and theoretical results. These results also will give a possibility to predict the final mineral composition along the sulfur filled borehole and column with radioactive waste. Study of simultaneous processes of heat transfer (the rock heat conductivity is known) and diffusion (the diffusivity along cracks is unknown and nearly impossible to evaluate theoretically) should give estimates of the time scale for the mineral processes involved. These results are urgently need to be incorporated in the computer program modelling the HLW disposal behavior during loading, initial heating and future development.

The set of experiments was started in the Institute of Experimental Mineralogy RAS by Natalia A. Kosyakova. Its results will be published elsewhere and here are reported as a principal summary.

The samples for further analysis were obtained with a rock in sulfur exposition from 8 to 24 hours for fixed temperature in the range 120 to 500C. We made two types of mineralogical experiments. 1) Closed in vacuum quartz containers with a rock sample covered with melted sulfur were heated in stove and fast cooled after the exposition period. 2) To emulate a situation in real borehole we drilled a cylindrical hole in granite rock samples, filled it with sulfur and use a thin electric heater inside the hole with the open air contact and external cooling. The maximum reached temperature in the second type experiment was 160C. Then both type samples were cut and their surface analyzed optically and by electronic microanalyzers (the installations CAMEBAX and LINK) which generally give possibility to obtain a detailed mineral analysis along the samples with high space resolution.

The results of our mineralogical experiments occur to be in a sense unpredictable. First of all, we found that the liquid sulfur permeability in granite is very high. An exposition of a few hours in closed containers leads to full penetration and nearly homogeneous distribution of sulfur along samples of about a centimeter in size. In samples of the second experiment type we observed a visual sulfur gradient of a few millimeter deep from the hole, it was also statistically confirmed by the instrumental analysis. However, results given by electronic microanalyzers are rather difficult for simple mineralogical

interpretation. We found practically no exact mineral compositions with sulfur, moreover, the sulfur content was highly variable along the sample with fluctuations about the order of magnitude. Our preliminary conclusion is that the liquid sulfur penetration into solid granite is very effective along the micro cracks in between granite grains. Sulfur fills all the empty micro spacings in granite but only slow chemically interacts on surfaces of mineral grains.

First experiments to establish the durability of obtained sulfur exposed granite materials show that their strength is nearly unchanged for temperatures lower 160C and is drastically reduced at temperatures somewhere in the interval 300-400C. Moreover, we found even some increase of sample durability at low temperatures, but this conclusion is yet not statistically significant due to small number of tests.

EXPERIMENTS IN SULFUR WITH ELECTRICAL HEATING

Theoretical calculations for the processes going when hot capsules melt sulfur and move in the melt prove to be rather difficult since viscosity of sulfur strongly and unusually depends on temperature (after melting point it has a minimum, then sharply increases up to the maximum which is more than two orders higher than the minimum value, then slowly decreases). The other reason to determine the sinking rate experimentally is a complex convective structure in the upper part of hot capsule, especially in the case when small water impurities are residual in sulfur.

The sulfur melting experiments were fulfilled by Barrikad V. ZAMYSHLIAYEV, Vladimir M. CHERNYSHEV, Viktor S. OSOLOVSKY in Central physical-technical institute, Sergiev Posad (9). The general scheme of this experiment is rather simple. A metal sphere of diameter about 10 cm with an electric heater inside is placed on the surface of sulfur in a cement tube imitating the borehole. Two variants of initial sulfur constitution were tested: a)melted and then solidificated sulfur and b)dispersed sulfur with small content of water in spacings. These two variants correspond to the possibly different methods of the borehole filling with sulfur. This study was intended also to determine the proper method of the borehole filling. The tube and capsule radii and the thermal power are chosen to be close to real parameters of borehole and high level waste.

When the capsule heating is on, it begin to sink in sulfur inside the cement tube. Its position was measured by microwave detector and temperature measurements. This experiments gave velocities of capsules in dependences on the applied thermal power. The results generally confirmed the theoretical evaluation. The sinking velocity in melted and solidificated sulfur for a single sphere of 100 mm in diameter and the thermal power of 60 W proves to be about 1 mm/min that corresponds to 0.5 km per year. It means that sinking of a chain of consequent capsules of even lower power can reach an average velocity of a few kilometers per year.

Heating of the same capsule in dispersed sulfur with water remnants in spacings leads to reaching a final depth of about the sphere radius. The observed water convection in a cavity formed over the sphere was a main source of the heat removal. However, application of the additional air pressure inside of about 1.2 atm (extra) prevent the water boiling and lead to continuation of the sinking process with velocity close to the case of solid sulfur.

MICROWAVE HEATING EXPERIMENTS

A set of experiments in transparent solid-liquids with model microwave heating results the structure of convective flows (10). In the experiments with microwave heating paraffin was uses as a model solid-liquid media. The microwave installation of Institute for Dynamics of Geospheres with videotape recording gave possibility to obtain convective patterns around a hot droplet which models a magma chamber with HLW heating. Since current positions of capsule models (which absorb radiation) are changeable, a moving mechanical support with back regulating connection was designed and constructed. The process was recorded into videocassets for further investigation.

The second experiment with liquid heaters absorbing microwave radiation was intended to obtain experimentally a fraction of capsules remaining in the solidificated column. Its result was rather impressive. It was found that liquid heat evolving bodies in the process of sinking have no tendency of self separation or fractionating. Moreover, it was found that the equilibrium form of such hot moving droplets is not a tailed sphere with partial dispersion as it was supposed before experiments, but a dynamic stable saucer-like body.

The computer modelling of heated liquid body which melts down is under development (Yuri A. Paveshenko, Institute of Applied Mathematics RAS, Moscow). The other purpose of this calculations is to establish the distribution if thermal stresses around a cooling trace of a HLW self burial. In a long-term run these stresses can in principle cause multiple cracks in fresh rocks which could lead to water convection driven by the radioactive heat source. However at such depths due to heightened temperature and pressure the reverse process of cracks closing becomes also active.

The possible additional output of these experiments and computer modelling can be also valuable for study of natural geophysical processes such as magma ascending and gravitational separation of minerals.

CONCLUSION

The final goal of the current stage is to implement the obtained experimental results into a computer program with changeable interactive input and graphical output. The program should model all stages of the disposal development and predict the final underground fields of temperature and mineral depositions. In their turn, they can reveal in the first approximation the final radioactivity distribution.

This research was made possible in part by Grant M47000/47300 from the International Science Foundation and support of Ministry of Science.

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32-5
U-234/U-238 FRACTIONATION IN VADOSE ZONE PORE WATERS OF THE APACHE LEAP TUFF

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ABSTRACT
Decay of U-238 in the natural environment leads to production of U-234 (t1/2 = 2.48 x105yr). In a closed system of sufficient age the U-234/U-238 activity ratio (AR) approaches unity, but is > 1 in typical groundwater. The isotopes exhibit equivalent kinetic behavior in natural chemical reactions, so fractionation must result from the energy released by alpha-decay. The AR was analyzed in surface waters, near-surface fracture waters, perched waters, and vadose zone pore waters from the Apache Leap Tuff. This tuff has features that are similar to Yucca Mountain tuffs and amenable to investigation. Novel methods were used for sampling vadose zone pore waters from preserved drill cores. The AR for all surface, fracture, and near-surface pore water samples was in the range 1.4 to 2.1. A deeper transition in pore water AR from 2 to 5 correlated with trends in saturated matrix hydraulic conductivity and magnetic susceptibility, delimiting the penetration of oxidative weathering into the vadose zone. From C-14 apparent ages (1) it is evident that this multiple transition does not reflect the former high stand of a perched water table. A lumped parameter model was used to evaluate the relative contributions from selective leaching and direct recoil mechanisms to perched water fractionation at Apache Leap, integrating mass balance, radioactive decay, sorption, isotopic exchange, and direct recoil effects. The strategy was to identify a set of model parameters describing the total U concentration and AR for perched waters from different sampling locations without direct recoil, then recompute the ratios with direct recoil. The results showed that leaching selectivity is probably much greater than the observed mobile phase AR,

and that within the likely range of effective partitioning behavior the contribution from direct recoil to observed activity ratios is probably < 1.

INTRODUCTION

This study was motivated by the observation of elevated U-234 activity in perched waters in tuffs at Yucca Mountain, Apache Leap, and elsewhere. The general objectives were to investigate the spatial distribution of pore water fractionation, evaluate alternative conceptual models for fractionation, and identify implications for transport of natural U. Methods were developed to sample vadose zone pore waters for U isotopics, and applied at the Apache Leap Research Site (ALRS) to investigate U-234 fractionation throughout the vadose zone.

Davidson (1) described sampling pore waters from preserved tuff cores by core squeezing using the method and the equipment of Peters et al. (2). In the Apache Leap Tuff welding increases with depth, and this method was effectively limited to the uppermost 30 m of the section because of low water yield. Using the chemical separation and isotopic analysis methods described by Goldstein et al. (3) the U concentration and U-234/U-238 activity ratio (AR) of squeezed pore waters were compared with other waters from Apache Leap including surface waters, fracture waters from shallow borings, and perched waters (Table I, Fig. 1). A mixing plot shows that U leaching near the surface is different from the process that produces fractionation in deeper waters (Fig. 2).

Table I

Fig. 1

Fig. 2

PORE WATER SAMPLING BY FLUSHING OF PRESERVED CORES

A method was developed to inject Milli-Q water through tuff cores at a pressure of 15bars. Cores were preserved in Protec-Core packaging from the time of drilling. Leachates from 3 to 5 cores were composited yielding a total U mass of as little as ~1 ng. Accordingly, control and measurement of contamination were critical. A tracer was used to quantify leakage of confining fluid into the sample, and contamination from this source was concluded to be negligible. A blank of ~60 pg U was estimated for a typical isotopic measurement, which is less than the ~180 pg U blank estimated for core squeezing.

Using the flushing method for more densely welded samples, the AR was analyzed for pore waters from as deep as 100 m. A total of 35 cores were leached and 7 composites analyzed isotopically. Activity ratios from flushing are indistinguishable from ratios for squeezed pore waters (Table I) indicating that both methods sampled the same phase. The observed AR is a lower bound on actual pore water composition since the sample may acquire less fractionated U from the solid phase. As a byproduct of this method the matrix saturated hydraulic conductivity (Ksat) was measured.

The flushed pore water isotopics show that a transition occurs between 20 and 30 m depth in the DSB (Table I, Fig. 3) where the AR increases from 2 to ~5. Using equivalent U concentrations derived for the flushed pore water samples, a mixing plot (Fig. 2) shows that these new data are intermediate between near-surface waters and perched water.

Fig. 3

FRACTIONATION MECHANISMS

Selective leaching of U-234 from recoil-damaged sites apparently occurs in surface soils and near-surface fracture zones, given the short

residence time and the relation between total U and the AR for these waters (Fig. 2). A wide range of concentration was observed in squeezed near-surface pore waters, all with AR~2 (Table I).

The three perched water samples analyzed had similar C-14 ages (~3,000 yr) but samples from the tunnel discharge and Oak Flat borehole had 3 times more U (Table I). This is probably because the perched water table at Oak Flat lies within the near-surface porous tuff layer which is thicker there. At the ALRS the perched water table lies deeper within the formation where hydraulic conductivity is dominated by fractures. In the tunnel discharge (which probably recharges in Oak Flat) the AR is also moderately high (Table I). Where similar AR's occur over a wide concentration range the fractionation is generally attributable to selective leaching.

As pointed out by Petit et al. (4) congruently dissolving mineral phases that contain U-238 tend to dilute the excess U-234 derived from selective leaching, limiting fractionation. They argued that this always occurs given the U abundance and etching rates of rock-forming minerals. Attributing the observed perched water fractionation at the ALRS to selective leaching could imply unrealistic rates of congruent dissolution for U bearing minerals (e.g. <10-13 m/yr). Petit et al. proposed an auto-oxidative leaching mechanism instead, in which primary U-238 existing as U(4+) decays down to U-234(6+) and fractionation occurs because U(6+) is more soluble.

If sampled near-surface fracture waters represent recharge, then a selective leaching model for perched water at the ALRS requires a highly fractionated leaching "end member." The needed selectivity is directly related to the intensity of isotopic exchange with an adsorbed phase. Secondary minerals are present which tend to be good adsorbents but such partitioning could produce conditions favoring direct recoil, in which U-238 at or near the liquid:solid interface decays and the a-recoil leads to U-234 accumulation in the mobile phase. Recoil from U-238 adsorbed to secondary oxides has been suggested as the fractionation mechanism for perched waters from tuffs on the Nevada Test Site (Zielinski and Rosholt, 1978).

In the vadose zone at the ALRS, pore water residence time controlled by matrix Ksat might be a determinant of the direct recoil effect. Contrary evidence may be found in the C-14 apparent ages reported by Davidson (1) which ranged from 1,000 yr in the upper part of the tuff section to 3,000 yr in the perched zone, suggesting that residence time does not vary enough to account for the observed AR transition (Fig. 3). Alternatively, increased fractionation with depth could be caused by increased leaching selectivity at the depth to which weathering has penetrated and the rock is substantially oxidized, in accordance with the auto-oxidative leaching mechanism. Mechanisms for fractionation in vadose zone pore waters at the ALRS are the subject of ongoing investigation.

The AR transition at the ALRS correlates (within a depth range of < 10 m) with steeply decreasing Ksat and with a transition in magnetic susceptibility caused by oxidative weathering (6). At the nearby Oak Flat borehole, similar transitions in hydraulic properties and magnetic susceptibility occur at the modern perched water table. Vadose zone pore water AR's are less than the perched water AR, suggesting decay from a former high stand of perched water table roughly 105 yr ago. However, this possibility is contradicted by the C-14 apparent ages reported by Davidson (1), so it is evident that fractionation in deep vadose zone

pore waters developed independently of the perched water table elevation at Apache Leap.

The remainder of this paper evaluates the relative contributions from direct recoil and selective leaching in the perched zone at Apache Leap. This requires a model that integrates mass balance, radioactive decay, sorption, isotopic exchange, and direct recoil. The strategy was to describe the fractionation and total U in DSB and tunnel perched waters using selective leaching only, and then to use these parameters in the same model with direct recoil added.

STEADY STATE MODEL FOR FRACTIONATION IN THE PERCHED ZONE AT THE ALRS

Matrix hydraulic conductivity at the ALRS declines sharply with depth so that below a depth of 20 to 30 m recharge must move through fractures. Davidson (1) showed that perched water at the ALRS has low chloride similar to surface water, whereas the pore waters sampled had significantly more chloride which was likely leached. Thus surface waters move downward through fractures without acquiring much solute from vadose pore waters. Recharging waters from the surface are relatively unfractionated (AR~2) so it may be inferred that fractionation in the perched water (AR~6.2) occurs mostly in the perched zone. This is the rationale for a lumped parameter approach to modeling perched water fractionation.

The model is based on a system of equations (Fig. 4) solved using a Runge-Kutta method with adaptive step size (7). Initial conditions correspond to chemical and isotopic equilibrium of both the mobile and adsorbent phases with the combination of inflow and leaching. Radioactive decay begins at t=0, and time stepping to 1 My brings the system reasonably close to steady state U-234 concentrations in the mobile and adsorbent phases depending on the rate of isotopic exchange. Perched water residence time of 3,000 yr is based on the C-14 apparent age. Recharge with 100 ppt U and AR=2 similar to near-surface fracture waters is the inflow to the model.

Fig. 4

Preliminary data from selective extraction of Mn from pulverized tuff show that as much as half the whole rock Mn exists as MnO_2 (depending on the extent of oxidative weathering) containing about 50 ppm U (Fig. 5). Assuming typical values for MnO_2 abundance and pore water U concentration the whole-rock U partition coefficient is about K=100mL/g. The effective value for isotopic exchange is probably smaller because of desorption hysteresis and because part of the MnO_2 phase is inaccessible. The effective K for direct recoil is probably smaller still since this involves only U adsorbed within a few nanometers of fluid-filled pores in communication with the mobile phase.

Fig. 5

Leaching rates can be estimated under the assumption of a chemical steady state, from total U concentrations (Table I) and residence time from C-14 apparent age (3,000 yr). For the deep slant borehole (DSB) at the ALRS the corresponding leaching rate is 0.033 ppt/yr, and for the tunnel discharge it is 0.167 ppt/yr. This does not constrain the leaching selectivity, defined as the AR of leached U.

A key parameter for which few constraints are available is k, the rate constant controlling isotopic exchange. Whole rock U and Th analyses by fusion and ICP/MS show that the Th/U ratio changes little through the tuff section, therefore the rate constant must be large enough to permit sorption of leached U. If $k<10^{-14}$ sec-1 then more than a few percent of

the whole rock U would be lost over 1 My, given realistic leaching rates. Constraint on k was also derived by running the model with the initial condition of zero adsorbed U, showing that a k value of 3x10-14 sec-1 is sufficient for sorption equilibrium within a fraction of the age of the tuff (Fig. 6). If k exceeds this then significant isotopic equilibration occurs in <1My which is unrealistic given the observed fractionation.
Fig. 6
Calculations plotted in Fig. 7 show the relation between mobile phase AR, leaching selectivity, and partitioning. Strong partitioning causes dilution of the mobile phase AR even for very selective leaching, because of isotopic exchange and radioactive decay. The same values for the rate constant and leaching selectivity were used for both perched waters, but the leaching rates differed. The curves in Fig. 7 are based on concentrations of 200 and 600 ppt for the DSB and tunnel waters, respectively, with recharge having 100 ppt and AR=2. By assuming a value for leaching selectivity it is possible to pick K values for each perched water type (LAR~12 is about the smallest value that works for both). The difference between these K values reflects model mass balance requirements. The tunnel discharge K value is larger than the DSB value, which may mean that the effective adsorbent abundance is greater where there is more flow through the tuff matrix. Another application for the model could be to estimate K values from independent data, assume a selectivity, and compute values of the rate constant k for different waters.
Fig. 7
Adding direct recoil (Fig. 8) confirms that the incremental fractionation is important for larger values of K. Within the likely range of effective partitioning behavior the contribution from direct recoil to observed activity ratios is probably < 1, under the model assumptions.
Fig. 8
CONCLUSIONS
Fractionation in perched waters from the Apache Leap Tuff is attributable to selective leaching, with a minor contribution from direct recoil. Fractionation in surface and near-surface waters from the ALRS (AR~2) is attributable to less selective leaching from weathered rock. Activity ratios up to 5 were found in deep vadose pore waters and perched waters, however, the AR transition correlated with a trend in magnetic susceptibility representing the penetration of oxidative weathering. From apparent C-14 ages (1) it is evident that this multiple transition does not reflect a former high stand of the perched water table.
Using a mechanistic model it was shown that leaching selectivity is probably much greater than the observed mobile phase AR. Assuming a minimum value for selectivity consistent with the observed AR, the model permits estimation of the in situ partition coefficient. Model assumptions include a hydraulic steady state, first-order isotopic exchange kinetics, and chemical equilibrium with respect to U-238. Fractionation encountered in perched water at Yucca Mountain (AR~7.5, J. Stuckless, personal communication) is probably not limited to perched zones but exists in vadose zone pore waters as well. Fractionation can be interpreted in terms of conditions that promote or inhibit retardation of natural U. These results in conjunction with the C-14 data of Davidson (1) show that rapid recharge pathways through the vadose zone can be identified from the isotopic character of pore waters.
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RADIOACTIVE WASTE DISPOSAL IN A NUCLEAR SUBSIDENCE CRATER

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ABSTRACT

The United States Department of Energy has used subsidence craters from underground nuclear detonations in Yucca Flat on the Nevada Test Site for disposal of low-level radioactive waste since the late 1960s. Two adjoining craters in the Area 3 Radioactive Waste Management Site were filled primarily with bulk radioactive waste and temporarily capped with a minimum of 30 cm of native alluvium in 1987. Effects of the subsurface detonation on the hydrologic properties of the alluvium near a subsidence crater, the effect on the soil moisture regime, and the migration of chemical constituents and radionuclides from the waste were examined in this study. Two angle boreholes were cored and drilled to provide samples for analysis and comparison. One passed under a subsidence crater used for waste disposal while the other passed through undisturbed alluvium. In situ water content and water potential measurements on core samples

from the two boreholes were similar. Water potential gradients were steepest near the surface and decreased with depth and indicated evaporative conditions in the zone penetrated by the boreholes. Stable isotope ratios for soil pore water were consistent with evaporative conditions. Chemical analyses of samples for waste constituents showed no evidence of migration of metals into the alluvium below the waste. Similar results were found for the gamma emitters 241Am, 60Co and 137Cs. Tritium concentrations, however, were found to be somewhat elevated. Estimates of tritium diffusion from the detonation cavity indicated that the waste was the probable source of the tritium observed in the borehole samples.

INTRODUCTION

The thick vadose zones found in alluvial valleys of deserts in the western United States have characteristics favorable for shallow burial of solidified radioactive waste. The great depth to the water table and the arid climate in these regions combine to inhibit transport of radionuclides to groundwater below the waste. Shallow disposal in alluvium usually requires excavation of pits or trenches in the alluvium to receive the waste. However, in Yucca Flat on the Nevada Test Site in southern Nevada hundreds of subsidence craters remain as a result of the nuclear weapons testing program. An early proposal for disposal of radioactive waste in deep, dry alluvial deposits of Yucca Flat considered use of a crater formed by the near-surface detonation of a nuclear device as a waste cell (1).

Four subsidence craters in the Area 3 Radioactive Waste Management Site (RWMS) in Yucca Flat either have been or are now being used for disposal of low-level radioactive waste. Tests that led to the formation of the craters in the Area 3 RWMS were conducted hundreds of meters below the ground surface in shafts bored vertically into the alluvium. Upon detonation of the nuclear device, pressures and temperatures increased forming a cavity in the alluvium. When the gas pressure dissipated to a point at which it could no longer support the overburden, the roof of the cavity collapsed and the void region propagated upward to the surface forming a saucer-like subsidence crater on the surface (2). The region through which the void propagates is referred to as the chimney. An important consideration in the use of craters for waste disposal is whether the ability of the climate and geologic formation to isolate the waste from the water table is compromised by the presence of the chimney. Although hundreds of nuclear tests have been conducted in the vadose zone, little is known about the properties of chimneys (3). The little that is known relates to the size distribution of rubble formed in competent rock such as welded tuffs and granodiorite and cannot be extrapolated to alluvium or friable tuffs (3). Some measurements of saturated hydraulic conductivity have been made on core taken from near the surface of subsidence craters in alluvium (4). Values of hydraulic conductivity were similar to those found in an undisturbed area (5). Although hydraulic properties of chimneys in alluvium may not be different from undisturbed alluvium, studies of a crater in Yucca Flat (4) have demonstrated that craters capture precipitation and may capture runoff. Ponded conditions develop which enhance infiltration through the surface of the crater. The magnitude of the enchanced infiltration depends not only on the precipitation but also on the size of the catchment area outside the crater boundary. This increased infiltration could potentially result in recharge to the water table. Tyler et al.

(1992) (4) estimated that the infiltration rate through the bottom of a crater in Yucca Flat was approximately 0.6 m yr -1.

In the present study hydrologic properties of the alluvium in the vicinity of a subsidence crater were compared with those from an undisturbed area to determine if these properties were affected by formation of the crater. The effect of the presence of the crater and its use for waste disposal on the soil moisture status was examined, and sampling was conducted to determine if migration of chemical constituents and radionuclides from the waste had occurred. Two angle boreholes were cored and drilled to provide samples for analysis and comparison. The first borehole passed under a subsidence crater used for waste disposal whereas the other passed through undisturbed alluvium adjacent to the crater.

SITE DESCRIPTION

The RWMS is located at an elevation of approximately 1220 m in an alluvium-filled basin typical of the Basin and Range Physiographic Province. Alluvium beneath the RWMS is chiefly sand-sized clasts derived primarily from Tertiary volcanic rocks and Paleozoic carbonate rocks. Estimates of the thickness of the alluvium range from 370 to 460 m, with the uppermost aquifer being encountered at a depth of 490 m. Characterization of physical properties conducted in an area within a few kilometers of the study site suggest that undisturbed alluvium beneath the Area 3 RWMS has no flow-impeding layers or preferential flow paths. Air temperatures in Yucca Flat are typical of the upper Mohave Desert with large diurnal and seasonal variations. Precipitation is minimal and also highly variable with an annual mean of 15.9 cm and a standard deviation of 7.9 cm. Evaporation from an open water surface was estimated to be approximately 1800 mm/yr for a site in Yucca Flat (R. H. French, unpublished data). The small ratio of precipitation to evaporation implies that the moisture distribution in the upper part of the alluvium is strongly influenced by evapotranspiration.

A waste cell with approximately 228,000 m3 capacity was formed at the Area 3 RWMS by excavating alluvium between two adjacent subsidence craters, U3ax and U3bl, as shown in Fig. 1. Prior to excavation, U3ax was 19m deep and 138 m in diameter and U3bl was 14 m deep and 122 m in diameter. Both craters were formed from underground nuclear tests conducted in 1962. Disposal in U3ax, the deeper crater, began in the late 1960s. Disposal in U3bl began in 1984, as the level in U3ax reached that of U3bl. Waste forms consist primarily of contaminated soil and scrap metal with some construction debris, equipment, and containerized waste. Small quantities of oil, fuel, and ethylene glycol remaining in machinery and tanks were the only liquid wastes possibly placed in U3ax or U3bl. The radioactive constituents were estimated to be approximately 1200 curies with 85 percent of the activity being due to tritium. The next largest contributor was 60Co which accounted for only 1.6 percent of the total curie content. Most of the curie content was disposed during the years 1976 to 1981. The only known hazardous constituents were lead and cadmium. A temporary cover of native alluvium was placed over the waste in December of 1987. The cover is from 0.3 to as much as 4 m thick.

Fig. 1

METHODS

Two 45o angle boreholes were cored and drilled to a vertical depth of 50m to provide samples for analysis and comparison (Fig. 1). The first borehole passed under one of the subsidence craters used for waste

disposal (U3bl) shown in Fig. 2 whereas the other passed through undisturbed alluvium. The craters U3ah and U3at shown in Fig. 1 are currently being used for the disposal of low-level bulk radioactive waste.

Fig. 2

A casing advance under-reaming system was used to drill the boreholes. Air was the only drilling fluid used in this system to minimize the disturbance to the alluvium samples. The method used to obtain representative core samples was to drive a sample tube containing segmented polycarbonate liners 8.9 cm in diameter. Multiple core segments were collected from each 0.8 m core run. Measurements of the ambient water potential were made on the core samples using a water activity meter. Stable isotope concentrations were determined by mass spectroscopy following toluene distillation. The following analyses were conducted on drill cuttings samples. Lead and cadmium concentrations were determined by atomic absorption spectroscopy. Activities of 241Am, 60Co and 137Cs were determined by gamma spectroscopy. Tritium concentrations were determined using liquid scintillation.

RESULTS

The following results are presented as profiles in vertical depth from the surface. Water content profiles indicated that water contents are generally low with a mean value of 0.11 m3 m-3 and exhibit no trend with depth. The water content profile under the cell was indistinguishable from that in undisturbed alluvium. Water potentials increased with depth from -6 MPa near the surface to approximately -1 MPa at 50 m depth. Once again, the profile under the cell was indistinguishable from that in undisturbed alluvium. Analyses of drill cuttings samples from the borehole under the cell did not reveal the presence of lead or cadmium. Analyses of these samples for 241Am, 60Co and 137Cs showed that these radionuclides were not present in concentrations exceeding background. Tritium, however, was found in detectable concentrations under the cell. Profiles of tritium concentrations in soil water with depth for both boreholes are shown in Fig. 3.

Fig. 3

DISCUSSION

The similarity of the water content and water potential profiles for the two boreholes suggests that the presence of the crater has not altered the soil moisture regime at this site. A comparison of isotopic compositions of pore water from the boreholes showed that the isotopic composition of pore water from the borehole under the cell is indistinguishable from that in undisturbed alluvium. This indicates that processes affecting the correlation between dD and d18O are similar in both locations. The compositions diverged from the global meteoric water line in a manner indicative of a soil moisture regime strongly influenced by evaporation.

Movement of water in soil occurs as the result of a gradient in total potential. Here the total potential is taken to be the sum of the matric and gravitational potentials. These potential profiles for the borehole under the cell are shown in Fig. 4. The osmotic component of the water potential for the alluvium was estimated from the chloride concentrations to be less than -0.001 MPa in the near surface and zero below a depth of 10 m. Water potentials are thus taken to be equivalent to matric potentials. For the total potential, a negatively sloped line indicates that flow is upward, a positively sloped line that flow is downward and a

vertical line represents the static condition. Figure 4 shows that the zone below 35 m, where cores were taken from under the cell, is nearly static. The slope of a best fit line through the total potential data from a depth of 39 m to 49 m indicates a slightly negative gradient in this region. Thus, the flux of liquid water under the cell is negligible. This indication of a negligible flux is supported by the lack of metals or waste-related radionuclides in the samples taken from below the cell.
Fig. 4
Tritium concentrations in undisturbed alluvium (Fig. 3) are low and decrease quickly with depth to levels near or below the detection limit of 350 pCi/l. Samples with concentrations less than the detection limit are plotted at the limit. The concentrations of tritium found in the borehole under the cell generally increase with depth to nearly 10,000 pCi/l at the deepest point sampled. These concentrations are two orders of magnitude greater than expected from atmospheric deposition of tritium but 1/100 of the Nuclear Regulatory Commission's concentration limit of 1 x 106 pCi/l for effluent released to an unrestricted area.
These samples represent not only tritium concentration with increasing depth but also with increasing proximity to the chimney below the center of the waste cell as shown in Fig. 2. The only potential sources of tritium that could conceivably account for the concentrations found in the samples are the waste placed in the crater and the cavity located 160 m below the crater. The activity of the waste has been estimated to be 1200 Ci, 85 percent of which is attributable to tritium. The other potential source is the tritium remaining in the cavity following the test. The migration of tritium in dry soil was examined by Smiles et al. (1995) (6). Their diffusion model predicts that the movement of tritium from a source in dry alluvium will be limited to a distance of less than 100 m. This result implies that the elevated tritium concentrations under the cell are due to the migration of tritium from the waste. The evidence of negligible liquid flux under the cell provided by the water potential profile and the lack of metals or radionuclides other than tritium suggests that tritium migration from the cell is primarily due to diffusion.
The similarities in water content, water potential and stable isotope ratios between undisturbed alluvium and alluvium below the cell and the absence of waste constituents below the cell demonstrate that for this crater no enhanced recharge has occurred. The implications are that nuclear subsidence craters with minimal catchment area outside the crater boundary can provide a means for shallow burial of low-level bulk radioactive waste that will effectively isolate the waste from the groundwater below.
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Session 33 -- INTERNATIONAL SPENT FUEL & VITRIFIED HIGH-LEVEL WASTE MANAGEMENT
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33-1

STUDY ON SPENT FUEL AND RADWASTE MANAGEMENT POLICY FOR THE REPUBLIC OF KOREA

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ABSTRACT
The Republic of Korea has ten operating nuclear plants and six under construction and another seven planned. Spent fuel and low level radwaste has been accumulating at the nuclear plants since the start of operation of the first plant in 1978. This study looked at several options for spent fuel storage and different disposal facility with proposed operating dates and radwaste processing options to provide guidance in the planning and policy making with regards to storage and disposal in the future.
This paper will describe the study parameters, various scenarios under consideration, different away from reactor spent fuel storage design concepts, and different operating dates spread over several years. Postponing the operating date for the disposal facility greatly increased the storage needs.
Many key factors involved in the study and had different influences on the outcome of the study, including the options recommended to help formulate future policies for the back end of the fuel cycle.

INTRODUCTION

Electricity from nuclear power began in the Republic of Korea with the commercial operation of Kori Unit 1 in 1978. The plant owned and operated by the Korea Electric Power Company (KEPCO) was the first of the ten operating nuclear plants in Korea. Six more plants are under construction and seven more are planned for operation by June 2006. The plants are

listed in Fig. 1. The two types of plants in use and planned are either pressurized light water reactors (PWRs) or heavy water moderated reactors (PHWRs).

Fig. 1

There are no disposal sites for either spent fuel or low/intermediate level radwaste, and all radioactive wastes are temporarily stored at the nuclear plants. Because the spent fuel pools at the older existing nuclear units are getting full, a need for updating the plan and schedule for the safe management of spent fuel was needed. The Nuclear Environment Management Center (NEMAC) of the Korea Atomic Energy Research Institute (KAERI) authorized Sargent & Lundy in the fall of 1994 to jointly study several options for spent fuel storage and different spent fuel disposal facility operating dates along with low and intermediate level radwaste processing options. The purpose of the study was to study the most appropriate option for the Government with regards to storage and disposal in the future.

The focus of this particular study was to develop feasible scenarios for long-term management of both spent fuel and radwaste, focusing on the once-through or direct disposal of spent fuel, and the effects of volume reduction on disposal of low and intermediate level radwaste (LLW/ILW). Reprocessing was excluded as a spent fuel option for this study.

The period of time for the study covered from the year 2002 when a central spent fuel storage facility, an away from reactor (AFR) facility is assumed to be operational, until 2080. As stated earlier, the planned nuclear plants and those already under construction would be in operation by the year 2006. Figure 1 lists the existing and future plants until 2006. For purposes of the study, nuclear plants of the "Advanced" types, APWR and APHWR, were assumed to be placed in service starting in 2007. Between 2007 and 2030, the last year of new nuclear plants, 19 APWRs of 1000 MWe would be installed and five 700 MWe and three 1000 MWe APHWRs. The operating life of the conventional nuclear plants was assumed as 30 years and for advanced plants as 40 years. The last of the advanced plants would be taken out of service in 2070. The spent fuel would be stored onsite for five years and all of the fuel would be shipped to the repository by 2080. Electric generating capacity from nuclear plants increases from 8,616 MWe at the end of 1995 to a peak capacity of 32,300 MWe in 2026.

To temporarily solve the shortage of AR storage, KEPCO the sole national electric utility in Korea, exerted a great effort. For example, Wolsung Unit 1 a PHWR operating since 1983, has already installed some dry storage casks for spent fuel storage. KEPCO may also tranship some spent fuel from one unit to another and from one plant to another.

SPENT FUEL DIRECT DISPOSAL OPTIONS

Several countries with operating nuclear plants are planning on direct disposal of spent fuel. When reviewing direct disposal of spent fuel in a geologic repository, several scenarios were brought into consideration; different operating dates for a high-level waste repository for intact fuel assemblies, fuel rod consolidation prior to storage and disposal, and disposal of the PWR spent fuel assemblies separately from PHWR spent fuel with a different operating date for disposal for the two types of reactor spent fuel.

Each scenario for direct disposal of spent fuel includes some spent fuel storage. Storage includes the use of the spent fuel pools at each nuclear plant and a planned central facility to provide away from reactor (AFR)

spent fuel storage from several nuclear plants. The length of time for storage at the AFR and the amount of spent fuel stored will be dependent on the operational date of the disposal facility. The amount of time that spent fuel is stored prior to disposal, can change the economics of the fuel cycle. The design of the repository may be changed, if the spent fuel has been stored 40-50 years versus 10-15 years because of the reduced decay heat and lower radiation levels of the older fuel. However, the structural integrity of the fuel elements may be less with a longer storage time.

The longer storage scenario allows for more time to collect funds for the cost of the repository, but it increases the cost for storage of the spent fuel. The longer storage allows for more research and development to determine optimal disposal facility design and allows more time for other countries to obtain experience with their high-level waste repositories, which could benefit Korea.

Three different scenarios were selected for further study along with two different subscenarios. Variations in the economic parameters used for the scenarios was also considered for the discount rate, ie., cost of money minus inflation. Three discount rates, 2%, 5% and 8.5% were used for each of the three scenarios. The basis for these discount rates is discussed later.

The three direct disposal scenarios are based upon the timing of the start of operation of the waste repository:

Scenario 1 Early Introduction in 2030
Scenario 2 Intermediate Introduction in 2050
Scenario 3 Late Introduction in 2070

As discussed earlier, the basis for the earliest date for operation of the repository (2030) was selected as probable earliest date that a repository could be operational in Korea. While it might be possible to have a repository operating earlier, there are so many variables that could delay the start of the facility, 2030 appears to be a practical date that could be met.

The late introduction date for the repository was selected as 2070. The input information for the study projected the last advanced nuclear units being added in 2030. With a lifetime of 40 years for the advanced design reactors, the last reactors would be retired in 2070.

The intermediate introduction date was selected as 2050, a date halfway between the early and late introduction dates. By using three dates, the economics for the range of different operating dates was made. The three dates are also within the range of dates planned by other countries for their high-level waste repositories. Some countries plan to be earlier than 2030 and some later than 2070.

Two of the reasons to operate the repository early are regulatory and public opinion. Nuclear regulatory officials in the U.S. have always believed that nuclear waste, either high or low level, should be placed in a repository as soon as possible, as it is considered the safer option than storage. From a political standpoint, many anti-nuclear advocates have used the argument that the nuclear industry does not have a solution for disposal of high level waste, and therefore should not construct (or operate) any more nuclear plants until the nuclear industry has a place to dispose of the waste. Others feel that the disposal facility is needed sooner rather than later, not to put the burden of disposing the waste to future generations of people. However, spent fuel and low/intermediate level waste has been stored safely for 25 to 30 years.

INTERIM STORAGE

Interim storage is required to some extent with all nuclear fuel cycle back end options, reprocessing or direct spent fuel disposal. The flow diagram for interim storage in the back end of the fuel cycle is shown in Fig. 2. Earlier studies by KAERI, selected that the spent fuel capacity for the first AFR would be 3,000 metric tonnes equivalent (MTU), and that it would be a wet pool type. Subsequent AFR's would be of the dry type.

Fig. 2

The size of the subsequent AFR's was not predetermined. Early in the study, it was thought that there may be some cost advantages to add AFR capacity in the same size as the first AFR, reducing the engineering and construction costs. However, with the amount of spent fuel being generated in the future, It came to the point where an AFR had to be added every couple of years. The final choice for AFR capacity, was to make the second AFR twice as large, 6,000 MTU and the third and subsequent AFR's with 12,000 MTU capacity. The flow chart to show the dates for AFR additions and a projected second AFR site, is shown in Fig. 3. The number of AFRs increases if the repository startup date is later. The early operating date for the repository would require only one AFR site with study parameters chosen for the AFR site. Later dates would require a second AFR site to be selected and developed.

Fig. 3

In the U.S. some nuclear plants have reached the limit of the fuel pool capacity with re-racking and have turned to the use of dry concrete casks for additional spent fuel storage capacity. Because the many nuclear plants are owned by many different electric utilities, the central AFR for storage has not been contemplated. A big advantage to using the dry concrete casks for the individual nuclear plants is that several different dry cask designs are pre-licensed for use. This is very cost effective for the U.S. utilities versus the cost required to license an AFR. Also the different timing for the need for more spent fuel storage makes it difficult for a utility to construct an AFR. U.S. utilities have been promised by law that a monitored retrievable storage (MRS) facility operated by the Department of Energy would be available in 1998. At this point in time utilities are not expecting the MRS for quite a few more years. In Korea, with one electric utility and less transportation problems, a central AFR is feasible.

This study focused on the different scenarios for direct disposal of spent fuel. However, the cost for constructing additional spent fuel storage by expansion of the original AFR in steps to accommodate more spent fuel, was an important step to the economic evaluation. The costs for spent fuel AFR storage varied by the type of storage method chosen, but only one dry type storage cost estimate was used in the economic study for disposal costs. The timing of the storage cost varied with the date of operation of the spent fuel repository, and that became a factor in the economics.

Some countries are using wet storage for central AFRs, with dry storage being used in some. Future plans for some countries show consideration for dry systems. In the U.S., commercial nuclear reactor plants are using dry cask storage of three different type designs. Two other dry storage designs are the air cooled metal or concrete cask which may store 20 to 40 fuel assemblies and the modular dry vault storage systems. The dry storage concept is usually based upon natural convection cooling without forced air. The building design for the modular vault could be designed

for either natural for forced air depending on the heat removal requirements.

LOW AND INTERMEDIATE-LEVEL WASTE MANAGEMENT OPTIONS

The low and intermediate-level waste management portion of the study reviewed the projected LLW/ILW quantities of existing and future nuclear plants and reviewed volume reduction technology. Waste quantities for the future decommissioning of nuclear power plants (NPPs) is a significant amount and was included in the projected amount of waste. Waste generated by the storage of spent fuel is negligible as compared to the normal operations waste quantities and that of decommissioning.

Volume reduction of LLW/ILW is recommended for the NPPs to reduce the cost of developing and operating an LLW/ILW repository. Every NPP does not require all of the possible volume reduction systems. Some major systems such as incinerators should be located initially at two NPPS and supercompaction could be located at one facility. The economic evaluation of volume reduction was very beneficial in reducing disposal costs and, to a lesser extent, transportation costs. It was recommended that volume reduction equipment be purchased and installed as soon as possible, to also reduce the cost of building and operating storage facilities at the NPPs.

SUMMARY

The economic analysis considered the sensitivity of the discount rate for money in Korea of 2%, 5%, and 8.5%. These discount rates had been used for other studies in Korea and were selected to provide a range of rates to show the effect of the discount rate on overall economics for selecting the direct disposal scenario. The total cost estimate for the HLW repository was estimated at $11,000,000,000 in 1994 dollars. This cost estimate was based primarily on cost estimates published in Swedish technical reports. American reports were also reviewed as to their cost estimates. Because no country has constructed or operated an HLW repository, the actual cost could vary due to many factors. As another sensitivity factor to selecting the scenariooptimum disposal operating date, the economic analysis varied the estimated cost of the repository by $5.0 billion. That made the range of repository costs used in the study $6 - $16 billion.

With both sets of sensitivities considered, the economic analysis slightly favored an operating date of 2070 over the 2050 date. The non cost factors favored an earlier operating date rather than a later date, by a wider margin than the economic difference. Therefore, the recommended scenario for the HLW repository for direct spent fuel disposal, was with an operating date in 2050.

The recommended scenario for spent fuel management was to provide away from reactor centralized spent fuel storage facilities until a high-level waste (HLW) repository can begin operation in 2050.

Considerable R&D activities are required prior to the development of an HLW repository and some items should be started in the next few years.

33-2

A NEW GLASS VAULT INTERIM STORAGE AT LA HAGUE AND ITS ADAPTATION TO MULTIPURPOSE STORAGE FACILITIES

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ABSTRACT

The La Hague reprocessing plants, UP2 and UP3, were equipped with vitrification facilities, R7 and T7 respectively, in which fission products are vitrified for final disposal. Each of these facilities has an interim buffer storage capacity designed for cooling and transport of vitrified residues.

While awaiting geological disposal, French requirements also demand new interim storage capacity in addition to the existing capacity. An installation was also needed to rehandle the stored glass packages, load them into transport casks and ship them to Cogema's clients.

To meet these requirements, a modular interim storage facility called "Entreposage Verres Sud Est" was constructed as an extension of the T7 facility. With an initial capacity of about 4,000 packages (corresponding to approximately 5500 tU reprocessed) and a first module containing two vaults, it can be extended gradually by adding modules of three identical vaults. The facility's main characteristics is its use of cooling by natural convection associated with an innovative storage pit design which enables the removal of up to 24 kw per pit. The facility has its own installation for receiving and rehandling packages by shuttle, with a flow of about 1000 packages per year.

Based on this technology, as well as that used at the CASCAD Cadarache Facility, where exotic fuel from CEA research reactors is stored before reprocessing or final disposal in 50 years, SNG has developed the "Advanced Cascad System" for long term storage of HLW or Spent Fuel.

THE NEW GLASS VAULT INTERIM STORAGE FACILITY

The first module of the new glass vault interim storage consists of two buildings. The first building contains all of the shared equipment and auxiliary system such as utilities, the electrical system, access functions and shuttle reception air-lock. The second primarily contains two interim storage vaults, each with its stack and air intake, and a common handling bay for operations of the transfer crane and the package transfer casks. (see Fig. 1). The bay and the air intake can continue to be used as capacity extensions are added.

Fig. 1

The primary functions performed at the installation called "Entreposage Verres Sud Est" (EVSE-Glass Canister Storage in the Southeast Facility) are:

- package reception in a shuttle,
- package transfer from the shuttle to the interim storage wells,
- package cooling by natural draft ventilation,
- package transfer from the wells to the shuttle,
- package containment in the wells,
- shuttle inspection and shipment.

Package Transfers by Shuttle

The packages enter and leave the facility in a shielded shuttle transported by a flat-bed vehicle. The facility equipped with an enclosure at the loading station, enables the packages to pass between the shuttle and the transfer casks. The enclosure has seven openings corresponding to the seven cavities of the shuttle (See Fig. 2).

Fig. 2

Loading the Storage Facility by Cask and Transfer Crane

The vaults are loaded by a transfer cask carried by a transfer crane. The transfer casks, whose weight is limited to 60 t, is a cylindrical containment consisting of:

concentric and contiguous elements which form the radiological shielding,

a removable upper portion, accommodating a camera and other equipment,

a rotary floor mounted on a ball ring and receiving the packaging and plug support systems,

high reliability electro-mechanical tongs, consisting of five high-strength stainless steel fingers, designed to grab the packages and plugs, operating satisfactorily at 400C.

The transfer crane with a capacity of 700 kN, is of the twin-beam box girder type. Placed on the slab of the storage facility, it has the following components:

a springer with travel and guide rollers,

a perpendicular steering trolley supporting the cask,

drive system for the different travel and lifting functions,

The wells and loading/unloading openings are served in automatic mode. Accordingly distances are measured by laser. Data transmission, including images from the casks onboard camera, is also performed by laser.

During loading operations, the transfer cask is used to handle the biologically shielded plug of the storage wells and to lower the packages inside.

Cooling by Natural Draft

The desire to guarantee optimal safety, and to achieve intrinsically safe cooling of the storage facility, guaranteeing a glass temperature under 510C in all circumstances, led to the selection of a natural draft cooling system.

The design of the EVSE facility from its outset took into account disturbances connected with weather conditions at La Hague (storms, rain and fog, frost, snow and drifts) to ensure that no succession of plausible events could cause the complete shutdown of the cooling system.

The selected design helps to avoid the following causes of failure:

mechanical, since there is no equipment,

electrical, since no power supply is needed,

control, since there need be no control system,

utilities since there are no fluids,

of human origin, since no human action is necessary.

In interest of compactness, packages are stacked 12 high in the wells at short 85 cm intervals. Each vault contains 18 rows of 10 wells.

To remove the heat dissipated in each well (up to 24 kw/well) by natural convection, the following design was implemented. A liner around each well forms a double jacket, and the cooling air circulates in the annular space thus formed. Fresh air, previously filtered through electrostatic filters, enters at the bottom of the vaults at the lower end of the wells, and travels upward in the annular space around the wells. The driving force is provided by passage of air in the double jackets and in the 35 m high stack, designed to withstand earthquake as well as the most violent winds.

A thermally insulated containment supplemented with air circulation protects the structural concrete from exit air temperatures over a range of 140C.

Wind tunnel tests were performed on a mock up, optimizing the shape of the stacks to create the best negative pressure caused by the wind at the top of the stack. The tests also took into account the effect of the site's environment within a 500 m radius. These tests showed that the winds in general, particularly those from dominant sectors, assist the

natural draft by producing temperatures in the vaults that are lower than those obtained without wind.

During the commissioning tests, thermal tests were carried out with heat releases up to 240Kw (10 loaded wells). The results of these tests were in good accordance with detailed calculations performed by qualified computer codes such as TRIO and FLUENT demonstrating a good understanding of all the mechanisms involved and a sufficient safety margin.

ADAPTATION TO MULTIPURPOSE STORAGE FACILITIES

Based on the above EVSE system as well as on the experience gained by SGN in the construction of interim storage facilities for fuels, high level, medium level and low level wastes, SGN has developed the Advanced CASCAD System to offer its clients solutions tailored to their needs. The proposed facilities may be specific to a given type of product (fuel elements or waste) or designed for the interim storage of different types of waste in a single facility (1,2.)

Such is the case for design of the facility to be constructed by the Dutch company COVRA (Centrale Organisatie Voor Radioactief Afval) to store:

- spent fuel
- vitrified reprocessing waste
- cemented hulls and end-pieces, cemented technological waste and bituminized waste from fuel reprocessing
- miscellaneous waste from research reactors and nuclear facilities.

SGN is currently designing this facility with main following characteristics:

The two first types of waste will be stored in 3 vaults, in vertical pits cooled by natural convection, using the technology implemented for the EVSE facility.

The two last types, packaged in cemented containers or stainless steel drums, will be stored in 3 bunkers similar to those implemented at La Hague for such waste.

Storing such various types of products in a same facility presents advantages from a cost standpoint since only one unloading and canistering unit with its associated auxiliary functions (power supply, ventilation, utilities) is needed for all the modules.

CONCLUSIONS

The new EVSE facility commissioned at La Hague for the interim storage of glass canisters presents innovative features such an increased storage density and cooling by natural convection of high heat generating products. Therefore, this facility offers very high technical and economical performance while maintaining high safety standards.

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33-4

ENCAPSULATION PLANT FOR SPENT FUEL IN SWEDEN

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ABSTRACT
Sweden currently has a system in operation that will manage all types of radioactive waste in the country for a long period ahead. The system comprises a final repository for short lived operational waste (SFR), a central interim storage facility for spent nuclear fuel (CLAB) and a transportation system. The remaining system components now being planned are an encapsulation plant and a deep repository for final disposal of spent fuel and other types of long lived waste. The Swedish program for the encapsulation plant and the deep repository is well advanced. According to current plans, the encapsulation plant will be built as an extension to the CLAB facility. The co-location of the two facilities will enable the use of existing infrastructure and staff at CLAB. The fuel handling system of the encapsulation plant can also be directly connected to the storage pools for spent fuel in CLAB.
The spent fuel will be encapsulated in a copper canister containing an insert. The copper canister development work has been going on for more than 10 years and in 1995 full size demonstration canisters were fabricated successfully. The Basic Design for the encapsulation plant is scheduled for completion in 1996. The licensing application is programmed for submission to the Swedish authorities in early 1998. The construction work is expected to start at the end of the century. The first commissioning of the facility would then start in 2005 and delivery of canisters to the deep repository is planned for 2008. The design capacity of the plant is one canister per working day, corresponding to an annual output of approximately 210 canisters. One BWR/PWR canister will contain approximately 2,2/1,8 tons of uranium respectively. The Swedish nuclear program is expected to generate approximately 8 000 tons of uranium and about 75 percent is BWR fuel. The total investment cost for the plant is estimated to be in the order of 1 700 MSEK (260 MUSD).
THE SWEDISH WASTE MANAGEMENT SYSTEM
The Swedish nuclear power utilities are responsible for the safe management and disposal of spent nuclear fuel and other radioactive waste from the 12 Swedish nuclear power stations. In order to fulfil this responsibility, the utilities have jointly created SKB, the Swedish Nuclear Fuel and Waste Management Co, with the task to plan, build, and operate the necessary waste management facilities and systems. SKB has developed a system that ensures the safe handling of all kinds of radioactive waste from the Swedish nuclear power plants for the foreseeable future. The keystones of this system are:

- A transport system which has been in operation since 1983;
- A central interim storage for spent nuclear fuel and core components, CLAB, in operation since 1985;
- A final repository for low and intermediate level, short-lived waste, SFR, in operation since 1988.

The remaining parts of the system, now being planned are:

- An encapsulation plant for spent nuclear fuel;
- A deep repository for encapsulated spent fuel and other long-lived radioactive wastes.

The Transportation System
As all the nuclear power plants, SFR and CLAB are located on the coast and have their own harbours, SKB has developed a sea transportation system. This has many advantages, such as a high load capacity and low interference with other traffic. The system comprises a purpose built

ship, the M/S Sigyn, 10 transport casks for spent fuel, 2 casks for spent core components, 27 transport containers, type IP II, for transporta tion of low and intermediate level waste and 5 terminal vehicles. The terminal vehicles are used for the land transport from the reactor site to the harbour and from the harbour to CLAB or SFR.

Central Interim Storage for Spent Fuel (CLAB)

CLAB is located at the Oskarshamn nuclear power plant on the east coast of Sweden. Operation started in 1985, and at the end of 1995 more than 700 casks containing some 2300 tonnes of fuel had been received. Also, approximately 70 casks with activated core components, eg control rods, had also been received. The arrangement of CLAB is shown in Fig. 1.

Fig. 1

The main complex above ground is the receiving building, where the fuel transport casks are unloaded. The unloading is performed under water. The CLAB storage section is located in a rock cavern, the roof of which is 25-30 metres below ground level.

The present capacity of CLAB, approximately 5 000 tonnes of uranium, covers the requirements until 2004. The Swedish nuclear program is expected to generate approximately 8 000 tonnes of which approximately 75 percent is BWR fuel and 25 percent PWR fuel. CLAB must therefore be expanded by adding storage pools in a new rock cavern close to the existing one. According to the current plan, the construction of the second cavern will start in 1999. The spent fuel will remain in CLAB for approximately 30-40 years to decay before encapsulation.

Final Repository for Radioactive Waste, SFR

The operation of SFR started in April, 1988. It is a repository for low and intermediate level waste, built in the bedrock under the Baltic Sea, close to the Forsmark nuclear power plant. 60 m of rock cover the repository caverns under the sea bed. The first stage, which is in operation, include buildings on ground level, tunnels, operating buildings and disposal caverns for about 60 000 m3 of waste. A second stage is planned to be built and commissioned after year 2000. Until the end of 1995, a total of approximately 18 000 m3 of waste has been disposed of at SFR.

THE DISPOSAL CANISTER FOR SPENT FUEL

Development Work

The spent fuel is planned to be encapsulated in a disposal copper canister with an insert. The canister development work has lasted for more than 10 years and full size demonstration canisters have been fabricated during 1995. The development work with electron beam welding (EBW) of the canister copper lid is done in co-operation with TWI (formally The Welding Institute), Cambridge, England. A preliminary design of a disposal canister with an BWR insert for 12 fuel assembles is shown in Fig. 2. A canister with a PWR insert for 4 fuel assemblies will have the same outer dimensions.

Test welding of full size lids on copper canisters with reduced length was carried out during 1994/95. Figure 3 shows the machining of the inner surface of one of the full size canisters.

The Lid Welding Pilot Plant

A crucial function in the encapsulation plant is the welding of the lid to the copper canister. This will be done remotely with high accuracy, and in such a way that the weld can be checked afterwards by non-destructive testing (NDT). The welding method preferred at present is EBW at reduced atmospheric pressure. A full size pilot plant is planned which

will be used for testing of EBW and NDT equipment for the copper canister lid weld. The experience gained will be used in the continued design work of the encapsulation plant. The welding in the pilot plant is planned to start early 1998 and the results from the pilot plant will be used to support the licensing case.
Fig. 2
Fig. 3
THE ENCAPSULATION PLANT
General
A feasibility study for the Encapsulation Plant was performed in 1993/94 by four different European companies. In June 1994, BNFL plc, (via its subsidiary BNFL Engineering Ltd) England, was selected as the main consultant for the Conceptual and Basic Design of the encapsulation process. In parallel, ABB Atom, Sweden, was selected as consultant for service and auxiliary systems and service areas of the encapsulation plant. ABB Atom was also selected for the Conceptual and Basic Design work for the extension of the storage capacity of CLAB. The coordination between BNFL and ABB Atom and the plant layout work is done by SKB.
The Basic Design of the encapsulation plant will be completed in 1996 and will be the base for the Preliminary Safety Report. The licensing application is programmed for submission to the Swedish authorities in early 1998. The construction work for the encapsulation plant is expected to start at the end of the century. The commissioning operations of the facility would then start in 2005. Delivery of disposal canisters to the final deep repository is planned for 2008. The design capacity of the plant is one canister per working day, corresponding to an annual output of approximately 210 canisters. Core components and reactor internals will also be conditioned in the encapsulation plant. These activities are planned to start approximately year 2020, after evaluation of the first operation stage of the spent fuel encapsulation and disposal operation.
Fig. 4
Plant Layout
The encapsulation plant is planned to be built adjacent to the existing CLAB receiving building. The main functions and the principal arrangement of the plant are shown in Fig. 4. In addition to the handling and process equipment necessary for these operations, auxiliary, service and control systems as well as different workshops and staff facilities are required. Great advantage can be achieved by utilizing the existing corresponding systems and functions in CLAB.
The encapsulation plant consists of a main building connected to CLAB and the existing fuel elevator. The new handling pool in the encapsulation building is parallel to the receiving pools in CLAB. The handling cell and the different process stations are arranged to permit flexibility in the plant layout for future modifications of the plant. During the construction and commissioning the encapsulation plant will be separated from CLAB and will have its own site entrance. Later, during active operation, the encapsulation plant will be fully incorporated with CLAB operation.
The location as an extension to CLAB provides possibilities to extend several existing functions into the encapsulation plant. These functions include the fuel elevator, cooling systems, water purification systems, draining and fire fighting systems as well as electrical power supply. The local control room in the encapsulation plant is staffed only during the daytime shift. The central control room in CLAB is planned to be used

for supervision and limited operation of the encapsulation plant outside the daytime shift. The CLAB central computer may also be used for recording of all relevant data from the encapsulation process. The layout of the encapsulation plant is shown in Fig. 5. The dimensions of the building is approximately 80 m x 70 m x 25 m (LxWxH).

THE ENCAPSULATION PROCESS

The following will present a more detailed description of the functions and work foreseen in the different parts of the plant. The location of the different parts are marked in Fig. 5.

Transfer Channel, Handling and Connection Pools

The storage canisters with fuel (or core components) are transferred from CLAB into the new pool in the encapsulation plant using the existing fuel elevator system, which is already prepared for this extended function. In the pool, checking of each fuel assembly will be done. The checking before placement in the disposal canister will include a gamma measurement. This measurement may also be utilized for the determination of the residual power and safeguards verification of the fuel assemblies. The results of these measurements are compared with expected values and transferred to a computer for confirmation and final selection of fuel assemblies with regard to the maximum allowed decay heat, which is approximately 1.5-2.0 kW, per canister. The selected assemblies are placed in a transfer canister with 12 BWR or 4 PWR loading positions. The transfer canister is moved to the ramp elevator leading to the handling cell.

Fig. 5

Handling Cell

The transfer canister, in the ramp elevator cage, is moved up the ramp and is stopped at a position above the water level and below the handling cell. In this position the transfer canister is allowed to drain. The canister is then lifted out of the cage with an in-cell crane and is transferred to one of the two drying stations in the cell. The fuel is dried for approximately 12 hours by means of hot recirculating air in a closed system with a HEPA filter.

A disposal canister, placed in a sleeve and shielded frame, is docked from the transfer area below the cell. The steel lid of the insert is removed and temporarily stored in the cell in such a way that it is protected from surface contamination. After drying the fuel assemblies are lifted, from the transfer canister, one by one and placed into the disposal canister insert. The fuel assemblies may also be identified to ascertain the final canister contents.

The steel lid is temporary placed back onto the insert prior to undocking. The lid provides shielding and tightness during the transfer from the cell. The shielded frame is also provided with a shielded gate on top. The canister is then ready to be transferred to the next process station for inerting and final lidding of the insert.

Transfer Vehicle and Transfer Area

The disposal canisters are transferred between the different process stations in shielded transfer vehicles. A vehicle consists of a transporter and a shielded frame, which allows docking of the canister in its sleeve to the handling cell and the process stations in a way that provides radiation shielding and prevents contamination of the transfer area. The vehicles are remotely operated and bring the canisters to the different positions in the transfer area. The vehicles can also be brought to the active workshop for maintenance.

The transporter is currently planned to be of air cushion type, which offers flexibility in movement and position adjustment. The shielded frame has a lifting feature which lifts the canister in its sleeve into the docking port of the different stations. The shield is telescopic with an outer and an inner cylinder.

Inerting and Lidding Station

Operations in the inerting and lidding station are performed in order to change the atmosphere in the canister insert and to finally seal the steel lid. The station has a vacuum chamber with equipment for handling and bolting the steel lid and is connected to systems for vacuum and inerting with argon or helium gas.

The filled canister with the steel lid temporary fitted arrives below the station on the transfer vehicle and is docked to the chamber. The steel lid is lifted to permit the exchange of the air with the inert gas. After gas filling, the steel lid is placed back on the insert and is bolted tight.

The seal between the lid and the insert provide gas tightness during the subsequent welding of the copper lid.

Welding Station

The transfer vehicle brings the filled canister to the welding station where the canister is docked to the vacuum chamber from below in such a way that vacuum conditions can be achieved at the welding area at the top of the canister.

The vacuum chamber contains equipment for cleaning the weld surface and for welding. The copper lid is posted into the chamber before the canister is docked and positioned.

The canister is rotated during the welding process. The EBW equipment is fixed on an adjustable mount so that the beam can track the joint between the lid and the cylinder during welding.

The welding starts with a gun conditioning and testing using a copper target situated directly above the canister. This enables the welding parameters to be adjusted immediately before welding of the canister. The copper lid welding is then completed following a pre-set procedure.

NDT and Machining Station

In this station the copper lid weld is machined and checked with non-destructive testing (NDT). The station is provided with equipment for machining of the welded zone and for ultrasonic and radiographic testing. There is also equipment for machining for repair of failed welds. The canister arrives from the welding station in the shielded frame and is docked to the station from below.

Canisters with failed welds are either rewelded immediately or put aside and stored for later rewelding and repair. Canisters rejected after repeated rewelding can be cut open in the NDT and machining station. The fuel will then be removed in the handling cell after opening of the insert. The insert can be reused but the copper canister will probably have to be scrapped.

Monitoring and Decontamination Station

From the NDT and machining station the canister is brought to the export port of the transfer area. There it is lifted up from the shielded frame by a shielded handling machine belonging to the buffer store. This machine transfers the canister and lowers it into the monitoring and decontamination station. The canister surface is checked with remote smear testing and if contaminated the canister is cleaned with high

pressure water. Smear test samples are brought from the station remotely through a shielded opening and measured locally at the operator platform. The clean canister is transferred from the station to the canister buffer store by the shielded handling machine.

Buffer Store

The buffer store serves as a buffer for filled canisters prior to transport to the repository. The canisters are stored under a shielding floor with shielding plugs over the storage positions. Thick walls protect the surrounding area from radiation. Cooling is provided using air on a once-through basis. The shielded handling machine place the canister into an empty storage position after first lifting the shield plug. Before leaving the position the machine replaces the plug. When a canister is to leave the store the machine moves it into the cask entrance area where it is lowered down into a transport cask.

Core Components

The conditioning of core components in concrete moulds will have a route similar of that of the disposal canisters. The storage canisters with core components are dried in the handling cell and then placed in a mould in a shielded frame similar to the ones used for canisters. The main outer dimensions of the concrete mould could be 1.2 m x 1.2 m x 4.8 m and with a wall thickness of 200 mm. The lidding of the mould and the eventual filling of the void with grout will be done in separate lidding and grouting stations. The monitoring and decontamination and the buffer store with the shielded handling machine, are planned for use for both canisters and moulds.

Transport Cask Loading

The buffer store handling machine is used for transferring canisters and moulds into the transport cask. A special docking position and port in the storage area is provided for this operation. Below the position a transport cask is positioned for receipt of a canister.

The Dispatch Hall

The dispatch hall is divided into a large area for transport and temporary storing of goods, connections to the buffer store and a transport cask maintenance area. A passage from the hall leads to the active workshop. The hall has an air lock for entering and dispatching goods. The dispatch hall has equipment for lifting transport casks and containers and for handling the empty canisters and moulds. The empty canister will be tilted vertically and placed in a sleeve, checked and prepared for positioning into the shielded frame. There are also general areas for storing empty canisters and moulds. Items to be transported are transport containers for empty copper canisters, casks for filled canisters, containers for empty moulds and transport cask for filled moulds. Copper lids for canisters and EBW copper test targets are also handled in the hall.

The transport procedure for filled canisters starts with a transport cask in the dispatch hall air lock. After removing the shock absorbers, the cask is raised to a vertical position and lifted into a bogie with a surrounding platform. The platform enables the bolts of the lid to be removed. The bogie is moved into a shielded export area where a filled canister is lowered directly into the cask with the buffer store shielded handling machine. The cask lid is positioned, the cask withdrawn from the loading bay and the lid is tightened.

SUMMARY

The plans and design of the encapsulation plant are well advanced. According to current time schedule, the commissioning will start in 2005 and the first copper canister delivery to the deep repository will be in 2008. The production target for the encapsulation plant is one canister per working day corresponding to an annual output of about 210 canisters or encapsulation of approximately 400 tonnes of uranium per year. The Swedish nuclear program is expected to generate approximately 8 000 tonnes. The total investment cost for the encapsulation plant is estimated to be 1 700 MSEK (approximately 260 MUSD).

33-5

STORAGE, EMPLACEMENT AND DISPOSAL OF RADIOACTIVE WASTE IN THE U.K.

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ABSTRACT

This paper describes the current treatment and storage approach for the management of low level and intermediate level radioactive waste in the UK. The current Nirex conceptual design for a deep waste repository incorporating vaults at a depth of 650m below ordnance datum is described.

The emplacement of the waste within underground vaults is discussed together with the envisaged vault construction methods.

Finally, the principle of the Nirex disposal concept is included together with an indication of the current status of the plans for a deep waste repository in the U.K.

Nirex have been investigating a site near Sellafield in west Cumbria and a public inquiry into planning application for a Rock Characterization Facility has just concluded and Nirex anticipate that a decision on this phase of their proposed development will be made in 1997.

INTRODUCTION

During the latter part of this century significant resources have been devoted to the safe storage and disposal of the UK's radioactive wastes. Of prime consideration is the length of time for which the waste should be isolated from man's environment, a time-frame considerably longer than any for which human society must usually plan. Sea Disposal was used for some LLW, between 1950 and 1982, but the bulk of these wastes have been, and continue to be, disposed of at BNFL's Drigg site in West Cumbria and a similar but smaller facility at UKAEA's Dounreay site in the NE of Scotland. Most radioactive wastes continue to be stored at the sites at which they arise.

A solution which tackles the required isolation of the waste, long term, is to bury the waste underground where the rock provides a suitable host environment. An underground excavation or 'repository' created in a suitable host rock must be engineered for the safe, efficient and cost-effective emplacement of radioactive waste and its ultimate disposal. This paper discusses the current arrangements in the UK for the storage of solid, intermediate level and low level wastes and the technology being applied for its final emplacement and ultimate disposal in a deep waste repository.

RADIOACTIVE WASTE IN THE U.K.

Radioactive waste represents 0.04% by volume of the total waste generated in the United Kingdom. Of a total annual radioactive waste generation of approximately 46,000m3, 44,000m3 is low level, 2000m3 is intermediate

level and 30m3 is high level. This compares with 4,500,000m3 of solid and liquid toxic waste generated annually (1).

CURRENT ARRANGEMENTS FOR LOW LEVEL WASTE PACKAGING AND DISPOSAL

Low level waste (LLW) has a small radioactive content and requires few special precautions to be taken by those handling it. LLW is typically made up of paper, packaging materials, plastic sheeting, protective clothing, scrap metal and building rubble. It originates from a variety of sources, principally reprocessing activities at BNFL Sellafield nuclear power stations, hospitals, research establishments and other industry.

LLW has been safely disposed of at the Drigg disposal facility in Cumbria since 1959. This 110 hectare site is owned and operated by BNFL and located 6 km south of Sellafield. Since it began operations, Drigg has received in excess of 750,000 cubic meters of waste.

Since the late 1980's disposal of LLW at Drigg has been within a purpose built, reinforced concrete lined surface vault designed to accept containerized waste in a form that will extend the useful life of the site well into the 21st Century. The waste is placed within the vault in steel containers that are stacked in an orderly manner resulting in good utilization of the space available. Once the vault has been filled an impermeable layer is placed over the waste and the vault is capped and landscaped to blend with the surrounding environment.

CURRENT ARRANGEMENTS FOR INTERMEDIATE LEVEL WASTE PACKAGING AND STORAGE

The volumes of intermediate level waste (ILW) are considerably less than those for LLW, but the increased activity associated with this waste makes ILW more complex to handle and dispose. Typically, ILW comprises irradiated fuel cladding, reactor components, chemical process residues, ion exchange resins and filters. Currently, no final disposal route exists for ILW pending the availability of the deep waste repository. The ILW already in existence is stored in a form that makes it readily accessible for future deep disposal.

One of the ways this is achieved is by conditioning the waste into a manageable and consistent solid waste form by encapsulation within a cement grout matrix in stainless steel drums. Solid ILW is placed inside a purpose-designed 500 liter drum and liquid grout infill added to encapsulate the waste. Following initial curing, the contents of the drum are capped by the addition of a further wet grout mix. After capping, the drum is lidded, decontaminated and monitored prior to storage within an engineered facility.

Liquid waste is encapsulated in a similar fashion, the main difference being that the grout is added to the drum contents dry rather than wet. Drums for liquid wastes incorporate 'in-drum' mixing features to allow the contents to be mixed with the dry grout powders. These drums are subsequently capped, cleaned, monitored and stored in the same way as for the solid wastes.

In addition to conditioning and storage of ILW in 500 liter drums, other standard containers are also in use. Waste is currently being placed in these containers with a view to interim storage and ultimate disposal within a deep repository, and these contain the de-commissioning wastes and items unsuitable for the 500 liter drums.

There are a number of such engineered stores throughout the UK but the Government continues to favor a policy of deep disposal rather than indefinite storage of ILW.

GOVERNMENT POLICY

In the UK, the impetus for site-specific work on radioactive waste disposal dates from the late 1970s. It sprang from a report of the Royal Commission on Environmental Pollution (2) examining Nuclear Power and the Environment. To underpin new investment in a nuclear power generation programme, the Commission saw it as essential to demonstrate safe "permanent disposal" for long-lived radioactive wastes. That statement is as true today as it was then, and, viewed from any angle, safe disposal of the inheritance of wastes from past, present and future nuclear operations is essential.

In May 1994, the Department of the Environment announced a Review of Radioactive Waste Management Policy and the subsequent White Paper published in 1995 (3) re-affirmed that the UK government continued to favor deep geological disposal for intermediate level radioactive wastes.

UK NIREX LTD

Nirex, founded in July 1982, is the company in the United Kingdom responsible for providing and managing new national disposal facilities for solid ILW and some LLW. Since 1987, Government Policy has been that ILW generally should go deep underground, and that the same facility could be extended to take LLW, as a cost effective alternative to developing a "green-field" site.

The principle of deep disposal of smoothwall radioactive wastes is recognized internationally and is reflected in the radioactive waste policies of many countries . Several countries including the UK are already disposing of radioactive wastes in near surface or excavated facilities. Most of these countries are now considering developing a deep repository.

Nirex has no responsibility for high level waste disposal.

REPOSITORY DESIGN - EMPLACEMENT CONCEPTS

The bulk of the LLW generated from the UK Nuclear Industry will continue to be disposed of at Drigg. The current planning basis for waste to be disposed of in an underground repository is a range of 200,000m3 to 275,000m3 of radioactive waste. This includes 15,000m3 of LLW containing long-lived radioisotopes, making this waste unsuitable for disposal at Drigg.

Since 1987 Nirex has concentrated on identifying suitable locations for a deep disposal facility in the United Kingdom for intermediate and low level wastes. In March, 1989, the company announced its intention of undertaking detailed geological investigations, including the sinking of boreholes, at Sellafield in Cumbria, North West England and at Dounreay in Caithness, Scotland. The investigation established that both sites have the potential to meet the demanding safety requirements for a deep repository, but, in July 1991, Nirex announced that it was to concentrate further investigations at a site adjacent to BNFL at Sellafield.

The current Nirex design concept is based on drift tunnel access running underground to a depth of about 650m below ordnance datum from near BNFL's Sellafield works, where approximately 65% of the waste for deep disposal orginates.

There are numerous types and configurations of underground spaces that could be used for emplacement of radioactive waste, however the current Nirex preferred concept is to emplace the waste in cavern vaults deep underground. This concept design is under constant review.

Fig. 1

LLW Vaults

The cross sections of the vaults will be influenced by the size and configuration of the waste packages in order to make maximum use of the available space within the excavation profile. LLW packages are typically standard ISO freight containers with nominal dimensions 2.4m(w)x2.2m(h)x 4m(l) and can be stacked up to 6 high. At the Deep Waste Repository, Nirex propose to adopt a similar philosophy to that at Drigg of local manual control for LLW emplacement equipment. Packages arriving at the underground facilities from the drift transport system will be off-loaded using an overhead crane and temporarily parked in a buffer store area adjacent to the emplacement vault. A heavy duty forklift truck will be used to collect the packages from the buffer area and transfer them into the vault. This form of handling allows the maximum use to be made of the vault excavation cross section as illustrated in Fig. 2. A single vault is currently envisaged to accommodate the total planned LLW capacity. Some self-shielded ILW may also be emplaced alongside the LLW where appropriate.

ILW Vaults

Higher radiation levels for the unshielded ILW packages preclude local manual intervention in the waste emplacement operation. Once the waste is removed from the heavily shielded containers in which it is transported to the repository, unshielded package handling must be carried out via remote control. Inlet facilities enable remote removal of the waste from the transport containers for onward transfer to the vault. Handling of waste within the vault is by overhead crane enabling remote, accurate placement of packages to be achieved. The majority of the unshielded ILW for disposal is in the form of 500 liter drums, these are handled using a stillage which contains four of these drums, this lends itself to efficient overhead handling and is currently handled in this manner during storage. Since an open crown space is required above the waste stack to accommodate the crane structure, unshielded ILW emplacement by this technique will lead to less efficient utilization of the excavated vault cross section compared to the LLW vault (Fig. 2). Approximately 6 vaults will accommodate the planned ILW capacity over the 50 year operational period.

Fig. 2

Characteristics of the Host Rock

The geotechnical setting within which the repository is to be excavated has a major influence on the excavated dimensions and orientation of the vaults. The rock mass quality and structure, the intact rock quality, discontinuity geometry, in situ stress, rock behavior characteristics, and hydraulic conditions are just some of the factors which must be considered. Hence the design process has a significant interface with the Nirex site investigation program.

Vaults appear to be the most popular form of underground excavation with countries such as Sweden, (Forsmark) Finland (Louiisa), and the USA (WIPP) having such excavations as part of their nuclear waste repositories.

Construction

The general design of a layout incorporating vaults is relatively simple and practicable with uncomplicated waste, ventilation, drainage, spoil removal and backfilling routes, together with easy accommodation of services and ancillary facilities. This assumes that the host rock has no features that preclude such a layout. A general principle to be adopted in the construction of vaults is to establish as quickly as possible a

connection between the access roadways (pilot headings). These pilot headings provides an extra free face for blasting and can also be used to prove the geology ahead of the main excavation. Having established a ventilation circuit, and a means of spoil clearance, the connection can be enlarged to the required dimensions of the vault by a series of benches worked progressively along the length of the vault.

Vaults can be constructed with either conventional drill and blast techniques but peripheral damage to the excavation profile would need to be controlled by the use of appropriate techniques, such as smoothwall blasting.

The nature of the project demands a different approach from that usually applied in underground mining, although many of its functions are similar. The development of vaults to ensure structural integrity would necessarily be to a standard that took into account the restrictions on access once the emplacement of waste had begun. Also the layout of the repository will have to reflect the need to excavate and furnish new vaults at the same time as waste is emplaced in constructed areas. Prime consideration is given to the ventilation circuits for both emplacement and construction areas and the ability to isolate these as necessary.

ALTERNATIVE VAULT CONCEPTS

At this stage of repository development Nirex has progressed design to the concept stage only and must keep an open mind to the design of emplacement systems for a deep repository. The following alternative emplacement systems have been considered by other waste disposal agencies in other countries.

Silos

An alternative emplacement arrangement for unshielded ILW is the use of a silo and a shielded package handling machine similar in principle to that used in the Encapsulated Product Stores at Sellafield. A feature of the silo design is the shielded 'cap' over which the emplacement machine operates. The cap supports an array of shield plugs which are removed by the emplacement machine to enable access to the silo. The shield plug is removed from the top of the channel, the waste emplaced in the silo and the plug replaced. The major benefit of the shielded emplacement machine and the shielded plug cap arrangement is in the unhindered man access afforded by the nature of the operation. Silo dimensions are influenced by a number of factors including package integrity, materials handling, backfill placement and ventilation. To date silos have been incorporated in the underground repositories in Sweden (Forsmark) and in Finland (Okiluto).

They can have distinct advantages over vault options should the area of suitable host rock be limited due to adverse geological and geotechnical features. The geotechnical issues relating to silos will be similar to that for the vaults. Silos can be accommodated in a smaller plan 'footprint' but may be influenced by changes in vertical stratigraphy.

Silos can be constructed by conventional drill and blast 'blind' shaft sinking technology with spoil removal via the top of the silo. The diameter and depth of the silo would not be constrained by the method of construction. Alternatively, if a series of silos are being constructed, then the provision of an additional access at the base of the silo for spoil removal would allow the use of raise boring equipment. This involves the drilling of a pilot hole which would be reamed to the required diameter.

Boreholes

Another potential emplacement option involves the adoption of large boreholes drilled in modules which afford similar construction advantages as the silo option. The construction of these boreholes would be by large diameter drilling techniques using raise boring techniques.

THE NIREX DISPOSAL CONCEPT

Nirex, in common with other disposal agencies internationally, has developed a concept of deep geological disposal for radioactive wastes which uses a multi-barrier containment system. It is implicit in the requirements that Nirex should provide a disposal system that does not rely on monitoring or intervention to ensure safety once the repository has been filled and closed.

The concept makes use of both engineered and natural barriers, working in conjunction to achieve the necessary degree of long-term waste isolation and containment.

A physical barrier within the repository will be provided by the steel and concrete packaging within which the radioactive material is immobilized. The long-term containment properties of the engineered system in respect of radionuclides dissolved in groundwater will stem from the establishment of uniform chemical conditions and high sorption capacity across the repository. This will be achieved by surrounding waste packages with the required amount of a cement-based backfill, Nirex Reference Vault Backfill (NRVB) which has been developed and patented by Nirex to meet the needs of the long term safety case.

The multi-barrier containment system is illustrated in Fig. 3.

Fig. 3

NIREX ACTIVITIES

The engineered repository system combines nuclear technology for the conditioning, handling and care of radioactive waste packages and mining technology in terms of providing underground openings for the emplacement and ultimate disposal of these wastes.

The final repository design and configuration will be dependent upon a number of inter-linked parameters, the most important being an understanding of the geological environment.

In 1992 Nirex stated its intention to undertake underground laboratory work in the first stages of repository development. The purpose of that work would be to complement investigations conducted from the surface and so develop a fuller understanding of the surrounding geology and hydrogeology. That understanding would then provide a basis for refining analysis of long-term safety and geotechnical information that would be required for subsequent detailed design and construction of a repository.

Based upon the results obtained from the continuing programme of regional borehole drilling, a planning application for an underground Rock Characterization Facility (RCF) was submitted in July 1994.

The purpose of the RCF is to provide:

a) rock characterization information to permit firmer assessment of long-term safety.

b) information needed to decide the detailed location, design and orientation of a repository.

c) geotechnical data to inform decisions on repository construction methods.

A public inquiry into the planning application for the RCF has just been concluded and Nirex anticipate that a decision on the outcome will be made in 1997.

The UK Government continues to favor a policy of deep disposal rather than indefinite storage for ILW and considers it appropriate that Nirex should continue with its programme to identify a suitable site. It has decided that once a suitable site has been found, it should be constructed as soon as reasonably practicable. The precise timetable will depend on the granting of planning consent and compliance with regulatory requirements including the establishment of a sound safety case.

CONCLUSION

Through continued site investigation, scientific and engineering programmes and ongoing design development, Nirex will increase its understanding to ensure the most appropriate repository system is developed for operation.

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33-6

FAST FLUX TEST FACILITY SPENT FUEL OFFLOAD TO DRY CASK STORAGE

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ABSTRACT

The United States Department of Energy owned, Westinghouse Hanford Company operated, Fast Flux Test Facility (FFTF) is one of many facilities located on the Hanford Site in southeastern Washington State. The FFTF is a 400 MWt liquid metal-cooled research reactor which began operation in 1980. In December, 1993, the Department of Energy ordered Westinghouse Hanford Company to initiate the transition of FFTF to permanent shutdown. Accordingly, Westinghouse Hanford Company prepared and implemented the FFTF Transition Project Plan which delineates the progression of activities required to place the reactor facility into a long-term, low maintenance, industrially safe shutdown configuration pending final decontamination and decommissioning. The primary objective of the Plan is to provide the logic/approach and project baseline for the transition in a cost effective and quality manner while ensuring safety, security, and environmental compliance.

One of the challenging activities of the FFTF Transition Project is to offload the inventory of highly radioactive spent reactor fuel assemblies to above-ground dry storage casks. This paper will present an overview of the FFTF fuel offload, including a description of the interim dry cask storage system and the processes required to transfer the fuel assemblies

from sodium pool storage to dry cask storage. Additionally, the paper will discuss pertinent design philosophies and key regulatory criteria that are applicable to the FFTF dry cask storage system. The design features that were implemented to address the concern of fuel cladding degradation, will also be addressed.

INTRODUCTION

The FFTF is the largest, most modern, liquid metal-cooled test reactor in the world. Originally constructed to support the U.S. Liquid Metal Fast Breeder Reactor Program, the FFTF supported various missions from 1980 to 1992, including both national and international breeder reactor programs, production of medical and industrial isotopes, material testing for the fusion and space programs, and providing customized neutron environments to meet a variety of customer's needs. In December, 1993, the FFTF was ordered to shut down when the Department of Energy concluded there was no possibility of financial viability and the reactor was no longer needed to support its missions. The FFTF Transition Project Plan (1) was developed which delineates the activities required to place the reactor facility into a long-term, low maintenance, industrially safe shutdown configuration for long term surveillance and maintenance prior to final decommissioning. Approximately seven years are currently estimated to complete the transition of the FFTF to this shutdown state. The major activities to accomplish this task include: reactor defueling, fuel offload to dry cask storage, Sodium Storage Facility construction, sodium drain from the plant systems, and auxiliary plant systems shut down.

Fuel offload from sodium pool storage to dry cask storage, the subject of this paper, forms a significant portion of the overall integrated FFTF Transition Project. This activity includes implementation of the new hardware and facility modifications that are required to transfer the fuel to dry storage casks. The FFTF dry storage system hardware includes the Interim Storage Cask and the Core Component Container packages, which are combined to store up to seven fuel assemblies each. After loading with fuel, the Interim Storage Cask and Core Component Container packages will be placed within the newly constructed Interim Storage Area, which is adjacent to the reactor complex. Approximately 55 Interim Storage Casks and Core Component Containers will be required to store the 371 component inventory of FFTF fuel. The facility modifications required to load irradiated fuel into the Interim Storage Casks and Core Component Containers consist of: 1) installation of ion exchange columns in the existing sodium removal/fuel washing system to substantially reduce radioactive liquid waste by recycling the wash water, 2) design and fabrication of 15 new tools and 13 major new components for remote handling operations, and 3) installation of new fuel handling equipment interface capabilities for the Interim Examination and Maintenance Cell, a remotely operated shielded hot cell, and the Cask Loading Station, a below grade dry cask transfer station.

Following completion of the facility modifications and hardware acquisition, an extensive Acceptance Testing Program was initiated on June 21, 1995 to demonstrate the ability to safely transfer irradiated fuel to the Core Component Container and Interim Storage Cask for dry storage using the new, existing, and modified equipment and facility stations. The initial phase of the Acceptance Testing was performed with "mock" assemblies to verify equipment interfaces. The mock assemblies conformed dimensionally to a fuel assembly but were not irradiated nor did they contain fuel. The "cold" testing phase performed all operations

required to receive, load, handle, transfer, and unload both an Interim Storage Cask and Core Component Container using the new interfacing equipment and several new operating procedures developed for the fuel offload process. This phase was also used to provide hands-on training for the hot cell operators prior to proceeding with remote fuel operations. Approval to proceed to "hot" testing with irradiated fuel was granted after acceptable completion of the cold testing phase. The function of the hot testing was to perform shielding verification tests on all new components and to conduct final operating procedure validations in preparation for unrestricted operations. The hot phase of the testing program was successfully completed on January 22, 1996 with the transfer of the first loaded storage cask to the Interim Storage Area. Unrestricted fuel offload activities are expected to begin in early-February to retrieve the remaining fuel that currently resides in liquid sodium pool storage and to transfer these fuel assemblies to the dry cask storage system. Offload of the fuel to the Interim Storage Area is expected to be complete in approximately three years.

FFTF DRY CASK STORAGE SYSTEM

The requirements considered when developing the FFTF dry cask storage system included: capability to implement the storage system within the time-frame required by the FFTF Transition Project schedule, consistency with the requirements established for commercial nuclear spent fuel storage as specified in 10CFR72 (2), ability to demonstrate a safe storage configuration, capability to interface with the facility's existing equipment, flexibility to relocate the storage system to a central Hanford storage complex at a future date, and retrievability of the fuel at any time during its storage life.

Following the DOE shutdown order, it was critical that the dry storage system be provided in a short time-frame to support the aggressive FFTF Transition Project schedule. Until the fuel is removed from sodium pool storage and transferred to the dry storage system, drain of the two sodium-filled fuel storage vessels, shut down of the associated auxiliary systems, fuel offload processes and equipment, and ramp down of the related staff cannot occur. The cost of supporting the sodium pool storage, fuel offload systems, and the associated personnel is on the order of $20 million per year. Consequently, any delay to the fuel offload would have high economic impact. Approximately 55 Interim Storage Casks and Core Component Containers will be required at an estimated cost of $15-20 million to offload the inventory of FFTF fuel. One year of additional storage and maintenance of the necessary support systems and personnel would off-set the cask procurement costs. In addition to the avoidance of the schedule delay costs, dry cask storage also provides a more economic and low maintenance configuration for interim storage of up to fifty years.

The schedule and economic needs for the FFTF shutdown, combined with the existing fuel handling equipment compatibility, resulted in the development of a vertical top-loading Interim Storage Cask that is smaller in size than a typical commercial dry storage cask. The smaller cask size eliminated the need for extensive and time consuming facility modifications. The contract for design, analysis, and fabrication of the Interim Storage Casks was awarded to a vendor that could demonstrate previous Nuclear Regulatory Commission licensing experience for either a transportation or a storage cask. In addition to the requirements of 10CFR72, the design specification also required the vendor to provide a

storage cask to meet the facility- and fuel-specific interface and retrievability requirements. Additional design criteria were imposed to allow for future on-site relocation of the Interim Storage Casks to a central Hanford storage complex. While the Interim Storage Cask can be relocated at Hanford due to the administrative limitations that can be imposed and controlled on-site, the cask does not meet the requirements for licensed shipping casks and cannot be used for off-site transport. Although the Interim Storage Cask is smaller than a typical commercial dry storage cask, the storage system's design basis functional requirements are the same. The primary functions of the dry storage system are; passive removal of decay heat, criticality control, shielding, and confinement. The cask confinement boundary is required to prevent the release of radioactive material, to provide a benign storage atmosphere via an inert gas blanket (helium), and to control the in-leakage of air and moisture to limit any fuel cladding degradation or fuel oxidation reactions such that the condition of the fuel assemblies is preserved for the duration of the storage life. There are similarities between the FFTF fuel and commercial fuel, and additional conservatisms that were incorporated into the FFTF dry storage system, that allow the application of the commercial dry storage concept to the FFTF fuel. The similarities that exist are that both commercial and FFTF fuel are oxide fuel forms and extensive irradiation histories are available for each. Very conservative operating controls were imposed on FFTF fuel during all phases of fabrication, reactor operation, storage, and subsequent handling. During reactor operations, avery sensitive tag gas monitoring system was used to detect cladding breaches and the reactor was shut down whenever an unacceptable breach occurred. The monitoring system was capable of identifying the breached assembly with 100% accuracy. Historically, only four significant cladding breaches occurred during the reactor's operation. These breached assemblies will be disassembled in the Interim Maintenance and Examination Cell and the failed pins identified and encapsulated prior to dry storage.

In addition to the irradiation history of the fuel assemblies, the strict operating controls imposed to maintain fuel cladding integrity, and the cask confinement boundary, a very conservative decision was also made to containerize the FFTF fuel within the Core Component Container. Commercial dry storage does not typically require this boundary, in addition to the cask, unless there are gross cladding defects in the fuel. As a result of this decision, the functions of the Core Component Container are to provide a sealed confinement boundary, packaged handling capability, and long-term geometry control for the stored fuel. The added barrier provided by the container was determined to be desirable for the FFTF fuel for two reasons. First, there is no long term data that is directly applicable to document the FFTF fuel cladding condition following extended time in dry storage. Application of acceptable commercial dry storage data was not considered to be conservative due to the inherent differences between the FFTF fuel assemblies and commercial fuel. Some differences between the fuel assembly types are: 1) the cladding is constructed of different materials (FFTF fuel cladding is stainless steel versus zircalloy for commercial fuel), 2) materials testing was a primary function of FFTF and there are three different cladding materials, 3) the reactor environments were different, 4) the FFTF fuel sustained higher burn-ups than commercial fuel, and 5) the FFTF reactor coolant and fuel storage environment is liquid-metal sodium

versus water for commercial fuel. Secondly, a concern was raised regarding the need for data to support the absence of cladding degradation during extended dry storage. This concern arose due to observations of cladding degradation during post-irradiation examination of a small number of non-FFTF fuel pins at other facilities. This phenomena is known as "hot cell rot" and is thought to be a complex stress-corrosion cracking and fuel oxidation process involving reaction of sodium, fuel, oxygen, moisture, and cesium fission product with the stainless steel fuel cladding. Similar cladding degradation has never been observed during post irradiation examination of FFTF fuel in the Interim Examination and Maintenance Cell. Additionally, the limited data associated with these observations indicate that extended storage in hot cells and air may have contributed to the cladding degradation. Since "hot cell rot" is currently defined as a corrosion process, which requires oxygen and moisture to occur, it is considered unlikely that significant cladding degradation will occur in the environment of the FFTF dry storage system. However, the concern could not be firmly refuted due to the lack of actual data for long-term storage of the FFTF fuel in this manner. Consequently, additional conservatism was implemented into the FFTF dry storage system design.

The Core Component Container (Fig. 1) was designed by Westinghouse Hanford Company. Specific features were incorporated into the design to address the cladding degradation concerns described above. The container has a bolted closure with a metallic seal and separate closed bottom support tubes for each fuel assembly. The support tubes are fabricated from stainless steel and inconel to ensure corrosion resistant storage. The Core Component Container is unshielded, provides seven storage locations, and is designed for remote grappling and dry transfer capability. Its storage tubes and supports provide the geometry to ensure criticality control during handling and storage of the fuel, even in the unlikely case of full water moderation. Although not expected, if the accident condition occurs whereby the fuel cladding degrades, the Core Component Container will provide for safe and controlled retrievability of the fuel. Additionally, for retrieval of the Core Component Container after extended storage, it is required that the actual condition of the storage atmosphere within the Interim Storage Cask cavity be assessed prior to removing the cask's closure bolts. The Interim Storage Cask closure is designed with a penetration port with a quick disconnect fitting that is used to perform this sampling. If the storage atmosphere sample results are acceptable, then operations can proceed normally with Core Component Container retrieval via a shielded transfer cask. If the atmosphere sample results are not acceptable, then appropriate measures must be used to protect personnel and the operations area from potential contamination.

Fig. 1

The Interim Storage Cask (Fig. 2) was designed by General Atomics and Sierra Nuclear Corporation. The contract to design, analyze, fabricate, and test the prototype design was placed in September 1993. The first Interim Storage Cask was delivered in June 1995 after successful completion of performance testing. The Interim Storage Cask design is required to meet the requirements of 10CFR72, without licensing, and consists of a passively ventilated concrete and steel shielded cask with a stainless steel confinement boundary. Maximum weight of the cask, with the 2,268 kilogram (5,000 pound) Core Component Container payload, is

52,000 kilograms (114,200 pounds). The outside diameter of the cask is 216 centimeters (85 inches) and the overall length is 462 centimeters (182 inches). The internal cavity is 53 centimeters (21 inches) in diameter and 373 centimeters (147 inches) in length to accommodate the Core Component Container. The Interim Storage Cask provides a secondary leak-tight boundary for a defense-in-depth approach to confinement of the spent fuel. Coupled with maintaining the inert atmosphere, the cask eliminates the introduction of oxygen and moisture in-leakage that could accelerate spent fuel assembly degradation over the interim storage lifetime. The Interim Storage Cask is designed with a bolted closure which has redundant metal seals. The cavity penetration port in the closure has redundant covers which are independently seal welded after inerting the cask with helium following loading. The confinement capability of the loaded Interim Storage Cask is ensured by helium leak testing both closure seals after assembly and dye penetrant testing both seal welds of the penetration port cover plates. The confinement capability for the liner is assured by a combination of inspection techniques. During fabrication, each confinement weld requires radiographic or ultrasonic inspection and a complete helium leak test. The maximum permissible leakage rate for the Interim Storage Cask confinement boundary is "leaktight" as defined in the American National Standard for Leakage Tests on Packages of Shipment of Radioactive Material, ANSI N14.5 (3). At this leakage rate of 1 x 10-7 scc/sec, oxygen and moisture are prevented from entering the cask. As such, there is no known mechanism for accelerated corrosion of the spent fuel. In addition to the confinement boundary, the Interim Storage Cask also provides the necessary passive decay heat removal, shielding, and structural integrity for safe dry storage of the spent fuel. An impact limiter is located at the bottom of the internal cavity of the cask to protect the spent fuel if an inadvertent drop of the Core Component Container was to occur during loading or unloading. The FFTF fuel to be stored in the Interim Storage Casks has decayed since cessation of reactor operation in April 1992 and a substantial reduction in heat load and fission gases has occurred. As of August 1995, the highest decay heat assembly was 329 watts and the average decay heat per assembly of the fuel inventory was 81 watts. By November 1996, all assemblies will be below the 250 watt decay heat limit imposed for dry storage. The maximum fuel clad temperature predicted for storage of 250 watt assemblies is less than 390C (736F) during the hottest summer day. Even for the accident case, where no credit is taken for the passive ventilation system of the cask, there are no storage system components that exceed any normal condition limit. Based on these results, a high margin of safety for passive heat removal is demonstrated and there are no safety requirements to monitor the ventilation ducts. However, surveillances will be performed to ensure that the ventilation ducts remain open, in order to maintain storage temperatures as low as possible. These conclusions, which were based on the analytical temperature predictions, were confirmed by performance testing that was conducted prior to loading the Interim Storage Cask with fuel. The FFTF fuel cladding temperature limit for storage is 482C (900F). This limit was based on preventing stress rupture of the clad for the duration of the storage life (50 years) based on the most limiting type of fuel clad material. An irradiation history of < 150,000 MWD/MTHM (Pu taken as 29.3% of the heavy metal) followed by a decay period of four years was used to

conservatively bound the majority of the FFTF fuel. This source term also forms the basis for the radiological release calculations. For all design basis accident conditions, the storage system analyses demonstrate there will be no releases. To further demonstrate the margin of safety associated with dry storage, two radiological release accidents were analyzed. These cases are the "beyond design basis" and "hypothetical" radiological release events. The beyond design basis release accident assumed administrative handling limits were violated and this resulted in a cask drop accident beyond the currently analyzed height of 2.44 meters (8 feet). The beyond design basis release resulting from this accident consisted of 100% of the radioactive gas from all the fuel pins and other fission products resulting from exposure of 1% of the fuel material. No credit was taken for either the Interim Storage Cask or Core Component Container in mitigating the event even though such an impact would not be expected to produce a major escape path. The resulting dose consequence determined for off-site and on-site was 4 millirem and 4.5 rem, respectively, well within allowable limits. An additional non-mechanistic or hypothetical accident release was also analyzed. This case was similar to the beyond design basis case above except that it also assumed a higher source term is present and a crushing/shearing of the fuel stored within the cask occurs whereby 100% of the radioactive gas is released and 5% of the fuel is crushed and exposed. Again, neither the Interim Storage Cask or the Core Component Container were assumed to prevent or restrict the release. The hypothetical analysis resulted in a site boundary dose of 160 millirem. Accordingly, it was concluded that the FFTF fuel storage is not a public health and safety hazard, even under the most extremely unlikely hypothetical event.

Fig. 2

The Interim Storage Cask was designed to provide the radiation shielding required for handling operations and above ground storage such that the dose rate at the cask surface is less than 2 millirem/hour and the dose rate at the Interim Storage Area fence perimeter, located 23 meters (75 feet) from the stored casks, is less than 0.05 millirem/hour. Extensive radiation surveys were conducted during the Fuel Offload Acceptance Test which demonstrated the effectiveness of the cask shielding. The total collective dose measured throughout the fuel loading sequence was determined to be 15 person-millirem. This collective dose is considered higher than what will be experienced during normal handling and loading operations because the testing required numerous surveys of transient interfaces and a larger number of personnel observed the activities than will be present during a routine evolution. The total collective dose from the loading and handling of the remaining fuel inventory is not expected to exceed one person-rem.

FFTF FUEL OFFLOAD PROCESS OVERVIEW

In addition to the Core Component Container and Interim Storage Cask, there are several stations and pieces of equipment that perform vital functions of the fuel offload process. These components are best described by their respective functions in the spent fuel offload sequence. This sequence transfers the spent fuel, one assembly at a time, from its initial location in sodium pool storage to its final dry storage configuration. In the final storage configuration, the fuel will be located inside a sealed Core Component Container, which is inside a sealed Interim Storage Cask, which is located in the Interim Storage Area.

The offload sequence starts with removal of a single fuel assembly from its sodium pool storage location and transfer to the Interim Examination and Maintenance Cell using the existing reactor refueling machines. In the argon inerted hot cell, the sodium-wetted fuel assembly is placed into the sodium removal chamber. In this chamber, the sodium removal process is conducted. This process is comprised of injecting moist argon gas into the chamber to react the residual sodium in a controlled manner. Following the moist gas phase, the assembly is rinsed several times to ensure the requisite cleanliness is achieved. Final cleanliness is determined by the conductivity of the final rinse water (i.e., < 20 microsiemens/cm) and visual examination of the assembly. Cleanliness is required prior to long-term dry storage of the spent fuel due to the corrosive and reactive nature of any residual sodium. Additionally, after each assembly has been washed, it is dried prior to placing it into a clean Core Component Container.

When the Core Component Container is loaded with spent fuel assemblies, the cover is installed which mates to the container body crushing a metallic o-ring between them. The argon atmosphere of the hot cell is used as the inserting gas for the Core Component Container. As such, limitations are imposed on cell atmosphere impurities to limit both oxygen and moisture to a nominal 100 ppm. The Core Component Container cover is installed with 12 bolts which are torqued to 21 kilogram-meters (150 foot-pounds) using the hot cell manipulators. After the cover is installed, a pressure decay leak test is performed to ensure proper seal installation and the Core Component Container is transferred into the Solid Waste Cask.

The Solid Waste Cask is an existing transfer cask that is used in conjunction with the facility cranes and an electric rail transporter to provide the necessary shielding and sealed atmosphere to transfer the Core Component Container from the hot cell, located in containment, to the Cask Loading Station, which is located in the Reactor Service Building. The Solid Waste Cask was previously used to transfer non-fuel irradiated components to a waste burial cask. Several upgrades have been completed to provide safe handling of the spent fuel in the Core Component Containers. These upgrades included a new and more reliable control system and the addition of seals, an argon purge system, and a grapple locking device to be used during rail transport to the Cask Loading Station. The Cask Loading Station has also been modified to accommodate dry fuel transfers of the Core Component Container. These modifications consisted of upgrading the structure to meet seismic qualification, providing sealed interfaces with testable connections, providing shielded adapter plates for the Core Component Container transfers, and providing specific interfaces to position the Interim Storage Cask for remote dry loading. At the Cask Loading Station, the Solid Waste Cask uses an electric driven hoist to lower the Core Component Container into the Interim Storage Cask and ungrapple it. Once the unloaded hoist is raised back into the Solid Waste Cask, the shielded closure valves for both the transfer cask and the Cask Loading Station are closed, which allows the Solid Waste Cask to be safely removed. (See Fig. 3)

Fig. 3

After removal of the Solid Waste Cask from the Cask Loading Station, the closure is remotely installed into the Interim Storage Cask. The first step is to attach a lift rod to the 30.5 centimeter (12 inch) thick, 815

kilogram (1800 pound), bolted cask closure and to place it on top of the Cask Loading Station moveable shield valve. Then, additional shielding is installed over the lift rod and closure assembly. Once the proper shielding is assembled, the closure is lifted slightly via the overhead crane which is attached to the lift rod. Once the closure is suspended, the shielded closure valve is opened and the closure is slowly lowered into the Interim Storage Cask. When complete set-down is confirmed, as indicated by the lift rod travel, the shielding equipment is removed and the closure is secured to the Interim Storage Cask by torquing the sixteen closure bolts to 207 kilogram-meters (1500 foot-pounds). This torque is required to crush the redundant metallic seals and to provide the design basis pre-load for both storage and on-site transport. After torquing the bolts, three evacuations and back-fills are performed to inert the cask cavity with high purity helium gas. After inserting, mass spectrometer leak testing is performed to verify the redundant metallic seals of the closure and plug seal on the interseal leak test port are functioning to provide a leak rate of less than 1 x 10-7 scc/sec.
The use of redundant metallic seals in the closure with an interseal leak test port and a removable fitting provides the capability to perform a functional test of the Interim Storage Cask seals. Continuous monitoring of the cask seals is not currently planned due to the expected short duration of fuel storage at the Interim Storage Area prior to consolidation with the other Hanford fuel inventories. Failure of multiple seals or degradation of the fuel is not likely to occur during this short time frame. The number of seal failures that would have to occur to provide a leakage path to the fuel include; failure of the inner and outer metallic seals of the Interim Storage Cask, failure of the Core Component Container metallic seal, and failure of the fuel cladding. Additionally, analysis of the seal performance for normal and accident conditions, including cask drop and tip-over accidents, does not result in seal failure. Therefore, after the cask seals have been certified as "leak-tight" and the pressure is verified within specification, redundant plates are seal welded over the penetration port quick-disconnect fitting. The Interim Storage Cask is then removed from the Cask Loading Station using a spreader bar to vertically lift the cask and transfer it to the transporter. During the transfer operation, shielding and contamination surveys are performed to verify dose rates are within specification and that there is no contamination prior to removal of the cask from the Reactor Service Building. The cask is placed on a doubledrop-deck trailer in a vertical orientation and is secured using an engineered tie-down system. The Interim Storage Cask is transported approximately 183 meters (600 feet) to the Interim Storage Area using a tractor. At the Interim Storage Area, the cask is unloaded from the transporter using a mobile crane and a weather protection cover and tamper indicating devices are installed. (See Fig. 4)
Fig. 4
The sequence described above will be repeated until all FFTF spent fuel assemblies have been offloaded from sodium pool storage to long-term low maintenance dry cask storage. Approximately three years will be required for this evolution. The final configuration will consist of an array of Interim Storage Casks located in the Interim Storage Area. Over the 50 year design life of the dry storage system, surveillances will be performed on the general storage area to monitor for security, perimeter fence dose rates, vegetation growth, build-up of dirt or debris,

condition of the perimeter fence, and adequate lighting levels. Surveillances of the Interim Storage Casks will also be performed to assess inventory control, to clean areas around ventilation screens to ensure storage temperatures remain as low as possible, to perform visual inspections on the general condition of the casks, to perform upkeep of the concrete shielding surfaces of the casks, to perform routine radiological surveys of cask surface dose rates, and to verify the absence of contamination on cask surfaces. These surveillances will be conducted to ensure that the storage conditions remain within acceptable limits.

CONCLUSION

The FFTF dry cask system will provide safe, cost-efficient, and low-maintenance storage and retrievability for the highly radioactive spent fuel in a manner consistent with the standards implemented by the commercial nuclear sector.

ACKNOWLEDGMENTS

This work was performed by Westinghouse Hanford Company for the Department of Energy.

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Session 34 -- STRATEGIES FOR EFFECTIVE PUBLIC INTERACTION
Co-chairs: Margery Olson, Consultant;
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34-1

STAKEHOLDER TRAINING IN NEGOTIATION AND CONFLICT RESOLUTION: A KEY TO SUCCESSFUL PUBLIC INVOLVEMENT

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ABSTRACT

We have seen many advances in the development of collaborative decision-making processes and techniques. The goal of each of these efforts has been to provide stakeholders with meaningful opportunities to become involved in public decision-making. But something is still lacking. Public involvement professionals often report significant conflict. Public managers are often reluctant to bring stakeholders together because of conflict and may conclude that collaborative decision-making simply does not work where it is needed most.

The primary reason collaborative decision-making projects fail is because stakeholders become bound up in conflict. Creating opportunities for stakeholder involvement, then, is only one part of effective public decision-making. Providing facilitation and mediation assistance is another. But mediators and facilitators can only do so much to help stakeholders identify and resolve differences. Another key is through the development of collaboration and conflict resolution skills that begin to create a unique "culture" within the stakeholder group. The parts work together to break down barriers and build agreement: Public decision-

making processes provide opportunities for stakeholders to work together. Facilitators and mediators help guide the process. And collaboration and conflict resolution skills give stakeholders the necessary tools and understanding.

INTRODUCTION

This purpose of this paper is to suggest a strategy for improving the effectiveness of public involvement in industry and government decision-making by training stakeholders in negotiation and conflict resolution skills. As decision-making power becomes more balanced through public involvement processes, industry and agency officials and other stakeholders will have to negotiate solutions and agreements that satisfy mutual interests. Most of us mistake the process of arguing and debating (I win/you lose) with the process of negotiating (we both win).

Negotiation and conflict resolution are critical parts of the public involvement process yet nowhere are we taught these vital skills. As a result, many public involvement processes limp towards unsatisfying compromises or they fail completely due to unresolved conflict.

The public involvement process serves as a "road map" from a starting point to a destination. The process itself is filled with opportunities for stakeholders to work together to resolve differences and reach agreement. Negotiation and conflict resolution skills are a set of tools that stakeholders take with them on the trip and use them, like they would a Swiss Army knife, to solve problems.

THE CONTEXT FOR STAKEHOLDER TRAINING

Public involvement has existed as a recognized discipline since the 1960's, although James Creighton (1) dates the modern era of public involvement to the 1970's. In the environmental field, public involvement can be traced to early legislation and regulations, such as the National Environmental Policy Act (NEPA), which created opportunities for the public to comment on proposals and projects that required environmental impact statements. Since that time public participation has evolved into a multi-disciplinary field that combines the traditional fields of public relations and communication, organizational development and psychology, environmental and public policy, and alternative dispute resolution. In fact, most public involvement practitioners come from one of these disciplines.

One could argue that public involvement has grown over the years as a result of the publics' interest in becoming more involved. In some ways the field has followed the ladder of citizen participation first described by Arnstein (1969)(2) and later adapted by Chess et.al. (1988)(3). Arnstein's ladder shows a progression of involvement from the bottom up:

- Delegation of Decisions
- Negotiation and Consensus
- Meaningful Consultation and Advice
- Pro Forma Consultation
- Informing the Public of Actions
- Acting Alone

In the "old days" industries and agencies simply did what they wanted when they wanted. For years that behavior was accepted by the general public and we stayed on the first rung of the ladder. NEPA was an attempt to change this pattern. With NEPA came public hearings and public comment periods and, for a time, people accepted this new rung on the ladder. Many had high hopes that their concerns would finally be considered.

Attempts to involve citizens through public hearings, presentations and public comment still work in some areas, but have dwindled over time due to growing dissatisfaction with this modest level of citizen input. Why? In part, it is because the public process created expectations on the part of stakeholders that their comments would either be integrated into the project or would actually cause the project to be scrapped. When neither occurred, stakeholders became disillusioned. These processes did a pretty good job of informing citizens of pending actions or policies, when they were done objectively, but did a poor job of actually involving citizens in important decisions. In most cases, the important decisions had already been made. As this became apparent to stakeholders, the bloom came off the rose.

Citizen dissatisfaction with public comment processes led to pressure on some industries and government agencies to take the next step. That pressure came in part from law suits to block or delay projects and in part from recognized dissatisfaction with the public comment processes themselves. Industries and agencies that have taken the next step up the ladder have done so voluntary since, in most cases, there is no regulatory obligation to go beyond the public comment process.

The next steps on the ladder involve some form of direct dialogue or conversation with stakeholders. Citizen advisory boards, policy dialogue committees, regulatory negotiations and other stakeholder forums place industry and agency officials in situations where they must work cooperatively with stakeholders to accomplish their goals. If they cannot work together, the process will likely fail. Expectations among stakeholders are very high and there is significant pressure to succeed. If they cannot work together, the process will likely fail. Agency and industry officials may feel "burned" by unreasonable stakeholders. Stakeholders may feel that industry and agency people were never serious about sharing decision-making power. Both sides may feel that they were "suckered" into a decision-making process that was bound to fail.

THE CASE FOR STAKEHOLDER TRAINING

To help avoid this potential pitfall, public involvement specialists have turned to facilitators and mediators. Facilitators and mediators, while trained in slightly different ways, help guide the decision-making process without interfering with or influencing substantive decisions. To put it another way, facilitators and mediators act as third-party neutrals and help stakeholders negotiate decisions that satisfy mutual interests (for our purposes, negotiation is defined as the process of resolving differences, solving problems and reaching agreement). Most facilitators and mediators receive some training in negotiation principles and skills and work within the group decision-making process to get stakeholders to negotiate with each other instead of arguing and debating.

Facilitators and mediators are extremely beneficial in helping parties negotiate. However, facilitators and mediators are limited in what they can achieve when stakeholders insist on arguing and debating. The process is at best inefficient and more likely is like "herding cats". Even when stakeholders are civil, they may be disposed towards strong positions that lead to win/lose outcomes, stalemates and compromises. When agreements are reached through argument and debate they seldom maximize joint gains.

When we are locked in a debate we are not trying to solve a problem, we are trying to win an argument. It is only when we stop trying to convince

the other party of our position and begin to take their interests, and our own, and structure an agreement that all parties can live with that we are finally solving a problem.

The distinction between debating and negotiating is important in public involvement because of the diversity of legitimate public interests at stake and the need to integrate a broad spectrum of public interests into a decision. Moreover, it is almost impossible to win a debate if winning means convincing the other party of the correctness of our position. The chances of doing that are almost none. Yet we persist.

Public involvement processes that create meaningful opportunities for stakeholders to influence decisions is only part of the solution. The other is the development of collaborative problem solving skills through training in negotiation and conflict resolution. The two work hand-in-hand: The public involvement process provides opportunities to negotiate. Negotiation skills better enable stakeholders to communicate in ways that resolve differences, build agreement and maintain positive working relationships.

BASIC NEGOTIATION AND CONFLICT RESOLUTION PRINCIPLES

Negotiation and conflict resolution skills are not the miracle cure for the ills of public decision-making. They are, however, a step in the right direction. As a science, negotiation has existed for a relatively short time. While researchers and practitioners have written about collective bargaining for many years, modern teachings on negotiation date back to the 1970's. The most popular and still most widely accepted publication on negotiation is the book Getting to Yes by Roger Fisher and William Ury of the Program on Negotiation at Harvard University (4). Fisher was one of the pioneers of the modern field of interest-based negotiation. Interested-based negotiation is the process of focusing on underlying interests as the basis for exploring options that lead to agreement. Interest-based negotiating is described in more detail below. According to Fisher and Ury (1981)(4), there are five main principles behind the concept of interest-based negotiation. They are:

- Separate Human Issues from Substantive Issues
- Focus on Underlying Interests Instead of Hard-line Positions
- Search for Objective Criteria and Fair Standards
- Determine Your Best Alternative To a Negotiated Agreement
- Invent Options for Mutual Gain
- Look for Solutions that Satisfy Multiple Interests

Separate Human Issues from Substantive Issues

Fisher and Ury recognize that parties attempting to solve substantive problems often get caught up in conflicts dealing with emotions, beliefs, values, perceptions, trust, power and relationships. These "human" issues often prevent the parties from solving the substantive problem. The conventional approach to dealing with human issues is to subtly negate them or discount their importance. Instead of directly addressing the fears of neighbors over the cleanup of a local site, for example, the Agency will say, in so many words, "Just be rational. Trust us. We'll take care of it."

Fisher and Ury advise us to separate human issues from the substantive issues and to address the human issues first. In fact, unless the human issues are addressed to the satisfaction of stakeholders, they simply won't go away. Human issues that keep surfacing in environmental decision-making arenas are usually those dealing with emotions, values, trust, perceptions and past relationships. Such issues emerge first as

emotions and need to be dealt with on that level. The four primary emotions that people experience are fear, anger, joy and sadness. Psychologists believe that our other emotional experiences are combinations of or variations on the four primary emotions. In environmental debates two of the four emotions usually emerge, fear and anger, but are expressed as only one of the four, anger.

How do we separate human issues from substantive issues? To do this we must deal directly with the emotions that are preventing substantive problem solving. First, negotiators need to be aware that emotions are present. This is somewhat difficult because most of us are "programmed" to deny or down-play emotions, especially in professional situations. Second, negotiators must recognize the emotion by stating its existence to the party in a non-threatening way. Simply saying "You seem very angry about this..." or "I can tell you're very upset..." lets the party know that the negotiator is aware of the emotion without judging it or them. This acknowledgment will allow the party to let go of some of the tension.

The third step requires "actively" listening to the party's concerns. Active listening requires the listener to carefully paraphrase the speakers words until the speaker feels that the listener understands his or her concerns. At the end of the process of active listening the speaker should be able to say, simply, "Yes, those are my concerns." At the point of understanding much of the emotion will have drained from the dialogue. The speaker will feel that someone is listening, perhaps for the first time. Note that the listener has not attacked the speaker's position nor has he/she agreed with it. It is simply understood.

Focus on Underlying Interests Instead of Hard-line Positions

The second step in negotiation is to focus on identifying and satisfying interests instead of positions. Your position is one way, your chosen way, of satisfying your interests. Your interests are your underlying needs. Why do we focus on interests instead of positions? Because it is easier to satisfy underlying interests than to reconcile divergent positions. If we negotiate over positions, our negotiations become a test of will and power. If we negotiate over interests, we are more likely to find integrative solutions to our problems.

Let's use an example from CCEM's TRIP (Technology/Regulatory Integration Project) Implementation Manual to illustrate. Two opposing positions regarding the cleanup of a contaminated site might be to 1) clean the site to pristine conditions and 2) clean only the hotspots and seal the rest of the site. A stakeholder debate over these positions would be very difficult to resolve. We can do one or the other but not both. How might focusing on interests be any easier? Looking at the underlying interests that support these positions, we might find that stakeholders taking the first position care mainly about 1) reducing health risks to zero or almost zero, 2) restoring the ecological integrity of the site, and 3) holding the agency accountable for its past actions. Stakeholders taking the second position care mainly about 1) reducing costs, 2) lowering health risks to acceptable standards, and 3) avoiding further disturbance to the site.

Negotiators now have an initial list of interests that they can address. Negotiators could discover additional interests by probing more deeply into the reasons behind the interests stated. For example, why is avoiding further disturbance to the siteimportant? Perhaps to preserve existing wildlife habitat. Or, why is reducing health risks to zero or

almost zero important? Perhaps to assure stakeholders of minimum health risk, or to protect vulnerable populations that live near-by.
Negotiators could then develop what we call a "Joint Problem Statement" for this particular issue:

How can we... reduce health risks to zero or near zero
protect vulnerable populations
restore the ecological integrity of the site
hold the agency accountable for its past actions
while also... reducing costs
lowering health risks to acceptable standards
avoiding further disturbance to the site
preserving existing wildlife habitat?

The illustration leaves us with a set of interests - some common, some opposing and some simply different - that we must try to satisfy by negotiating. Will it be easy? No. But it will be a lot easier than knocking heads over opposing positions.

Search for Objective Criteria and Fair Standards for Resolving Differences

Objective criteria and fair standards are used in negotiations to establish impartial and unbiased methods for avoiding and settling data conflicts and generating information that will be used to make decisions. Objective criteria and fair standards are also used in negotiations to determine the value of an item independent of the resources that might be available to acquire it.

There are hundreds of examples of how objective criteria and fair standards are applied to negotiation situations. Salaries are based on comparable pay in the field. Home prices are based on comparable sales, replacement costs or rental income. Used car prices are based on the "Blue Book." Risk assessments are based on a standard model. Objective criteria are usually based on some quantitative assessment or accepted practice that is widely recognized by the public or a particular profession. Fair standards, on the other hand, may be quantitative, but are more likely based on what reasonable people believe is fair. When decisions are made without being "anchored" to some impartial and recognized objective criteria or fair standard, those decisions seem arbitrary or biased to outsiders.

Environmental decisions frequently run into problems with objective criteria and fair standards. We refer to these problems as data conflicts. For example, many complex public decisions involve planning and engineering studies and environmental impact assessments. These studies, designed and carried out by one of the party's experts, generate data that are then analyzed, interpreted for the public and used to make or justify a decision. The supporters of the project believe the studies. The opponents of the project do not. Did the studies use objective criteria and fair standards that were acceptable to all stakeholders? Probably not. They may have used scientifically defensible methods to conduct the studies, but if other affected parties were not part of the study design team, the studies may be perceived as biased. What usually happens in a data conflict is that both sides hire experts that support their positions and then become even more locked into those positions. Negotiators need to avoid data conflict traps by insisting on and finding objective criteria and fair standards that are supported by all stakeholders before a study is conducted.

Understand Your Best Alternative To a Negotiated Agreement

The principle of Best Alternative to a Negotiated Agreement (BATNA) was developed by Fisher and Ury to better explain the power and decision-making dynamics in a negotiation. Literally your BATNA is what you will do if you don't reach agreement in a negotiation. Or, put another way, what will occur if you can't reach agreement. If, as a negotiator, your BATNA is poor, if it is worse than a proposed negotiated solution and you can't improve it, it is wise to accept the negotiated solution. If your BATNA is good, if it is better than the best negotiated solution you can obtain, it is wise to walk away and pursue your BATNA.

Your BATNA must be a real alternative or it must be an alternative that you can reasonably count on. If your best alternative to getting the raise you want is to take another job, you are much better off having a firm job and salary offer from another employer. If the negotiation over the raise doesn't produce the results you desire, you will have some place to go. And, you will be able to compare your current employer's counter-offer with your new offer.

BATNA's can also be used to explain why certain public interest groups participate in collaborative decision-making processes while others decline. Often parties that won't participate believe their alternatives (BATNAs) are better than any agreement they could reach by negotiating. This is partly because many believe that negotiating means compromising and they are not willing to compromise. Many environmental organizations are reluctant to enter negotiations and feel that court action, or the threat of court action, is a strong BATNA. Fisher and Ury believe we should enter negotiations with a clear understanding of our BATNA's and walk away only when our BATNA is clearly better than the proposed agreement. If we want stakeholders to participate, we need to make them aware that a negotiated solution can be better than their BATNA. Or we can help them understand that their BATNA may not be that great.

Look for Solutions that Satisfy Multiple Interests

Interest-based negotiation requires stakeholders to seek solutions that are good for all. In order to do this, parties must have opportunities to get together for the explicit purpose of solving a joint problem. They must identify common and divergent interests behind their respective positions and make proposals that satisfy those interests. To aid this process, facilitators and mediators often ask stakeholders to suggest solutions that they believe everyone can support. Fisher and Ury call this Inventing Options for Mutual Gain.

Inventing options for mutual gain requires parties to be creative. However, creativity is risky. There is a tendency in any creative endeavor to immediately judge ideas that are new or unconventional. Judgment, however, has the effect of stifling most creativity. After all, how many of us want to look bad for suggesting a dumb idea? So we suggest ideas that we know will be accepted by the group and we end up with status quo solutions or no solution at all.

To better assure creative problem solving we must separate the inventing of options (brainstorming) from judging and selecting among them. Negotiators need to establish a simple ground rule: No judging or evaluating options until the inventing is complete. A second ground rule is often necessary when dealing with public officials and others who may worry about being attributed with an idea expressed during a brainstorming session: No attributing any ideas to specific individuals. Such an individual may have a creative solution in mind that integrates many interests but may be reluctant to express the idea for fear of

having it taken the wrong way. These rules protect all parties from outside attacks while providing the freedom necessary to be creative. Smart negotiators offer proposed solutions that they believe satisfy mutual interests. Proposals are then improved by all negotiators. Negotiated decisions, therefore, tend to maximize joint gains or, to put it in perspective, maximize public benefits. Debaters, on the other hand, minimize and discredit the arguments and positions of their "opponents" and sometimes even attack them personally. When they win, they seldom integrate the legitimate public interests of others. We tend to get one-sided victories that fail to maximize public benefit. While one-sided victories are sometimes the only way to solve problems, in most cases there are realistic ways to maximize public benefits that are lost because we don't know how to negotiate.

ADDITIONAL TRAINING CONCEPTS

In addition to the basic negotiation and conflict resolution principles and skills discussed above, an effective training program would address the following:

- Building trust
- Conflict resolution styles
- Improving listening skills
- Creating incentives to negotiate
- The role of power
- Dealing with difficult people
- The Solomon trap
- Escalation traps
- What stakeholders really want
- Responding to conflict
- Raising conflict
- Analyzing conflict
- Group dynamics

TRAINING METHODOLOGY

Training workshops are usually between one and three days in length and are most effective when done with 18 to 24 participants. The workshops combine the theory of interest-based negotiation and conflict resolution with principles and skills that participants can immediately apply in their work. Skills are developed and practiced through interactive negotiation exercises where participants play specific roles in a dispute or problem-solving situation. Participants are given a set of instructions that contain a common description of the problem and a context for the dispute. They are also given confidential roles that they must play throughout the negotiation. The roles create a tension between typical positional and confrontational strategies and interest-based negotiation skills. Like learning to ski, participants develop and practice the skills under relatively easy conditions (on a gentle slope) and then must maintain the skills as the conditions become more complex and contentious (the slope gets steeper and steeper).

Participants typically begin with positional and confrontational strategies, and only respond when others start to probe for underlying interests or make sincere attempts to understand other perspectives or seek joint gains. The dilemma negotiators face is whether to be the first to use the interest-based approach. If no one risks going first, no one will follow.

Following the training exercises, participants are debriefed about both the substantive outcome of their negotiation and the negotiation process

itself. The debriefing provides rich opportunities for participants to talk about negotiation and conflict resolution strategies and skills, group dynamics, power relationships and other issues. Often, the discussions shed new light on communication patterns that produce conflict.

INTEGRATING TRAINING INTO THE PUBLIC INVOLVEMENT PROCESS

Training can be integrated into the public involvement process at any time but is perhaps best at the early stages of development. Early training will help stakeholders address and resolve difficult issues before they become contentious. Once conflict appears, it is usually more difficult to get stakeholders to focus on alternative techniques. Ideally, training should be part of the initial orientation for the project, however, it could be brought in later, once group members feel more comfortable with the public involvement process, with each other and with the convening agency. Finally, training could be introduced at a point when the stakeholder group reaches an impasse over an important issue or decision. It is less likely that training will be accepted at this point, but it is still worth a try.

There are three factors that may serve as barriers to training and must be considered when determining how to integrate training into the public involvement process. First, negotiation and conflict resolution training may be perceived by some stakeholders as a method of controlling or influencing the outcome of a decision, or inducing stakeholders to compromise their hard-fought positions. Stakeholders who value their confrontational styles or who do not trust the decision-making body may resist such training. When this barrier occurs, it may be better not to push the training too early in the process.

Second, some stakeholders may question whether the training is necessary. Stakeholders who believe they are good negotiators may feel that they don't need additional skills. This is often the case with senior level professionals. Our experience is quite the opposite. Supposed "good" negotiators are often those individuals who are the most competitive in their styles and who are capable of making and defending strong arguments. They are usually not very good at collaborating to achieve joint gains. In addition, agency representatives who serve as stakeholders often believe that the training is intended for "all those unruly types" and not for them. This attitude fuels the notion that the training is designed to control the angry and unruly elements.

Finally, negotiation training may be seen by some as a way to "arm the enemy." "Why would we want to help the other side? " they would ask. "Let them fend for themselves." This attitude shows a clear misunderstanding of interest-based negotiation where the purpose is to help everyone do better.

Training in interest-based negotiation and conflict resolution was originally tested and is gaining in popularity in union/labor negotiations where their is a history of conflict over wages and benefits. The purpose of training in the union/labor arena is two-fold: to break down communication barriers that prevent negotiators from working together and to provide new skills for solving problems collaboratively.

CONCLUSION

Facilitators and mediators now fill the role of helping stakeholders negotiate tough decisions. They help parties separate human issues from substantive issues, focus on interests instead of positions, search for

objective criteria and fair standards, invent options for mutual gain and remind parties of the consequences of indecision. Facilitators, in short, help stakeholders move from arguing and debating issues to identifying and solving joint problems. But facilitators must struggle against the tendency of stakeholders to be argumentative and positional. Since we are not taught interest-based negotiation skills anyplace in our lives, few of us really know how to use them to solve complex problems.

Through TRIP, we have learned that stakeholders can become more effective participants when they are able to use interest-based negotiation skills. We recommend as part of any public involvement process that stakeholders be trained in these skills. CCEM recognizes that effective environmental decision-making requires opportunities and resources for stakeholders to work together and the skills necessary to solve problems once they are together. Most stakeholder processes provide opportunities and resources but fail to address skills. We tend to assume that people know how to solve problems collaboratively even though both history and current events tell us that just the opposite is true. Moreover, we would never consider providing someone with the opportunity and resources to swim, ski, drive a car or fly an airplane without also providing the skills. Why should solving important environmental problems be any different?

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EXPLAINING RADIOACTIVE WASTE TO YOUNGSTERS

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ABSTRACT

Following the strong public opposition after the announcement of the results of possible site selections for the final low and intermediate waste repository in Slovenia in 1993, the Agency for Radwaste Management has completely revised its public information strategy. The information and education programs, that should lead to greater public acceptance, are described. At present the bulletin, leaflets, booklet, permanent exhibition and lectures in the Nuclear Training Centre are applied. Special emphasis is given to information program oriented towards young generation, age 13-15 years.

INTRODUCTION

On May 27, 1993 the Agency for Radwaste Management of Slovenia has organized the press conference to present the results of the third phase of site selection for the low and intermediate level (L/ILW) surface radioactive waste repository. During the presentation five potential

sites in Slovenia were announced. Only few hours later reporters of the national television went to the village Kalievci, one of the potential repository sites, to get the first impression directly from the inhabitants. People from the village had expressed strong negative opinion by kidnapping the media crew. The crew was released only when the local inhabitants received a written statement of the representative of the government that within their community there will be no repository construction without their permission.

Reaction of the residents of Kalievci and also appearance of NIMBY (Not In My Back Yard) syndrome in such aggressive form has not surprised only the Agency for Radwaste Management but also the politicians and representatives of the media themselves. It was obvious that the public relations strategy of the Agency must be revised and renewed.

Radioactive waste in Slovenia originates from medical and industrial applications of radioactivity, from research and from the operation of the Krko plant, which is the main contributor. The construction of that Westinghouse plant took place in late seventies when the acceptance of nuclear power in Slovenia was not problematic. The general public was informed about the construction and about the economic benefits of the plant but there was no systematic public information and education program on the technology and its implications. Disposal of radioactive waste was not considered a problem and was not given any priority in comparison with completion and operation of the plant. Siting of the repository for low and medium level waste was postponed into the period of plant operation when it became obvious that the on-site storage capacity (about 11000 drums of low and intermediate level waste) will be reached in a foreseeable time. The siting process finally started without any previous education or information program and reached the phase when the potential sites were made public exactly in the period of fierce opposition to anything nuclear. The above described public reaction was understandable and at given time unavoidable. The whole process of site selection without the serious public information work was wrong and had to be changed. Therefore Agency for Radwaste Management under the new leadership has developed a new information strategy and started with its implementation. Nuclear Training Centre plays an important role in it.

PUBLIC RELATIONS STRATEGY

The first step in creating revised and renewed public relations strategy was the analysis of past experience. The analyses have shown that the communication activities in the past were insufficient. Since the dialogue was not established, the media and the public have little confidence in the information coming from the institutions working in the field of peaceful use of nuclear energy.

The Agency has also studied the strategies of foreign Agencies for radwaste management in overcoming the NIMBY phenomenon. On the basis of this analysis the communication strategy of the Agency for radwaste management was prepared. Three important aspects were included: communications, education and negotiations. In the first phase communication activities are used to attain a higher level of quality in addressing the issue; the communication activities are targeted at the general public. The educational activities are a logical and more focused sequence of the informing activities. For this reason the Agency is preparing various educational programs that consist of diverse activities. Negotiations are a necessary part of the communication strategy as the final decision should be made by the local community.

Communication activities concerning the issue of the waste encompass various tasks. First, there is the need to overcome the rejection of dialogue on this sensitive issue and second, to seek dialogue with the general public which is of key importance in any decision making process concerning radioactive waste.

Therefore the plan of PR (public relations) activities has the following basic objectives:

To establish a well qualified team for successful and highly professional work in the field of communication and crisis communication management;

To prepare different written materials that can be used for information and educational purposes;

To organize educational and information activities.

INFORMATION

At the present there is a great interest on nuclear issues in the public. The information on radioactive waste management comes from various sources. Often the information is distorted or misguided, many times resulting in negative atmosphere about radwaste management activities. In last years the Agency has given more attention to the public information through the media. Several articles on radioactive waste management have been prepared and published in most popular Slovenian newspapers. In one year thirteen articles, seven reports, four press releases, three interviews and two statements have been published. It should be mentioned that before 1994 there were no such contributions. Through these activities many contacts with journalists were made that will facilitate further communication activities.

Agency's Bulletin has been published recently. The first number of Bulletin gives answers to the most frequently asked questions on radwaste management. Present and future Agency's activities are also briefly described. Special care is dedicated to the clear and simple language so that no special technical background is necessary to understand the subject.

INFORMATION AND EDUCATIONAL ACTIVITIES AT NUCLEAR TRAINING CENTRE

Part of the information and education activities of the Agency for Radwaste Management has been taken over by the Nuclear Training Centre, which is part of the "Joef Stefan" Institute. Several activities related to the Radioactive Waste have been prepared. It has been decided to concentrate on youngsters as the main target group and to prepare an information program consisting of lectures, radwaste exhibition, visit to the interim radwaste storage, at-a-glance information leaflets, radwaste booklet and video tapes.

The information and education program consists of the following activities and information materials:

The Lecture Radioactivity and Radioactive Waste

The lecture informs the students about the following subjects:

1) Radioactivity and Radiation,
2) What Is Radioactive Waste?
3) Radioactive Waste in Slovenia.

The lecture set consists of 30 viewgraphs and lasts approximately 45 minutes. There are additional 10 slides available for more detailed explanation of typical questions. We always encourage the discussion. Typical visiting group at the Centre consists of about 50 students, but we can accommodate also up to 100 visitors. The lecture can be presented also elsewhere in the country. Since September 1995 there were 16 groups

attending the lecture comprising altogether 890 students with 57 teachers.

In addition to this particular lecture about the Radioactive Waste Management only, we are offering also another lecture about the Nuclear Power in general. Radwaste management is also covered in the narrower extent. This lecture was attended by approximately 6000 students in the year 1995, which represents roughly 10% of total population of that age (13-15 years).

Radwaste Exhibition

A set of 16 panels explaining radwaste is added to the existing permanent exhibition Electricity from nuclear energy. The panels explain the origin of low, intermediate and high level waste, technology of waste disposal and examples from different countries. There are also a normal and compacted radwaste barrels and a mockup of the surface repository displayed. The visit to the exhibition usually follows the lecture, therefore it represents repeated and supplement information. Yearly about 6000 visitors see the exhibition.

Visit to the Interim Low and Intermediate Level Waste Storage at the Reactor Centre

The Nuclear Training Centre is part of the Reactor Centre in Podgorica near capital Ljubljana, which includes an interim waste storage facility intended for the waste originating from research, medical and industrial use of radioactivity in Slovenia outside the Nuclear Power Plant. During the visit the storage is shown only from the outside. On special request of smaller groups it is possible to see also the interior.

The lecture, the visit to the exhibition and the visit to the storage facility are typically merged during a single visit of student groups to our training centre. Every year students from around 100 schools come to the Nuclear Training Centre.

At-a-glance Information Leaflets

In the Nuclear Training Centre a set of four at-a-glance information leaflets has been prepared for the Agency for Radwaste Management describing the following topics:

1) Radioactivity and Radiation
2) Radioactive Waste
3) Disposal of Low Level and Intermediate Level Radioactive Waste
4) Disposal of High Level Radioactive Waste

The fifth leaflet, Radwaste Storage in Slovenia, is in preparation now. The purpose of the leaflets is to provide the reader with a lasting correct information for later reference. The leaflets have been designed in a style appealing to youngsters (see Fig. 1). They are widely distributed to anyone interested in the subject in Slovenia.

Fig. 1

Booklet about Radioactive Waste

The booklet is conceived as an information source to the readers who are willing to spend more time than just glancing through the leaflets. It describes the nuclear fuel cycle, radioactivity, sources of radioactive waste and radioactive waste disposal on about 40 pages. The booklet is in print now and will be distributed to anyone interested in the subject, primarily to the schools.

Video Tapes

Because Slovenia is rather small country it is, for the time being, too expensive to produce our own video tape material. Therefore we have

translated three video tapes about the Swedish radioactive waste management system to Slovenian language. In the future we intend to bring closer to our public some similar material from other countries and to produce some own material. The French video is being translated now.

Journalist's Guide to Radwaste

Speaking about technical matters is often confusing or hardly understandable for non-technical people. Therefore we are preparing the version of the booklet about Radwaste which is written by a journalist. We expect this to be more understandable for the wider public.

FAQ about Radwaste

Another material, that is in preparation, is a set of Frequently Asked Questions about Radwaste. Most of them are being asked by our visitors. We hope, that this will provide fast and easy answer to interested public.

Future Plans

The final objective of all public information activities is to obtain the acceptance of wider and local public for the radioactive waste repository. Keeping that in mind we are trying to produce material that is easily understandable and can be easily distributed. All available technical means should be used, especially newly emerging ones. So in the near future we intend to prepare information about radioactive waste for the World Wide Web, a part of the Internet system. An extensive material about nuclear energy in general has already been prepared and is available on http://www2.ijs.si/~icjt/kakoje/ (in Slovenian Language). Another media, very popular especially among the young generation, is a hypertext computer presentation on a CD ROM. We intend to publish some material about the Radwaste in a user friendly computerized way.

PUBLIC OPINION

Influence of our activities is hard to measure exactly. Immediate response during and after the visit is positive. The opinion of the general public is measured every year by a wider polling performed by the Faculty of Social Sciences in Slovenia. Some positive shift in the attitude towards nuclear energy in general can be observed from their answers. It is shown on Fig. 2, that number of those in favor of further operation of NPP is increasing.

Fig. 2

CONCLUSIONS

The problem of acceptance of radioactive waste disposal facility by the general public is becoming one of the key problems of nuclear energy everywhere in the democratic world. Recent examples in different western countries (Switzerland, USA) are not encouraging. It is also obvious there is no perfect general solution, that would work everywhere in the world. In Slovenia we are carefully studying other experiences and trying to find our own way to solve the problem. Next several years special care will be devoted primarily to the information and education of the people in order to make them understand the facts about radwaste. Only after a certain level of public confidence will be achieved, the further steps towards the final disposal will be taken.

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AN INDEPENDENT TECHNICAL REVIEW OF THE PROPOSED NEW DOUBLE-SHELL TANKS AT HANFORD

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ABSTRACT

This paper describes the approach used by an independent Technical Team created to advise the Dollars and Sense Committee of the Hanford Advisory Board regarding the need for new Double-Shell Tanks at Hanford, and presents the Team's conclusions and recommendations. The process used led to rapid acceptance of a dramatic re-direction of the new tank proposal by internal and external parties.

The Technical Team produced a safety-driven analysis based on a "worst case" budget scenario to ensure that minimum required facilities were available for waste management and operated with a manageable level of risk in the 200 West Area and 200 East Area tank farms, and that major activities of waste processing (e.g., evaporation, storage and waste volume reduction) will have the storage needed for the next several years. The risks of the various feasible options for waste management were balanced against cost constraints and the necessity to move forward with the remediation mission. These considerations guided the Team in developing its recommendations to the Hanford Advisory Board regarding the project. By design, the Team investigated only technical issues, but recognized that the Board had the ability and the obligation to combine technical insights with other important and valid considerations and values as it developed its own recommendations to DOE regarding the proposed new tanks and other features of the Tank Waste Remediation System.

The Technical Team focused its recommendations on waste storage needs from 1995-2004, the time period covered in the Draft EIS on Safe Interim Storage and the time period of current importance to the new TWRS program. After 2004, the waste treatment division at Hanford is scheduled to begin to retrieve and treat solids from Single-Shell Tanks; when this treatment cycle begins, the need for large volume, long-term storage capability will be reduced as treatment proceeds. Evaluation of treatment processes and systems and the need for (operational, short-term) storage specifically required by these treatment processes were far beyond the Team's assignment. However, if the start of treatment is delayed beyond 2004, then retrieval of SST solids must also be delayed--and this would have significant implications for the long term storage capability needed at that time. It is important to keep firmly in mind that tank needs after 2004 are determined by treatment facility configuration, retrieval and treatment rates.

The Team was limited by too few risk assessments as well as the level and frequency of change of the data provided to it regarding waste volume projections and tank waste classification. This was understandable; the situation during the Team's review period was a fluid one, indeed. Despite these limitations, the Team arrived at seven recommendations, focusing on the three areas of primary concern: 1) tanks, 2) the Cross Site Transfer System (CSTS), and 3) risk analyses. The Hanford Advisory Board unanimously adopted these recommendations, which were complex and technically sophisticated, for formal transmittal to DOE. The process

used by the Team ensured that the recommendations (and the reasoning were understood by the Board, other stakeholders, DOE and its contractor. The full paper presents the technical substance of the analysis and recommendations, as well as additional information on how the Team worked with stakeholders both without and within the DOE system.

INTRODUCTION

The Assignment

In late September 1994, Dr. Thomas Engel from the University of Washington asked the author on behalf of the Dollars and Sense Committee (DSC) of the Hanford Advisory Board (HAB) to organize a Technical Team to review issues, assumptions and alternatives surrounding the Multi-Function Waste Tank Facility (MWTF) being proposed by the Tank Waste Remediation System (TWRS) at Hanford. The original goal of the Technical Team review was to "enable DSC and HAB to understand and examine the assumptions and key policy decisions behind the decision to proceed with a project to construct six new Double-Shell Tanks, and understand the costs and alternatives so that the Board may offer timely advice and principles/values regarding policy choices." The Technical Team, led by Dr. Paulson, then Research Professor at the Illinois Institute of Technology, and assembled under the auspices of the firm of Paulson and Cooper, Inc., included Professor Frank Parker of Vanderbilt University and Dr. Michael Kavanaugh, a Principal of ENVIRON. This Team was able to address issues of policy, history, engineering (including design integrity, waste volume inventory and projections, and facilities required), alternatives, cost, and risk. In addition to the technical experts, the project was supported by Ms. Susan Sink, a technical writer and researcher, who reviewed and summarized documents and helped prepare drafts of this final report to speed the Technical Team members' review process. Logistical and research support in the Tri-Cities was provided by Ms. Jennifer Plemmons of Pasco, WA.

Re-scoping of Assignment

The details of the review assignment soon changed dramatically, as a result of the "Double-Shell Tank Inventory and Available Space" report of December 28, 1994, and the February 1995 recommendation from Westinghouse Hanford to DOE's Richland office is that no new tanks be built. However, the thrust behind the basic DSC/HAB request to "examine the assumptions and key policy decisions behind the [project] decision" was still valid. The Technical Team therefore examined in detail the assumptions and key policy decisions that led to the new recommendation (not to build tanks). Any decision on whether or when to build additional tanks is impacted by many considerations, including the risks, benefits, costs, uncertainties and impacts in future years of a decision not to build tanks soon. An evaluation of all these impacts was beyond the scope of the review. It was, however, possible to examine the potential consequences associated with various courses of actions until the Single-Shell and watch list tanks (not mutually exclusive) are stabilized. The review was limited in time, scope, and dollars, and could thus not be guided by long-term, integrated technical efficiency, but it nevertheless provided for a "No Regrets" outcome which meets near-term and mid-term safety needs and does not preclude additional action related to storage and treatment in the years beyond 2004. Just as important, the Team's review provided what the Hanford Advisory Board needed to know when it needed to know it--to provide focused, results-oriented to advice to DOE. In the present uncertain climate, while one must be sure that actions taken now do not

foreclose future options, strategic planning within limited time horizons may be the wisest course, both for DOE and its stakeholders, especially its formal site-specific advisory boards.

A Word on the Process

Extensive documents were obtained from DOE and its contractors, all of whom were extremely cooperative. The author, who led the Team, participated in selected meetings of the DSC and the HAB during this research period. With the help of DOE, the DSC/HAB, and its own contacts, the Team made aggressive efforts to reach other stakeholder groups not represented on the HAB, internal and external critics of the new tank proposal, retired "old timers" who had worked at the tank farms years and even decades ago, and others who might be able to contribute technical or other relevant information. While informal, this process was thorough.

In February, 1995, after completing its review of all relevant documents, the full Team spent several days at Hanford, holding a series of meetings with those identified above. The Team's visit was publicized by the Board and DOE; in addition, the Team established a local contact office for two months before the visit. These meetings could be classed as informal public hearings, in a sense; members of the HAB participated as observers much of the time, and appropriate DOE, Westinghouse, federal and state agency figures were also present. Several private citizens ("old timers") joined the Team's meetings as well, some of whom had traveled considerable distances at their own expense. The meetings, while informal, were vigorous in their discussions and debates. Before leaving the Tri-Cites, the Team met in private to develop its analysis, recommendations, and conclusions, and to begin drafting its report.

A draft of the Technical Team's report was provided in late February to several anonymous expert reviewers selected by the Team, and shortly after that a revised version was made generally available to interested individuals and agencies in the Pacific Northwest and at DOE headquarters. The DSC was fully briefed on the findings and recommendations in the revised draft report at its regular March meeting, and the final draft of the Executive Summary was provided to the full HAB in advance of its April meeting. Additional comments from agency and other reviewers were received in April, and the final report of the Team was submitted to the HAB in advance of its regularly scheduled May meeting, where a full discussion of it was held.

Project Areas Addressed by the Report

The Team evaluated in depth both the initial proposal in the August 1994 "Safe Interim Storage Draft EIS" and the revised February, 1995 proposal. The Team early on decided that the decision to build tanks or not build tanks cannot be considered in isolation from the other elements of the initial proposal (particularly the CSTS). The Team also assessed possible impacts of the new proposal for no tanks on the safety of tank farm operations in the medium term (until the start of pre-treatment and treatment, scheduled for 2004). The Technical Team considered the issues of projected increase/decrease in waste volumes, relationship of new tank construction to operational plans at Hanford, cost, as well as material selection, design, and other features of the proposed tanks.

Background Assumptions

The Technical Team based its recommendations on several key assumptions associated with the original proposal (six tanks) and the changed proposal (no tanks). The key were:

1) technical issues associated with the new proposed action regarding tanks (no new tanks)
2) tank waste storage needs until pre-treatment begins
3) risks and probable consequences of new proposed action
4) relationship between current waste volume projections and the new proposed action
5) relationship between compatibility of waste types and the new proposed action
6) availability of contingency plans and other waste management issues

It was the aim of the Technical Team to determine if the new proposed action was sound and whether or not new tanks are needed. Further, the Team identified what types (capacity and function) of tanks might be needed, resulting from an assessment of the consequences of the new proposal given current information on waste type, integrity of both existing tanks and the CSTS, and waste volume projections.

DESCRIPTION OF INITIAL APPROACH

The Safe Interim Storage Draft Environmental Impact Statement (DEIS) considering new tanks at Hanford and completed in August 1994 assessed the three major components: 1) six new double-shell tanks (DSTs), 2) the Cross-Site Transfer System (CSTS), and 3) the Initial Tank Retrieval System.

A full description of the proposed new tanks facility can be found in the DEIS and was the official preferred alternative by the Department of Ecology and USDOE Richland Operations Office through December 1994. The rationale behind this recommendation focused on the need for additional storage space to passively mitigate tanks 101-SY and 103-SY through dilution.

SUBSEQUENT APPROACH

Tanks

Based on the revisions to the Waste Volume Projections, the decision to actively mitigate tank 101-SY (and if necessary tank 103-SY) with mixer pumps, and driven by the then-current budget situation, TWRS in February 1995 recommended that no new DSTs be built. The new recommendation reflected a shift in the Waste Volume Projections, and clearly included assuming a higher level of risk and uncertainty than was associated with the original plans for 200 West Area. This risk had not been quantitatively assessed at the time, although one report offered a "high/medium/low" scale for evaluating risk of certain components of the new projections. The possible consequences of failure of any of the assumptions included in the Waste Volume Projection January 1995 revisions are assessed in section V of this paper.

The recommendation not to construct tanks was based on the following major assumptions connected with Waste Volume Projection revisions and associated Path Forward paper:

- Continued and maximized use of the East Area Evaporator
- Significantly reduced future waste generation from site generators
- Revised estimates of the "porosity" of incoming waste (increasing volume which will be treated by the Evaporator)
- Timely construction of the new CSTS
- Minimal waste produced by the planned sludge washing demonstration
- Active mitigation through mixer pumps for 101-SY and 103-SY (if needed)
- Consolidation of compatible wastes now found in several tanks into fewer tanks

With revised waste volume projections based on these assumptions, the waste volume would exceed the projected capacity in 1999 (by 0.22 million gallons), and by 1.22 million gallons by the year 2001. The proposed solution for storage of excess waste was to use evaporator support tanks for spare space and store additional waste in 102-SY, the current staging tank for all waste which must be moved from 200 West to 200 East. The Path Forward document of February 1995 predicted 0.5 million gallons will be stored in 102-SY as other space becomes unavailable. However, only TRU-compatible waste can currently be stored in this tank. The proposal is also dependent on the operability of the current and new CSTS to transfer wastes in 102-SY, and the successful resolution of TRU-compatibility issues for 102-SY.

Cross-Site Transfer System (CSTS)

The new proposed action strongly recommended, in fact requires, the construction of a new Cross-Site Transfer System. The volume projections which justify not building any new tanks require the use of the existing CSTS through 1998. The January 1995 proposal recommended construction of the new CSTS as developed originally and described in the DEIS, without considering modifications based on the fact that it will no longer serve the six originally proposed new tanks. The CSTS proposed consists of 2 lines routed over approximately 6.5 miles, parallel to the existing system except for the stretch just before it enters the East Area. The routing had apparently not been re-evaluated following the decision not to build new DSTs.

Initial Tank Retrieval System (ITRS)

The ITRS was no longer needed for tank 101-SY and 103-SY, since passive mitigation by dilution has been dropped as a solution to the hydrogen gas build-up situation. Thus, the ITRS no longer fell within the scope of the Team's assessment of the project.

NEW CONSIDERATIONS RAISED BY JANUARY, 1995 CHANGE IN APPROACH

The new approach, to drop plans to construct tanks in either the 200 East area or 200 West Area, was based on a set of major assumptions which may or may not hold true. The decision not to build tanks is based on all the assumptions holding true. In the view of the Technical Team, it was not reasonable to base such a major decision on a conclusion that all assumptions will hold true; a fuller understanding of risks involved is essential for sound decision-making. Key issues which needed to be considered as a result of the new proposed option include: risk analyses; accuracy and consequences of errors in waste volume projections; the interim approach to addressing leaking SSTs in West Area which may result due to incompatibility or failure of the CSTS; and the impact of the decision to actively mitigate 101-SY and 103-SY.

Critique of Risk Assessment in the "Path Forward" Decision Paper

In the Path Forward decision paper, only the risks inherent to the new path chosen were presented. Even this risk evaluation had problems for independent validation in that only safety, financial and programmatic risks were evaluated and then only in terms of "High/Medium/Low," with no definition of the numerical meaning of the terms. Even the "comments" were plagued by vague terms instead of quantifiable terms, and there were some important contradictions in the reports, raising a question as to whether predictions for cross-site transfer capability were reasonable, the failure of which would result in a serious waste storage shortfall. With no probabilities of the events given, with no estimates of the consequences given, and without the same information for plausible

alternatives and the costs for the alternatives (and with no sensitivity analysis), it was impossible to judge which alternative is best or even if any is satisfactory. The recommendation to proceed without any new tanks is a case in point. According to the data given to the Team in February 1995, in the year 1999, even with positive outcomes on all the waste reduction assumptions, there would be a shortfall of tank space requiring use of the Evaporator support tanks (with its impact on Evaporator operations), and in the year 2000, the use of space in tank 241-102-SY will be required to meet the shortfall, though preliminary indications already had cast doubt on the viability of this plan unless special actions are taken regarding 102-SY.

Additionally, the viability of the existing pipelines between 200W and 200E is required until the new CSTS lines are in place, projected at that time to be no earlier than 1998. There were no relative probabilities given for each of the actions required to make this plan succeed. In the Team's view, this was not a technically sound way to make decisions on complex matters of such importance.

All risk analyses start with incomplete information, but with sensitivity analysis one is able to determine what information is most critical and obtain that. There is certainly more risk information available than what was presented. For example, when we probed, we learned that a formal risk analysis had calculated a 70% likelihood of failure of the first mixer pump during its lifetime. Similarly, in a rough fashion one could calculate the likelihood of the remaining two cross-site transfer lines staying in operation until a replacement system is in place; we calculated this to be only 40%.

Issues Related to Available Tank Storage Capacity and the Waste Volume Projections

The five key issues regarding available tank storage at Hanford were, in the Technical Team's opinion, 1) waste volume projections; 2) waste compatibility; 3) areas of production; 4) waste transfer capabilities; and 5) evaporator effectiveness. These issues are explored in depth in this order.

The Technical Team was provided significant information that established the estimated waste volume projections from 1994 to 2004. The most recent projection reviewed indicated that in the year 2000, the current capacity of the DSTs (given the estimated capacity for the 28 DSTs is 31.28 million gallons) will be exceeded; a shortfall of available storage space will occur. The Team intended to generate a similar summary for the total volume of waste combining the existing inventory with waste volume projections. Unfortunately, the categories in the February 1995 document were different than those that can be extracted from other relevant reports, and we were not able to carry out this task in a manner acceptable to the Team within the schedule we had to finish our review.

The accuracy of the revised waste volume projections are absolutely critical for a decision to not proceed with construction of any new DSTs. The Team did not believe that the assumptions used in the most recent volume projections were documented clearly enough. While minor inaccuracies in projected generation and treatment would not affect the final recommendations of the Team, the possible failure of the other major assumptions to contribute to waste volume reduction (compatibility, transfer, evaporation, etc.) should be revisited and evaluated as part of development of a contingency plan and associated risk analyses. Just as important is the need for DOE and Westinghouse to monitor waste volumes

actually produced month by month. If the projections are exceeded--that is, if more wastes are produced than are currently foreseen--timely re-adjustments to the entire waste storage system will be called for, perhaps on very short notice indeed.

The second key issue regarding the management of the liquid and salt waste in the Hanford Tank Farm is compatibility of waste. A major issue is the compatibility of waste in SSTs in 200 West Area with the staging tank 102-SY. The best estimate we received from a staff member at DOE was that as much as 1/3 of the current liquid waste in West Area (or roughly 1.2 million gallons) might be incompatible with the staging tank. If this proves to be an accurate estimate, it greatly affects the scenarios developed as part of our final report.

Additional wastes will continue to be produced from various facilities in 200 East Area and 200 West Area over the time period considered. Documents the Technical Team reviewed gave waste generation figures only after evaporation, figuring all the waste "generated" as slurry or "solid" waste stored in 200 East. These figures do not demonstrate how much waste will need to move through 200 West Area, particularly through 102-SY (which accepted only TRU-compatible waste) or provide estimates of generated waste which will need to be stored in 200 West Area if the old CSTS is not operational. Therefore, these projections are based on two assumptions which involve considerable risk. Rev. 20 gave a figure of 13.01 million gallons for waste generated through 2004 without evaporation. By the time the new CSTS was expected to go on-line, in 1998, 6.68 million gallons of waste will be generated. If even a small portion of this waste is generated in the 200 West Area and not transferred for evaporation and/or storage, there will be a serious shortage of tank space in 200 West Area.

It appeared to us that sufficient capacity exists in the DSTs in the 200 East Area (25 tanks are located there, and only 4 are on the Watchlist). There is sufficient capacity in the remaining 21 tanks, particularly if certain wastes are consolidated, to accommodate waste generated even in unevaporated form over the near term.

Current and future waste transfer capability, both within each area and especially between the 200 West area (where storage capacity is low) to the 200 East area (where it is greater) is a very significant issue for both now and later. The Team did not evaluate in-depth the plumbing and pumping details of how waste moves within the 200 West and 200 East Areas, or from facilities to the 200 Area. However, it was unclear to us whether sufficient flexibility exists to move wastes in emergency situations, or if the existing transport system from other areas of the Hanford Site to the Tank Farm are sufficiently reliable to ensure low risk of waste releases. The Team concentrated on the risks associated with assumptions regarding the operability of the CSTS, although it recognized the importance of other transfer capabilities. Apparently appropriate above-ground emergency transfer lines exist to accommodate emergencies (leaks) within each Tank Farm, but there is no equivalent contingency to move waste between 200 East and 200 West Areas. The reliability of the waste volume projections, therefore, are primarily dependent on the existence of a functioning CSTS to maintain evaporation projections.

The Evaporator in the 200 East Area is the only operating evaporator system in the entire Tank Farm. The old Evaporator in the 200 West Area has not been operational for more than 10 years, and it would not be

justified to renovate it to treat the limited amount of liquid waste inventory in 200 West Area. Clearly, the continued operation of the East Area Evaporator and its capacity for further reducing the total volume of waste is an essential component of both the current and the long-term waste management strategy at the Hanford Tank Farm. The reported reduction of waste through evaporation in calendar year 1994 was 5.2 million gallons. Both campaigns were slightly below the reduction rate projected. For example, DOE staff reported that campaign 94-1 was projected to reduce 2.9 million gallons by 2.5 million gallons (85%). Actually, it reduced the waste by 2.4 million gallons, resulting in an unanticipated excess of 100,000 gallons. If more recent waste volume projections predict higher rates of evaporation, as we believe is the case given a further reduction in specific gravity (even when the initially proposed predicted factor was not met), the actual performance may fall even farther behind the predictions, significantly affecting the accuracy of the waste volume projections. Again, the capability of the Evaporator to reduce waste inventory is affected by other factors, including the operability of the CSTS in the near and long term, waste compatibility, and compatibility of sludge and slurry with storage space in East Area that is available.

The Team reviewed with great care the then-current waste volume projections, a subject of great importance of course, but fraught with too many details to cover in this brief paper. In short, it was impossible to determine how much waste was to be produced by each of the site generators. And even aggressive use of the Evaporator cannot be expected to reduce the waste generated to 0.

A key assumption in the 1994 projections is that the DST inventory specified in 1994 would remain constant until the year 2004. This volume was based on the inventory as of October 1994, and has been reduced through operation of the Evaporator. It was unclear to the Team why this number should remain constant over the 10-year planning period. A similar point can be made regarding the concentrated waste. Does this mean, for example, that the concentrated waste volume remains constant from the year 1996 beyond? Presumably the volume of concentrated waste should increase (while supernate waste decreases) as liquid waste is treated at the Evaporator and the resultant waste is transferred for storage in East Area tanks.

In the planning horizon of 1995-2004, at least two different projections were completed. One predicted waste tank shortfall capacity will occur in the year 2000. This shortfall could be exacerbated by the incompatibility of wastes. This analysis suggests that the Hanford Tank Farm is operating at, or close to, its capacity. If no additional tank capacity is installed in either 200 West or 200 East Areas, the uncertainties regarding the waste projections, particularly with respect to waste generation in the West Area which must be transferred to the East Area for treatment, indicate a significant storage shortfall could occur even before 2000. Specifically, if the new wastes are not evaporated, the cumulative volume of these waste would amount to approximately 1.1 to 1.4 million gallons per year. Given the current space available for storage of waste in West Area of 300,000 gallons to 1 million gallons (if successful transfer of the contents of 102-SY could be made), and the ability to store only TRU-compatible waste in the available tank in West Area, even a small amount of waste generated and not transferred to the

Evaporator could cause a shortfall in the West Area in the very near term.

Emergency Interim Report to Leakers Not Addressed

In its final report, the Team stressed the need for contingency plans to provide for failure of interim stabilization efforts for the remaining tanks which are not yet stabilized, and presented a series of Hobson's choices; space does not allow this important subject to be presented in depth here. But all of these unpleasant alternatives pointed to the urgency of getting the proposed new CSTS built and operational and with adequate redundancy.

Impact of Decision Associated with Active Mitigation of 101-SY

The need for any new DSTs in 200 East Area was eliminated by the decision to actively mitigate 101-SY (and if necessary, 103-SY) using mixer pumps. The primary rationale for constructing new tanks in 200 East Area was to accommodate the additional waste produced by a dilution ratio of 3:1 for both 101-SY and 103-SY, which would require transfer from West Area to provide storage space before evaporation and for storage of "solid" waste following evaporation of the dilute waste. Since there has apparently never been any additional perceived need for new tanks in 200 East Area either to accommodate additional waste generation or Evaporator operations or to address incompatibility and Watchlist issues in the East Area, the decision to actively mitigate 101-SY and 103-SY is sufficient reason to eliminate construction of the proposed storage DSTs in East Area in the near term. This decision assumes some risk, particularly related to the continued success of the mixer pump. If the currently installed mixer pump fails, a back-up pump exists. It is in light of this contingency that the Team believed the associated risk for the near term is not great enough to require additional back-up in 200 East Area. However, if the mixer pump fails or fails to be a successful solution for mitigating hydrogen build-up in Tank 101-SY and/or 103-SY, the issue of needs for new tanks and alternative mitigation methods will need to be revisited immediately. In any case, the question of whether or not to build additional tanks in 200 East Area will need to be revisited for the time frame beyond 2004, to assess storage needs related to process and treatment operations scheduled to begin at that time.

The decision to actively mitigate 101-SY and 103-SY does not in itself eliminate the perceived need for new DSTs in 200 West Area. The Waste Volume Projections and associated assumptions regarding leaking SSTs in the West Area and operability of the CSTS need to be evaluated before conclusions can be made regarding the need for new tanks in the West Area.

SCENARIOS GENERATED BY THE TECHNICAL TEAM

Given the information and set of assumptions on which the reductions in waste volume were accomplished, in conjunction with the existent compatibility issues, the Technical Team produced a series of decision flow diagrams to assess possible risks and outcomes to inform conclusions regarding the need for tanks in West Area. The scenarios were based on the following basic assumptions: a) all SSTs in West Area are assumed either to be currently leaking or to begin leaking at a rate of 1-2 additional tanks per year, b) waste storage space must be made available for these leaking wastes to provide a RCRA compliant storage alternative, c) whenever a CSTS is available for transport of SST wastes (either the current system or the new system), a staging tank (or tanks) of sufficient size and compatibility to accommodate transfer to the

Evaporator in East Area must be available, and d) no adequate emergency contingency plan exists for managing wastes if they are unable to be stored in the only available tank space (102-SY, currently at 300,000 gallons). A complete discussion of each scenario can be found in our complete report, which is available for the cost of printing, handling and shipping from our office, PO Box 1541, Jackson Hole, WY 83001.

RECOMMENDATIONS

The Team's recommendations were meant to help the DSC/HAB focus DOE effort on high priority items which will lead to a more robust and lower cost system capable of addressing all safety issues with minimal construction costs over the term of the next decade (through the expected pre-treatment/treatment date of 2004). They anticipate the risks as outlined in the series of scenarios developed, and generally reflected the budget constraints being imposed on the program. The recommendations fell into three basic categories: A) decisions regarding tanks, B) the Cross-Site Transfer System, and C) emergency contingency plan and risk analyses to guide operations and address leaking tanks in the near-term

Tanks

1) Evaluate, as a top TWRS priority, the possibility of a 102-SY clean-out to render this tank fully useable for storage and transfer of all types of wastes, not just TRU-compatible wastes.

2) As a second priority, evaluate the option of building smaller transfer tank (or tanks) to move SST liquid waste if 102-SY cannot be cleaned out.

3) Accept the DOE view that no additional immediate action regarding tank construction be taken for the 200 East Area.

Cross-Site Transfer System

4) Pursue and complete the fastest possible review and decision-making for the proposed new CSTS. Without pre-judging the final, broad-based decision on this new system, from a purely technical perspective the Team believed its analysis strongly justifies accelerated construction of the new CSTS.

Contingency and Risk Analyses

5) Develop a contingency plan to address the possibility that the current CSTS will not be operable in the 1995-1998 time frame (or until the new CSTS is operable).

6) Prepare more quantitative risk analyses for each element of the system, and to make these analyses available to interested and affected parties.

7) Carefully evaluate the new waste volume projections, and just as important, monitor on a frequent on-going basis, whether the predictions of the most recent projections are borne out in the months and years ahead, so that timely re-adjustments can be made to the operation of the entire Hanford waste storage complex.

THE AFTERMATH

As of today, not only did the HAB accept all seven recommendations as presented, but several other positive things have happened. Testing of the existing CSTS showed that to the surprise of many, including the author, the remaining lines are still functional, and the existing CSTS has been used to drain much of the pumpable volume of tank 102-SY from the 200 West to the 200 East area for evaporation. Important as part of the effort to decrease the total stored volume of wastes at Hanford, this step also frees up 102-SY for both routine and emergency uses. The compatibility of wastes with the sludge in 102-SY is also being further assessed; the more types of SST wastes that can be shown to be compatible

with the residual contents of 102-SY, the more effectively 102-SY can be used for its highest and best use, as the critical staging point to move wastes from the West to the East area.

The regulatory agencies quickly concurred in the use of the existing CSTS, which is definitely not up to current requirements, for the foreseeable future, in return for which DOE has increased the priority for the new CSTS. The schedule for the new CSTS has been accelerated, the draft EIS has been revised when published as the Final EIS to call for only the new CSTS and no new tanks.

The effect of the rest of the Team's recommendations is harder to determine at this point, at least to an outsider. Certainly the importance of waste minimization, contingency planning, better risk analyses, etc., was clearly seen by the Hanford Advisory Board as well as DOE and its contractors when the Team's report was published. One can only hope that these other recommendations have, as appropriate, been adopted and internalized by those responsible for the Hanford tanks and their contents.

REFERENCES

A twelve page bibliography of all references used by the Technical Team can be found in an appendix to the Team's full report, which is referenced below. The report can be obtained from the address shown below for $15.00, to cover the cost of printing, handling and shipping.

1. PAULSON, G., F. PARKER and M. KAVANAUGH, Technical Team Review of Proposed New Double Shell Tanks at Hanford, Paulson and Cooper, Inc., PO Box 1541, Jackson Hole, WY 83001(1995).

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THE TECHNOLOGY/REGULATORY INTEGRATION PROJECT: ENVIRONMENTAL SOLUTIONS THROUGH STAKEHOLDER INVOLVEMENT

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ABSTRACT

The US Department of Energy (DOE) is attempting to clean up hundreds of thousands of acres of contaminated land at federal energy facilities throughout the country, mostly in the West. Currently, the technologies do not exist, or are not fully developed, to do the job. As technologies are developed, they are often met with resistance from public and tribal stakeholders who don't trust DOE and don't fully understand the technologies and their environmental impacts. The problem is compounded by broad "cultural" differences such as tribal/government and public relations, an exclusive operating mode within DOE and a vast gulf in technical expertise. DOE's Office of Technology Development is trying to overcome these barriers through the Technology/Regulatory Integration Project or TRIP. TRIP is a systematic process for involving diverse stakeholders in complex scientific and technology decisions. TRIP may serve as an effective decision-making model for the 21 st Century.

INTRODUCTION

The purpose of this paper is to describe a model environmental decision-making process developed by the Colorado Center for Environmental Management (CCEM) known as the Technology/Regulatory Integration Project

or TRIP. TRIP is a comprehensive process designed to effectively integrate stakeholders into the technology and regulatory decision-making stream. CCEM has recently published a TRIP Implementation Manual and accompanying training program aimed at helping project managers, principal investigators and public involvement specialists integrate public and stakeholder needs for involvement with technology design, development and deployment. TRIP is funded by the Office of Technology Development of the US Department of Energy (DOE).

THE ENVIRONMENTAL CLEANUP PROBLEM

The Western Governor's Association (WGA) project to Develop On-site Innovative Technologies (DOIT) estimated in 1994 that federally owned and managed lands contributed to over 50% of the nation's environmental contamination and cleanup problem. According to WGA, the Department of Energy (DOE) manages approximately 4,000 sites encompassing thousands of acres contaminated with both radioactive substances and hazardous wastes. By its own accounts, DOE must clean up 137 contaminated energy facilities in 33 states and one territory spread over a total of 3,300 square miles. DOE faces the task of remediating nearly 10,000 individual hot spots. Contamination problems at other federal facilities are even more extensive. WGA estimates that the Department of Defense (DOD) has 18,795 contaminated sites and 1,800 installations included in its Installation Restoration Program. The Department of Interior (DOI) lists 422 sites on the Federal Facilities Hazardous Waste Compliance Docket. The total number of abandon mine sites on federal lands is unknown but may run as high as 500,000. The cost to clean up federal sites alone is almost $400 billion based on 1992 figures while the nation's total cost for environmental cleanup is nearly $700 billion.

Environmental cleanups on public lands have been hampered by many factors and these factors have differed to some extent at each site. Common themes do exist however: Stakeholder acceptance, technology advancements, and regulatory approvals are some of the barriers commonly put forward to explain the problem.

Stakeholder Acceptance

Barriers are erected when people who may be affected by an action to cleanup the environment are not adequately involved in making decisions. When people feel left out of a decision that could, from their perspective, have a significant impact on their health, livelihood or quality of life, they will likely resist not only the decision but the process, the institution and people making the decision.

When we think of stakeholders in the environmental cleanup arena we tend to think of angry citizens groups or neighbors whose health or property may be affected by the cleanup process. But a stakeholder can be any party affected or potentially affected by a decision or anyone who has a "stake" in the outcome. Stakeholders can be citizens, public interest organizations, labor organizations, technology developers, government regulators, plant workers, tribal governments, public officials, production industries and environmental consultants.

Technology Advancements

Environmental cleanup problems are currently much larger than the technologies we have to solve them. Many of the needed technologies don't yet exist or have not been approved for commercial use. The publics' repeated mandate to move quickly to clean up federal energy and defense sites and the environmental remediation industries' lack of the technology to do the job are in sharp contrast.

Regulatory Approval

DOE and other government agencies recognize that new technologies are critical to the cleanup of federal energy, defense and mining sites throughout the US. Yet one of DOE's most significant problems in nurturing new technologies is the time it takes to navigate the regulatory process. These time constraints, it is thought, act as a deterrent to the development, demonstration, approval and commercialization of new technologies. How? DOE believes the prime factor is the lack of interstate and interagency cooperation in the permitting process. State and federal regulatory agencies are required to test and approve new technologies before they can be used in the cleanup process. These agencies are reluctant to approve new technologies and are, for the most part, unwilling to accept regulatory approvals from other states and other agencies.

TRIP PRINCIPLES

The TRIP process is designed to help overcome the stakeholder, technology and regulatory barriers described above in part by integrating the key people who represent those barriers, in a systemic way, into the decision-making stream. The TRIP process, when implemented carefully, results in ownership among the parties holding decision-making and "blocking" power. Ownership breaks down barriers. TRIP considers a good decision-making process one that meets the expectations of stakeholders while satisfying the needs of site managers, regulators and technology developers.

TRIP provides six key principles for guiding effective stakeholder involvement projects and programs. The principles are not specific steps in a stakeholder process, but are designed to give guidance in the development of necessary components of a stakeholder plan. The TRIP principles are:

- Bring the right stakeholders to the table
- Develop a decision-making process that effectively integrates stakeholders
- Develop an effective information and data collection program
- Develop a communication and responsiveness program for all publics
- Jointly frame the technology/regulatory problem
- Develop an action plan that integrates technology, regulatory and stakeholder interests

Bring the Right Stakeholders to the Table

Basically, stakeholders are those who are potentially affected in some way by the decision or action being considered. Any interested party could potentially be a stakeholder and all stakeholders, whether they represent primary or secondary issues, should have access to a fair and open decision-making process. One of the most important challenges of effective stakeholder involvement, however, is to actively recruit the right stakeholders. The right stakeholders are those who can best represent the issues and interests at stake, those most affected by the decision or action under consideration and those who have the clear authority to make decisions for the organizations or groups they represent.

Develop a Decision-making Process that Effectively Integrates Stakeholders

This principle has two inter-related parts. The first has to do with empowering stakeholders or giving them the power to influence decisions

that affect them. The second has to do with requiring stakeholders to be accountable and responsible for their actions.

Empowering stakeholders means providing them with opportunities to influence decision-making in ways that meet their expectations. If stakeholders expect to be kept informed of decisions or actions and expect that they will have access to information when they ask for it, they will feel empowered if you meet those expectations.

The second part of the principle has to do with requiring stakeholders to be accountable and responsible for their actions. This goal is more difficult to achieve because most stakeholders, like each of us, will initially see the problem from only one perspective - their own and will likely resist attempts by others to see the problem from other perspectives. Overcoming this barrier is critical to the TRIP process and without it the process will not move forward.

The best way to get accountability and responsibility from stakeholders is to enlist their help in designing a stakeholder involvement process that meets their needs. If the process is reasonable and fair, and has been substantially designed via a partnership, they are more likely to be accountable for their actions and responsible to others and the process.

Develop an Effective Information and Data Collection Program

Decision-making processes often break down because of conflicts over information and data. The classic environmental battle is between two experts who disagree over the data, the interpretation of the data, the data collection process and even the assumptions made in designing the study. Conflicts over information and data occur all the time:

- How much of the contaminant is present?
- What is the risk to human health and the environment?
- What is an acceptable risk?
- How much of it should we clean up?
- What technologies should we use?

These are all questions that are pertinent to the clean up process and require extensive data and information. At every step in the decision-making process there is a need for information and at every step there is the potential for conflict.

TRIP reduces data conflicts by emphasizing joint studies and other procedures that decide on fair and impartial data collection methods before data is collected. TRIP also focuses on integrating risk assessment, management and communication into the stakeholder process, providing stakeholders with independent technical support to verify existing data, providing educational and training opportunities, making information and data collection accessible and presenting information in ways that meet stakeholder needs.

Develop a Communication and Responsiveness Program for all Publics

Many public involvement projects recognize and focus on a core stakeholder group but fail to keep the larger stakeholder group - those who are represented by the people at the table - involved in the process. The larger stakeholder group could be the citizens of a community or tribe, an industry group, plant employees, members of the environmental organization etc. It is not possible to involve all of these people directly in the TRIP process so it is necessary to select people who will represent them. Their involvement is just as critical as the immediate stakeholder group.

A communication program should run parallel to and compliment the stakeholder decision-making process. The communication program should be

easy to access and allow opportunities for people to keep abreast of current happenings by reading objective and relevant material, viewing presentations and videos, attending important events, touring sites and providing feedback.

Jointly Frame the Technology/Regulatory Problem

This TRIP principle requires us to describe the problem we are facing in an integrative way rather than in an "us and them" way. The problem then becomes our problem instead of my problem and your problem. An integrative problem, also called a Joint Problem Statement, requires stakeholders to look at the problem as a set of common and divergent interests, their own and others, instead of hard-line and opposing positions and to take responsibility for the whole problem instead of a single piece. The Joint Problem Statement encourages ownership of both the TRIP process and the problem itself.

The Joint Problem Statement integrates stakeholder interests into a description of the problem that all stakeholders can support and presents it as a question or series of questions. The question is essentially an equation that the stakeholders will try to solve:

How can we satisfy ABC while also satisfying XYZ?

A, B and C are the interests and concerns of the agency and X, Y and Z are the interests and concerns of the stakeholder group. Together they are the array of stakeholder interests and concerns that make up the overall problem.

Develop an Action Plan that Integrates Technology, Regulatory and Stakeholder Interests

An action plan evolves from the Joint Problem Statement. It lays out a plan to answer the question(s) posed by the Joint Problem Statement(s) and integrates the information and data collection program and communication program into the plan. The action plan should be developed, insofar as possible, by the stakeholders and project manager with the help of a facilitator.

The action plan should contain goals, objectives, strategies and tasks, and may also contain more specific Joint Problem Statements that result in goals, objectives, strategies and tasks for specific programs or more detailed problems. For example, the group may choose the Joint Problem Statement that asks:

"How can we bring new technologies on line quickly while assuring that they are adequately tested and approved by state regulators and accepted by the general public?"

Stakeholders now have a problem that they can tackle together. An information and data collection program could emerge from this statement that would integrate stakeholders concerns for trust and accountability with concerns for efficiency, objectivity and truth.

APPLYING THE TRIP PRINCIPLES: DESIGNING THE STAKEHOLDER PROCESS

Successful stakeholder involvement requires considerable planning and preparation. However, because no two stakeholder processes are alike and because the process is shaped, to a large extent, by the stakeholders themselves, the planning and preparation necessary to make the process successful also requires considerable flexibility. In this section we will discuss the design phase. There are six design steps that are consistent with the TRIP principles:

- Define and scope the problem
- Identify and select the right stakeholders
- Design the decision-making process

- Develop an effective data collection program
- Develop an interactive communication program
- Develop a preliminary joint problem statement

Define and Scope the Problem

It is important to define the overall problem in a way that is acceptable to all stakeholders. An acceptable problem definition is one that integrates the perspectives of all stakeholders. Their are two main sub-steps in the definition and scoping process:

- Define and clarify agency and project missions.
- Identify the contamination, technology, regulatory and stakeholder issues and concerns that make up the problem.

Your agency and project missions needs to be examined in the context of the organization and it needs to be understood and accepted by your stakeholders. The mission statements are important because they build credibility for the agency and project and announce your purpose to others. Also, you are mandated to carry out your mission. How you accomplish your mission, however, may be determined through a stakeholder partnership.

Begin by defining the contamination, technology, regulatory and stakeholder issues and concerns that you want the stakeholder group to address. You are preparing stakeholders to tackle the problem and the first step in this process is to define the problem as clearly and concisely as possible using terms and language that are widely understood. Describe the problem first in terms of issues and second as a statement of your concerns about those issues for your agency and for stakeholders.

Stakeholder Identification and Selection

This section describes a three-step approach for identifying and selecting the right stakeholders:

- Identify and select an initial group of stakeholders who represent, in the broadest context, the issues at stake.
- Using the initial group, select a more comprehensive working stakeholder group based on the TRIP principles.

This two-step procedure adds an additional layer of objectivity to the process and better assures that a comprehensive group is selected. The initial stakeholder group should represent all of the issues and concerns you are addressing in the TRIP process and should also be representative of the communities or regions where the project will occur. The comprehensive group should close any gaps that might exist. The most credible stakeholder groups are those with the most comprehensive and representative participation and those that are easily accessible to observers and new members.

Design the Decision-Making Process

Now that you have established a process for selecting stakeholders, let's go back to the problem you are dealing with and establish a process for making decisions. Remember that the decision-making process must integrate the interests of stakeholders. There are four sub-steps to this process:

- Establish a timeline for critical decisions.
- Determine acceptable decision points and decision rules.
- Integrate the timeline with decision points and rules.
- Determine an effective meeting forum.

As you begin designing the decision-making process, remember that your stakeholders may want to modify the process to fit their needs. That's

O.K. In fact, from the standpoint of establishing ownership, it's better if they change it.

The timeline should clearly illustrate the sequence of events that need to take place and the points along the way where decisions and advice are needed. Decision rules let everyone know just how decisions will be made. The most effective decision rule is consensus. A decision in a consensus-based process is triggered when a stakeholder, or the facilitator, asks for a proposal that satisfies mutual interests. When a proposal is introduced the group must try to reach consensus on it by offering improvements until everyone can agree.

Some meeting forums are unknowingly adversarial and actually invite conflict. Others allow and even encourage stakeholders to work together. Your challenge is to create the right forum and atmosphere for constructive dialogue and problem-solving. Forums that tend to invite conflict are those that pit stakeholders against each other in an attempt to win-over a decision-making body. Experience with TRIP has shown that stakeholders are most comfortable with a more traditional forum wherein individuals meet at a neutral location that is comfortable and accessible and where they have the time and energy to talk through important issues.

Develop an Effective Data Collection Program

The quality of information and data is critical to the TRIP process. Environmental decisions hinge on scientific information. Stakeholders need such information to make informed decisions and recommendations. Yet the process for gathering, interpreting and disseminating information is often so contentious that decision-making bogs down. There are five sub-steps in the data collection process:

- Determine stakeholder information needs.
- Identify existing information.
- Identify missing information.
- Identify potential data conflicts.
- Determine a process for collecting and disseminating information.

Stakeholders should rightfully have access to all information that is available to decision-makers and scientists except for information that requires security clearance. They should have access to information even when it may be obvious that the information is too complex to understand or too burdensome to manage. Moreover, stakeholders should not have to invoke sunshine laws to obtain information. Existing information should be identified and catalogued for easy reference and gathered at a location accessible to the public. You should identify all sources of information that exist, even if some of it is in raw form and not easily readable.

What gaps do you see in your data? As you consider the project, what information and data needs do you have that are not currently addressed by objective studies? More importantly, what will stakeholders likely say are the gaps in your data?

What about potential data conflicts? Any time you can reasonably anticipate that people will disagree about a scientific study or body of technical information, you may be entering a data conflict trap. Data conflicts occur when parties have different perceptions of the problem, different methods for addressing the problem or different ideas about what constitutes an acceptable solution to the problem.

As a general rule stakeholders and agency representatives should be thoroughly involved in determining a fair and objective process for

collecting and disseminating information and should be comfortable with the process before any of the information is collected.

Develop an Interactive Communication Program

Stakeholder process managers sometimes forget about the broad public and instead focus almost entirely on their core stakeholder group. It is important to remember that your core stakeholders only represent the broader public - your real stakeholders. You must establish one-way and two-way strategies for communicating both with the broader public and your core stakeholder group. There are two sub-steps in the process.

- Determine an interactive stakeholder communication process
- Determine an interactive public communication process

Your core stakeholder group will attend most meetings and will be kept informed through attendance. Nevertheless, you should have a process for keeping them up to date about decisions and actions that result from meetings and from agency decisions. The best way to keep stakeholders informed is through carefully kept minutes and flip-chart notes.

You will also need a process for communicating with stakeholders about decisions and actions taken by the agency outside of stakeholder meetings. A regular newsletter is the most common tool for such communication and can be easily developed in-house. Two-way communication with the general public is much more difficult and will depend on the nature and complexity of the problem and the magnitude of the stakeholder population. To start, you will need to define the "public" as that population that is broadly represented by the people at the table. Once you have identified these populations, your task will be to develop an effective one- and two-way communication program that will reach them. The most effective programs combine various forms of media to pass information along, monitor broad stakeholder satisfaction and receive input.

A final note on communication. Communicating with the public is no longer a public relations process. The public is too sophisticated to buy into the slick PR approach designed to make the company, agency or project look good. In fact, that approach is part of the problem. The way we communicate with the public should result in improved public relationships.

Develop a Preliminary Joint Problem Statement

The foundation of the TRIP process is the integration of technology, regulatory and stakeholders issues and concerns. In Section 3.5 we discussed the TRIP principle of framing the problem in a way that requires stakeholders to address it in an integrative way. We then called this new problem the Joint Problem Statement.

It is useful to list all of the issues and concerns in the statement, no matter how cumbersome it seems, and then go back and simplify it until it is clear and concise without losing its meaning. Also, use verbs to describe the interests and concerns so they fit directly into the sentence structure. Consider the following example:

How can we...	preserve historic structures and scenic views
	maintain wildlife habitat
	protect water quality and aquatic life
	assure adequate monitoring
	avoid public health risks and
	maintain property values...
while also...	reducing costs
	meeting project deadlines

avoiding duplicative efforts
complying with regulations and standards
reducing risks to workers and the public and
avoiding dueling expert conflicts?

The Joint Problem Statement is now in a form that can be revised and refined by stakeholders until it meets their needs. You can return to the statement at anytime to review your progress, determine whether you are addressing the real problem, and maintain your focus when you get lost.

Develop an Action Plan to Address the Problem

The Action Plan should primarily describe the step-by-step process you will go through to address the Joint Problem Statement. It should also include your timeline, information and data collection program and communication program. The Action Plan should contain the following:

- Project mission
- Goals and objectives for accomplishing the mission
- Tasks to be completed along the way
- Milestones for completing the tasks.
- Results

The TRIP Action Plan is not much different from typical action plans developed for technology or cleanup projects. Its main differences are that 1) stakeholders are involved in developing and refining the Plan, 2) stakeholder issues and concerns are integrated into the Plan and 3) stakeholders are involved to some extent in carrying out the Plan.

Now that you have integrated stakeholder issues and concerns into the Joint Problem Statement described above, you should be able to turn the Joint Problem Statement into a new project mission. Instead of wording the Joint Problem Statement as a question, it can be reframed as your project mission. As an example, let's reframe a Joint Problem Statement and turn it into a project mission. In section 3 we used the following example to illustrate the structure of a Joint Problem Statement:

"How can we bring new technologies on line quickly while assuring that they are adequately tested and approved by state regulators and accepted by the general public?"

Reframed, the project's mission would be to:

"Bring new technologies on line quickly while assuring that they are adequately tested and approved by state regulators and accepted by the general public."

The Action Plan is developed to accomplish the project mission. The value of framing your Joint Problem Statement as the project mission is that it reinforces both the integration of stakeholders in the Action Plan and the logical sequence of steps from the Statement to the Plan. The rest of the Action Plan flows from the project mission. As long as the mission truly reflects the Joint Problem Statement and the Statement is a true stakeholder collaboration, the Action Plan will meet the TRIP requirements.

CONCLUSION

In Section 4 we discussed the TRIP process from the perspective of one who is designing a stakeholder involvement project. The emphasis was on the design phase - knowing the terrain and planning a general path. We also discussed the need to anticipate and remain flexible to the changes that stakeholders will make as they become more involved.

For purposes of this paper we will conclude with the design phase. The next phase, Managing the TRIP Process, addresses the process from the perspective of one who is ready to convene stakeholders or is already

immersed in the daily management of a TRIP stakeholder project. The next phase requires TRIP process managers to revisit all of the steps in the design phase, along with stakeholders, to develop a process and plan that substantially meets their expectations.

34-5

LESSONS LEARNED FROM 17 PUBLIC MEETINGS ON THE FOREIGN RESEARCH REACTOR SPENT NUCLEAR FUEL DRAFT ENVIRONMENTAL IMPACT STATEMENT

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ABSTRACT

The Department of Energy (DOE) and Department of State conducted 17 public meetings during May-June 1995 as part of the public comment process on the Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and ultimately eliminate, highly enriched uranium from civilian commerce. One of the management alternatives in the Draft EIS is for the United States to accept and manage spent nuclear fuel from foreign research reactors (only spent fuel containing uranium enriched in the United States). Spent fuel would be stored and/or chemically separated at a DOE spent fuel management site.

To implement the alternative of U.S. acceptance, the Draft EIS identified and evaluated five candidate DOE sites and ten candidate ports of entry. Spent fuel would arrive into a port by sea and be unloaded for shipment to a DOE site by rail or truck.

In preparing the Draft EIS and preparing for the public comment process, DOE and Department of State set out to communicate the global and national issues behind the proposed policy and communicate the results of the impact analyses for the various alternatives. The 17 public meetings were held at candidate site and port locations and in Washington, D.C. DOE used an informal, interactive format for the meetings.

This paper will discuss the results of the meetings and effectiveness of DOE's meeting strategy. Over 900 individuals attended the 17 public meetings, with individual meeting attendance ranging from 2 to 250. There was an overwhelming difference, depending on location, in the public reaction and effectiveness of the interactive meeting format. Port communities, which were not generally familiar with DOE issues, focused on port-specific issues. The informal, interactive meeting format worked well in all locations except for the West Coast, where the attendees were highly organized in opposition and preferred a more traditional recorded hearing approach.

INTRODUCTION

In April 1995, the Department of Energy (DOE) and the Department of State released for public comment the Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. The proposed action is for the U.S.

to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and ultimately eliminate, highly enriched uranium from civilian commerce.

Several management alternatives to implement the proposed policy were analyzed, including U.S. acceptance of the spent fuel for management at one or more DOE sites, management of the spent fuel at overseas facilities (under certain conditions), and a combination of alternatives. For alternatives involving U.S. acceptance of the spent fuel, the Draft EIS identified five candidate DOE management sites and ten candidate ports of entry. These sites and ports are identified on the map in Fig. 1. Environmental impacts were analyzed for the shipment of the spent fuel by sea, entry and unloading at each candidate port of entry, transportation (truck and rail) to each of the five candidate DOE sites (representative routes), and the interim storage and/or chemical separation at candidate DOE sites. In addition to analyzing the environmental impacts of the various alternatives, the Draft EIS analyzed the policy implications of implementing or not implementing the proposed policy, including implications to U.S. nuclear weapons nonproliferation policies and objectives. Thus, the Draft EIS dealt with issues ranging from worldwide and national policy issues to community-specific issues.

Fig. 1

STRATEGY FOR CONDUCTING THE PUBLIC MEETINGS

The Message

In preparing for the public comment process, DOE and Department of State set out to communicate the global and national issues behind the proposed policy and communicate the results of the impact analyses for the various alternatives. As will be discussed later, local concerns became the major factor in many of the public comments at certain port communities. Informational materials were developed to complement the Draft EIS (summary document, fact sheets, video, poster exhibits). The materials attempted to communicate the full range of considerations in the Draft EIS in an easy-to-read manner. Again, the broader global and national issues were emphasized, along with general statements of potential impacts, which were found to be low.

Meeting Format

Unlike the traditional public hearing process, where commentors present time-limited oral testimony recorded by a transcriber, DOE and Department of State experimented with a more informal approach in order to encourage open dialogue, questions and answers, and one-on-one interactions. Meetings started with a one-hour "open house" where people could, at their own leisure, peruse the exhibits and materials and have one-on-one discussions with project personnel. A display area was also provided for informational materials of other groups. Following the open house session, the meeting session started with a 15-20 minute presentation by a DOE representative and then opened up to dialogue and questions and answers. A third-party facilitator was used to help keep the dialogue flowing, assure fairness, and record on flip-charts key comments and questions. As discussed later, this format worked well under some conditions but not in others.

Advertising and Information Dissemination

In addition to the required Federal Register Notice to announce public meetings, DOE and Department of State utilized additional means to advertise the meetings at the local level. An official DOE/Department of State announcement was published in local newspapers at all site and port locations at least one week before the meeting date and the day prior to the meeting. A seven-minute video was provided to local cable-access stations to run several times during the week prior to the meeting. Public service announcements were provided to NPR radio stations and TV networks. A distribution list of State officials and port authorities was used to mail out the Draft EIS in advance. In all advertisements, a 1-800 number was provided for the public to request copies of the Draft EIS or Summary.

Locations of Public Meetings

Initially, 16 public meeting were planned -- five site locales, ten port locales, and Washington, D.C. The idea was to get as close to the community as possible. A second Tacoma area meeting was added at the request of local officials in Tacoma (the first meeting was in Seatac, which was between Seattle and Tacoma). Figure 1 shows the meeting locations.

RESULTS OF PUBLIC MEETINGS

Over 900 individuals attended the 17 public meetings, with individual meeting attendance ranging from 2 to 250. The most heavily attended meetings, with about 70% of the total attendance, were at the West Coast port communities. Table I shows the approximate head count (not all attendees chose to register) at each meeting.

Table I

There was an overwhelming difference, depending on location, in the level of organized opposition and effectiveness of the interactive meeting format. The level of advertising and organization by opposing activist groups and local and Congressional officials played a major role in the attendance and demeanor of the meetings. A general observation was that the West Coast port locations were the heaviest attended (through organized activism), the most focused on local fears, and the least interested in interactive dialogue with DOE and Department of State. Although some East Coast locations showed lack of support of spent fuel shipments through their ports, the interactive format worked well and people seemed satisfied with the dialogue and question and answer sessions.

Issues Raised

The issues raised at the meetings differed depending on whether they were site or port locations. Site meetings received low attendance (2-20) and the issues focused primarily on existing problems at sites, funding issues, storage vs. chemical separation technologies, concerns about de facto storage, and nonproliferation issues.

DOE's experience at the port communities was new (other than during scoping meetings, which involved a lesser number of candidate ports). These communities were not familiar with DOE issues (with the exception of a few) and were, for the most part, opposed to shipments of spent fuel through their ports. Although many supported a proposed policy, in general, they did not want nuclear waste in their communities. In addition to the many statements of direct opposition to spent fuel shipments, comments and questions at the port meetings focused primarily on port-specific issues, such as:

- Port selection process

- Risks of accidents at port or on land
- Health risks from exposure to spent fuel
- Risks of terrorism or sabotage
- Local emergency response capability
- Local traffic conditions and routing
- Funding for training and equipment

Format of Meetings

With the exception of the West Coast port meetings, the informal, interactive meeting format worked well. The open house session could have been shorter (people tended to only spend five to ten minutes at the exhibits, then waited in the meeting room). However, people did peruse the exhibits and pick up the information materials. It also helped to provide a more comfortable and informal setting for the public. Written comment forms were also available for those who wanted to write their comments that night. With few exceptions, the meeting sessions lasted for two-to-three hours, until there were no more questions. In most cases, the atmosphere was friendly and the dialogue was productive. Many of the East/Gulf Coast and site meeting attendees commented positively about the interactive format. One observation that can be made was that the East/Gulf Coast and site meetings were small enough (2 to 60) to accommodate an interactive session. Such was not the case for two of the four West Coast meetings.

At the four West Coast meetings (Concord, Portland, Seatac, and Tacoma), the majority of public attendees were members of highly organized activist groups or were there as a result of widespread local press and organized activity. Attendees were largely unified in opposition to use of their port and wanted to present their statements for the record. There was little interest in interactive dialogue with the DOE officials at the meeting. DOE was flexible in changing the meeting format to accommodate those that wanted to read statements. Attendees at three of the meetings also objected to the fact that DOE did not have a court reporter to transcribe the statements for the official record. (DOE was accepting written comments from individuals for the record; and the meeting notetakers' notes from each public meeting would also be included in the Final EIS.) The second meeting in Tacoma was recorded on tape and transcribed for the record.

Advertising and Information Dissemination

The amount of advertising was often raised as an issue at the public meetings. Many people said they did not know of the meetings "until the last minute." Many said the newspaper ad was too small and did not provide enough information. Although the advertising was consistent in all locations, it was difficult to judge its effectiveness in that attendance varied so widely.

However, one case-study example on the effectiveness of ads is the meeting held at Tacoma. A full-page, provocative ad was published by a local activist group (picture of a baby with a radiation symbol on its forehead). The Tacoma meeting had 250 attendees, with 200 turned away at the door (because of fire codes). Obviously, the petition drives and media attention, combined with the ad, had a greater impact on attracting people's attention than DOE's level of advertising.

Many people commented that they did not know about the Draft EIS or how to get it. They commented that sending copies of the Draft EIS and Summary to the Public Readings Rooms was not sufficient.

Meeting Location

Because most meetings were held in the communities most affected, meeting location was not an issue except for the Tacoma port area. The first meeting was held in Seatac, which is half-way between Tacoma and Seattle. Many in the Tacoma area argued that the location was one-half hour away from the affected community and posed hardships on those wishing to attend. In response to overwhelming Congressional and local community requests, DOE added a second meeting in Tacoma proper.

KEY LESSONS LEARNED

Based on DOE's and Department of State's experience with these public meetings, several key lessons were learned that could be applied to future meetings of a similar nature. These include:

When setting the meeting format, recognize that the public in some areas, possibly those with more organized activist communities, may prefer a traditional meeting style where the public's statements are read into the record. Government agencies conducting such meetings would have to determine whether it is better to simply accede to such local preferences (which would generally give opposition groups a highly visible medium through which to make their case), or to make extremely clear in the meeting announcements the format that would be used for the meetings.

Location, Location, Location! Make sure the meeting facility is within the community most affected.

Utilize local libraries and other local services to the maximum extent possible to disseminate information. Multiple copies of Draft EIS Summaries should have been sent to each local library in the affected communities. Fliers would have been useful for distribution to schools, libraries, civic centers, malls, and other local places well in advance. (In Tacoma, activist groups distributed 3,000 of their own fliers, which contributed largely to the high number of attendees.) The announcement section of major and local community newspapers would have reached more people than an advertisement (unless the ad was large enough to stand out).

Recognize that no matter how far-reaching and global an issue may be, communities are most interested in how the issue affects them personally. Be prepared to address local issues.

34-6

NCPP ROCKY FLATS COMPETENCE PROGRAM - A CASE HISTORY

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ABSTRACT

The National Conversion Pilot Project (NCPP) in unique. The Project involves reuse in three ways; shutdown facilities, scrap materials and displaced workers. This is being done on a Department of Energy (DOE) site and will culminate in the award of a manufacturing license for a private company to operate refurbished facilities.

The Competence Program of the NCPP has responded to both the regulatory demands of the DOE and the needs of business.

The Competence Program has three major phases: 1) links to pre-NCPP history of potential employees, employee selection and Base Competence

Verification, 2) Additional Skilling & 3) Pursuit of Excellence. Phases 2) & 3) are collectively referred to as Competence Enhancement.
This topic will interest those involved in privatization, training or cultural change.

CASE HISTORY - THE NCPP IDEA

In 1993 Dr. Dennis Floyd, Vice President of Manufacturing Sciences Corporation (MSC), had the idea for "three-way" reuse of resources at the DOE's Rocky Flats Environmental Technology Site (RFETS) in Colorado. Firstly, to clean and refurbish machines and buildings previously used to form beryllium, depleted uranium and stainless steel. Secondly, to recycle ex-weapons component beryllium, depleted uranium and contaminated steel into commercial products. Thirdly, to reemploy ex-weapons plant workers. This idea eventually became what is now known as the National Conversion Pilot Project (NCPP).

MSC has a business association with British Nuclear Fuels plc (BNFL) in the United Kingdom. One of BNFL's plants in England, at Springfields near Preston, had been implementing some major, business-driven change programs since 1989. These programs were increasingly people-focused, rather than technology-focused. In particular, a competence program designed to cope with an uncertain business future, called the Additional Skilling Project, which involved a form of multi-craft work, caught Dennis Floyd's attention. This was because the Additional Skilling Project had already dealt with the difficult cultural and competence gap issues then facing the NCPP at the RFETS.

To test if American workers would have support for the BNFL Springfields type of approach, Dr. Keith Biddle, the BNFL Additional Skilling Project Manager, met with the leadership of the local United Steelworker's Union at RFETS in November 1993. He explained in detail how the system worked at Springfields and how the outcome affected the workers. The response of the Union leaders was positive; the main reason for that was the participatory and fit-for-purpose theme.

The NCPP commenced in April of 1994 with Stage I - Feasibility and Planning awarded to MSC. Stage II - Cleanup and Refurbishment began in October 1944 also awarded to MSC, and will take until late 1997 to complete. Stage III - Manufacturing Operations will follow after a competitive bid process.

WHAT WAS NEW ABOUT THE NCPP COMPETENCE PROGRAM?

Answer: a fresh approach to competence on a DOE site. The NCPP treats safety and technical competence as seriously as all good organizations. Also, the NCPP approach of competence needs analysis, structured training delivery, competence assessment and auditable evidence of competence is paralleled in the best organizations. What is different for the NCPP at Rocky Flats is the extra features of the Competence Program. These are a combination of: a) a team competence approach, b) business-led ("flexible working") training perspective, c) emphasis on "cultural" competence as well as safety-technical competence, d) competence activities are an integral part of the Project's master work and e) speed of training design and delivery.

THE TRAINING PLAN (COMPETENCE PROGRAM PLAN) AND THE "COMPETENCE ENHANCEMENT PROCESS"

The NCPP Training Plan lays out the scope, objectives, assumptions, qualification principles, original schedule, resources, management systems and controls of the Competence Program. It was approved as a Stage I deliverable of the NCPP by the DOE.

The Program has three major phases: 1) links to pre-NCPP history of potential employees, employee selection and Base Competence Verification, 2) Additional Skilling & 3) Pursuit of Excellence. Phase 2) & 3) are collectively referred to as Competence Enhancement. Figure 1 shows the relationship between these phases.

Fig. 1

The whole process is guided by and feeds into the NCPP Stage II Mission. See Fig. 2

Fig. 2

BASE COMPETENCE

NCPP employees brought a spectrum of skills to the Project at the time of hiring. For example, office skills, decontamination skills, engineering skills, managerial skills, specialist technical skills, health and safety knowledge, regulatory knowledge and knowledge of Rocky Flats/DOE systems. A key feature of the NCPP Competence Program is the recognition of these skills or "Base Competence."

Evidence of Base Competence is collected and is the baseline for the further training needs of the individual. "Taking credit" for the pre-hiring skills of a workforce is the smart thing to do from the business perspective also (the cost of some of the multimillion dollar Additional Skilling schemes at BNFL Springfields in the UK were halved by carefully integrating existing skills and other training initiatives rather than assuming that the workforce has little Base Competence). However, for quality assurance, adequate evidence of Base Competence MUST exist; in other words (in the 1900s) "if you can't prove it, then you haven't done it."

ADDITIONAL SKILLING

The term "Additional Skilling" is purposely chosen to convey several differences between "traditional training" (specifically as perceived by the workers at Rocky Flats) and the NCPP Competence Program. Additional Skilling has been designed with the comments of the workers in mind:

Table I

The steps for implementing Additional Skilling are systematic. Human resource aspects and the applicable skill areas are also addressed:

Table II

Table III

PURSUIT OF EXCELLENCE

The pursuit of excellence (i.e., pursuit of project specific quality and value as perceived by the customer) is just beginning on the NCPP, and is internally focused for now. The initial steps are: a) small shift towards competence assessment (without training) as the norm rather than training and competence is always the norm, b) scheduling the minimum training for the role up-front, then performing "just-in-time" training as an integral part of the decontamination or manufacturing schedule, c) the introduction of participatory Competence Currency reviews, d) two-way appraisal (upwards as well as downwards) with an emphasis on skills and personal performance, and e) continuous improvement through Quality Improvement Teams made up of practitioners (not theoreticians).

MANAGEMENT SYSTEMS

NCPP personnel have developed procedures covering: a) employee selection and verification of base competence (phase 1 above, b) production of Personal Competence Plans, competence record keeping, competence needs analysis, training package design, training delivery control and quality verification and competence assessment (phase 2 above), and c) training

scheduling and competence currency (phase 3 above). During Stage II of the NCPP, the Competence Program is overseen by DOE as part of the NCPP Quality Assurance process, through the NCPP Quality Program Plan and NCPP Quality Procedures. The net effect of these systems is a streamlining compared to prevailing Rocky Flats systems. The NCPP systems are inexpensive and productive, in that they adequately improve workforce competence without major budgetary impact.

ORGANIZATIONAL STRUCTURE

The NCPP organization has a Teamwork structure and is best illustrated by discussing the Decontamination Work Teams: Each Work Team is a "self sufficient work cell" in a network of such "cells." The Teams support each other and it is common for some members of one Team to temporarily transfer to another Team. Each Work Team has the personnel/skills mix to fulfill their Team's mission. The skills possessed/needed by the individuals in each Team are governed by the competence requirements of that Team as a whole.

In addition, other (small) teams are formed as needed within the whole NCPP network, to deal with specific issues such as technical problems or procedural blockages.

RESOURCES AND SCHEDULE

The Competence Program is a major integral part of the Stage II schedule. This is because the Competence Program has been resourced from within the Project and because it has always been seen as a critical path enabling activity (not an after thought). In particular, key personnel who are also practitioner or experts (such as Team Leaders and other managers/technical staff), with several facets to their role, are utilized as Skill Unit (training module) writers, Trainers or Competence Assessors. This brings workplace "ownership" of the Competence Program. But, to make it credible, the Program must be workable, i.e., well managed with the use of good scheduling tools. Also, most importantly, utilizing Work Team members as Instruction/Procedures writers and Skill Unit developers has been a major success and one of the best examples of Continuous Improvement that this author has seen.

Skill Unit (training module) manuals are developed, typically, by two practitioners or experts and the Competence or Training Manager, on a "lifeboat" basis in a Unit Development Team (UDT). The UDT utilize procedures and the real workplace to create the Skill Unit Manual. The authority to approve the Skill Unit lies with the Unit Development Team. This approach gives fit-for-purpose, workplace credible training. Speed of response is rapid. A typical Skill Unit can be developed within the Project, which has only one full-time Competence Program person, within two to three weeks using the UDT approach. This speed of response has served the Project well when faced with changing regulatory requirements or work scope.

Training and/or Competence Assessment then follows. These activities are also conducted by in-house practitioners or experts, usually by one of the UDT members. The use of the classroom environment is minimized in favor of the real workplace, e.g., presses, rolling mills, furnaces, power tools, elevated surfaces, confined spaces, industrial chemicals. A major emphasis of the program is proving competence in the same circumstances under which real work will be done.

Stage II of the NCPP started in earnest in January 1995. The schedule called for 35 Skill Units manuals to be created by the end of June 1995. Training delivery targets were 700 personunits (Skill Units x

participants) by the end of September 1995 and a further 165 by the end of January 1996.

The project is undergoing rescoping at present and targets for 1996 are not yet set.

CULTURAL CHANGE AND VEHICLES FOR CHANGE

The cultural change issue at Rocky Flats is a highly significant feature of the NCPP. Rocky Flats had slipped into a state of progressive atrophy since the enforced shutdown and subsequent change of mission during the early 1990's. Even though the Project is bold and positive, and the NCPP employees do not work for the Management and Operations Contractor or the Integrating Management Contractor organization, the people on the Project are either ex-Rocky Flats workers, spend a lot of time with current Rocky Flats workers or utilize Rocky Flats IMC systems for some aspect of their work. There is potential for a perceived lack of purpose to influence the day-by-day workplace behavior of Project personnel at all levels. This can be due to "old habits" resurfacing, enthusiastic ex-Rocky Flats NCPP workers being further frustrated or people new to Rocky Flats becoming "infected."

For the reasons above, the NCPP Competence Program was designed to be an integral part of the Project's cultural change initiatives from the start, and it is.

Specific education on cultural change is given to employees which covers the NCPP mission, why success & mission are interdependent, the change process & how people cope with it, team responsibilities towards safety, quality & continuous improvement and how day-to-day activities such as a training course or a dismantling task further the mission. However, it would be naive to believe that this upfront education was a one-shot panacea.

More over, ongoing and positive cultural change depends on planning and managing that change. The NCPP has an annual Cultural Action Plan (CAP). Commitment to cultural change by the workforce is not automatic, but is vital, and is recognized in the CAP. The CAP has an emphasis on participatory change; it is this author's experience that participatory change is much less difficult/"painful" than enforced change (and enforced change rarely works). Cultural change takes time; the implementation process must be monitored constantly, and the "tools" used to steer the change vary and need to be prepared in anticipation. Instant cultural excellence is unknown!

One of the most effective means of converting plans for change into actual change in the workplace is through a network of Work Teams and short-lived Quality Improvement Teams tackling real workplace problems. The Project has around 50 specific examples of words becoming action through this approach, and these vary from Team Mission revitalization to raising the quality of Operator Instructions or solving rigging problems for example.

ACHIEVEMENTS SO FAR AND STATUS OF STAGE II OF THE NCPP

Within six months 35 Skill Units manuals were created, thereby satisfying the initial demand as planned. The original target of 700 personunits of training delivery by the end of September 1995 was successfully met, as was the target of 810 personunits by the end of 1995. Note that the NCPP employs about 50 people.

The first annual Cultural Action Plan has been completed. Half of the first building has been decontaminated and work on the second half started in January 1996.

The reaction of the NCPP personnel to this approach has varied from tacit acceptance to disciple-like enthusiasm. The only negative response is one of frustration in not being able to push the Competence Program and other innovative aspects of the NCPP forward faster. The frustration arises from a realization that the systems experienced in the past were unnecessarily cumbersome, the usual problem for any project, i.e., not enough hours in a day and that the in-house practitioner can and should be empowered, and once empowered is the person most likely to "tune into" and address the competence needs of the Project.

During 1996 the aim is to consolidate the progress made so far and to begin the Project's pursuit of excellence process.

SUMMARY

The NCPP Competence Program has deliberately combined three aspects:

Competence Enhancement: a project managed process which is end-user friendly,

Work in Teams: the management of Team Competence and utilizing the Project's in-house work place experts to spread competence, and

Mission/Vision and managed cultural change.

So far the Program has worked successfully, and enabled the establishment of the NCPP infrastructure and the decontamination/refurbishment of millions of dollars worth of equipment and facilities.

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UNDERSTANDING EACH OTHER: THE DOE/NATIONAL ASSOCIATION OF ATTORNEYS GENERAL (NAAG) COOPERATIVE AGREEMENT

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ABSTRACT

Since 1993, the Office of Environmental Policy and Assistance has been engaged in activities pursuant to a Cooperative Agreement with the National Association of Attorneys General (NAAG). The primary purpose of this project is to foster open communication among the Department and State Attorneys General representatives, using NAAG as a facilitator on legislative and compliance issues in a nonadversarial atmosphere.

In these scarce budget times, it is hoped that expensive litigation can be avoided by engaging in discussion with the States on compliance problems before these issues become the subject of formal disputes. This project is a benefit to the Department, in terms of potential litigation cost savings, as an avenue to enhance DOE compliance with environmental regulations, and provide a forum for a unique stakeholder (NAAG) to exchange information with Departmental representatives.

INTRODUCTION

Birth of an Idea

As a result of interactions between the Department of Energy (DOE) and the National Association of Attorneys General (NAAG) during the Federal Facilities Environmental Restoration Dialogue Committee (FFERDC) meetings, the Director of the Office of Environmental Compliance (now the Office of Environmental Policy and Assistance), the Legislative Director of NAAG and the Environmental Protection Agency (EPA) Deputy Assistant Administrator of the Office of Federal Facilities Enforcement (OFFE),

began discussions to determine the feasibility of establishing a cooperative agreement among the three organizations for the purpose of facilitating communication on regulatory compliance issues with the goal of avoiding litigation. Such an agreement would also educate all parties on Departmental policies and procedures, provide a historical perspective of DOE's environmental programs and establish an atmosphere conducive to resolution of environmental issues prior to litigation.

The impetus to establish such an agreement was the realization that legal and technical issues regarding compliance and enforcement at DOE facilities will continually arise as the weapons complex is reconfigured and downsized. A large number of compliance agreements are currently either in place or being renegotiated in the twenty-two states hosting DOE facilities. These facilities have required substantial resources for operating regulatory and enforcement programs, and will continue to require considerable expertise, time and resources of the states and the Department.

The State Attorneys General play an important role in environmental enforcement and compliance issues, however, they usually are consulted only when negotiations fail and litigation is necessary. NAAG, in its role as the membership organization of the Attorneys General of the 50 states, Commonwealths, Territories, and the District of Columbia, assists the legal officers in fulfilling the responsibilities of their offices and assists in the delivery of high-quality legal services. NAAG was therefore particularly well-suited to facilitate communication among DOE, EPA, and the Attorneys General.

One benefit of the FFERDC discussions was the opportunity for the Department and NAAG to exchange information and gain an understanding of each other's roles in environmental restoration. To capitalize on the momentum of these discussions, DOE researched potential approaches to jointly work with NAAG. In 1994, the Department entered into a one-year interagency/ cooperative agreement with the Environmental Protection Agency (EPA) and NAAG. Upon the expiration of that agreement, DOE entered into an eighteen month cooperative agreement with NAAG in 1995.

Goals of the Agreement

The goals and objectives of this agreement are to: examine the relationships between enforcement and compliance at DOE facilities; analyze the roles of the states, DOE and EPA in facilities' matters with the goal of improving understanding of these issues at the state and federal level; ensure increased compliance by DOE with federal and state requirements; identify alternative dispute resolution mechanisms to reduce the need for enforcement through litigation by EPA and the states; enrich the states' understanding of and interaction with EPA regions and DOE on enforcement and compliance matters, with a focus on enhancing the states' role; and, provide an opportunity for state and federal personnel to interact and become aware of each other's professional responsibilities, expectations, and limitations.

ACCOMPLISHMENTS OF THE PROJECT

Since the inception of this project NAAG has periodically published the DOE Environmental Issues Bulletin. The bulletins have focused on major DOE environmental issues, such as renegotiation of environmental agreements and budget cuts. Articles are provided by State Attorneys General representatives and/or DOE environmental representatives. The bulletin is distributed to over 300 persons, including state regulators,

State Attorneys General, EPA staff, and representatives of DOE headquarters (HQ), field and contractors.

A DOE/NAAG workgroup has been established to promote open dialogue between DOE and the states on environmental compliance related issues with an eye toward developing creative solutions to potentially litigious issues and to provide an opportunity for State Attorneys General representatives and DOE representatives to examine and have joint discussions on legal issues related to statutes such as the Clean Water Act (CWA), the Resource Conservation and Recovery Act (RCRA), the Comprehensive Environmental Response Compensation and Liability Act (CERCLA) and the Endangered Species Act (ESA). This workgroup has been working on guidance which addresses provision of radionuclide information to the states upon request. Preliminary discussions by this workgroup have also focused on legal issues related to reauthorization of the CWA -- specifically issues related to a waiver of sovereign immunity, removal of the exemption for radionuclides and applicability of state groundwater protection laws to federal facilities. Additionally, discussions have been held on the possible impact of the reauthorization of CERCLA and RCRA on DOE restoration and waste management activities.

DOE and NAAG have also held three workgroup meetings to discuss various environmental compliance issues. The first meeting was held at DOE HQ in Washington, D.C. on May 17-19, 1994 and focused on the organizational structure of the Department and issues of interest to the State Attorneys General representatives. Topics discussed included DOE's budget process, the relationship between DOE and its M&O Contractors, CERCLA and CWA Reauthorization, RCRA/Atomic Energy Act (AEA) overlap and RCRA compliance issues.

The second meeting was held in Pasco, Washington on October 25-27, 1994 hosted by the Richland Office of Chief Counsel. This meeting began with a tour of the DOE Hanford site and approximately fifteen Attorneys General representatives, State Agency Attorneys, and EPA Region X representatives, DOE Field Attorneys and Program representatives participated in the meeting and the tour. The discussions focused on issues such as the budget process, RCRA/CERCLA integration, RCRA/AEA overlap, and the National Environmental Policy Act (NEPA). One of the highlights was a discussion from the DOE, EPA and Washington State viewpoint on the Hanford Tri-party agreement. The discussion regarding the Hanford Tri-party agreement also focused on the budget provisions in the agreement and how they had served to increase Washington State's involvement in the budget process. The third meeting was held in Washington, D.C. on October 11-13, 1995. Major topics discussed included DOE's Materials in Inventory (MIN) initiative, the Federal Facility Compliance Act (FFCAct) agreements from both a DOE and the state perspective, the Department's RCRA/CERCLA integration draft guidance, and the state perspective on pending federal environmental legislation.

EFFECTIVENESS AS A STAKEHOLDER TOOL

Stakeholder participation is normally perceived as Departmental officials working in tandem with citizens who live near the site and have a vested interest in the decisions that are made regarding cleanup at that site. In fact, the Department has made great strides during the current administration to educate the public to ensure meaningful participation in decisions affecting DOE sites. However, State Attorneys General, who enforce the laws of the state (including environmental laws) are also

stakeholders and traditionally have not been included in this educational process.

This unique arrangement between DOE and NAAG provides a mechanism for that educational process to occur. Attorneys General and their representatives normally do not come into contact with DOE until an impasse has been reached with respect to resolution of an issue. At this juncture, the situation has become polarized and often the only resolution is litigation. Through this agreement, the State Attorneys General representatives and DOE have the opportunity to openly discuss environmental compliance issues that could potentially end in litigation or that have been the cause of some conflict in the past and attempt to devise a way to avoid such conflicts in the future.

There is no magical process or system being utilized to foster openness and understanding between DOE and State Attorneys General representatives under this cooperative agreement. The key is communication, discussion and sharing of information about the DOE organizational structure, and current DOE issues through the use of vehicles such as: the DOE Environmental Issues Bulletin, joint conference calls, and workgroup meetings.

The DOE Environmental Issues Bulletin is a useful tool for educating the states about current issues at various DOE sites. Since the inception of this project, the bulletins have focused on a variety of issues of importance to DOE and the States. For example, the April-May 1994 issue focused on activities related to the Hanford site, such as the January 1994 amendments to the Tri-party agreement and the Mentoring Program at the Hanford Tank Farms. The March-April 1995 issue featured articles on DOE's Environmental Management (EM) program proposed budget cuts and the Federal Advisory Committee on External Regulation. The September-October 1995 issue featured articles on criminal liability concerns for federal officials and implementation of DOE's strategic alignment. This type of information is especially useful during these austere budget times when DOE must prioritize its cleanups. The Attorneys General's understanding of the total picture can only make prioritizing cleanup activities at DOE facilities easier.

The workgroup meetings are another useful tool in facilitating openness and communication between DOE and the State Attorneys General representatives.

The first meeting which was held at DOE HQ was invaluable because it gave NAAG and the Attorneys General representatives an overview of DOE's organizational structure. For example, a presentation was given on the relationship between DOE and its M&O contractors. A presentation was also given on the budget process which in the last couple of years has become of increasing interest to the States.

The second workgroup meeting held in Pasco, Washington, near the Richland Operations Office, was especially beneficial in facilitating openness because the State Attorneys General representatives were given an opportunity to tour the Hanford site, one of the largest sites in the DOE complex. By taking our enforcement officials on a tour of the site, DOE is promoting an air of openness and cooperation that has not always been the case in the past. It also provided the Attorneys General with a new perspective on some of the environmental issues DOE faces everyday.

The most recent meeting in October, 1995 continued to build upon the success of the two prior meetings. The members of the workgroup openly discussed contentious issues such as criminal liability. All parties left

the meeting with a greater awareness of the problems faced by the regulator and regulated communities.

Openness and cooperation between the States and DOE is also being fostered by the level of DOE participation in this project. For example, Field and Headquarters Counsel representatives have been designated to participate in discussions with NAAG and the States. It is especially important for DOE Counsel to participate in these nonadversarial discussions because they are the DOE officials most likely to come into contact with the Attorneys General representatives. Although we may not always be able to avoid litigation on a particular issue, it is hoped that the interactions that have taken place pursuant to this agreement will at least enhance DOE's credibility in the eyes of its enforcement officials.

FUTURE INITIATIVES

Natural Resource Damage Assessment Discussions

Under CERCLA, as amended by the Superfund Amendment Reauthorization Act (SARA) and the National Contingency Plan (NCP), monetary damages may be assessed when natural resources (NR)--land, fish wildlife, biota, air, water, groundwater, drinking water supplies and other such resources--are "injured by a release of oil or hazardous substances." That is, when the selected CERCLA remedy is incomplete in addressing the NR injury produced by a release, the Natural Resource Trustees may file a Natural Resource Damage Claim against the party responsible for the release. The claim is made for the value of the "residual" injury that was not or could not be addressed by the selected remedy. This liability for NR damages is in addition to cleanup costs for a release, and under SARA 120, federal agencies can be held accountable for the additional costs. Liability for NR damage claims based upon injury to natural resources was established under 107(a)(4)(c) of CERCLA.

In dealing with injured natural resources under its jurisdiction, DOE is required to play a "dual role." Pursuant to CERCLA, as amended by SARA, DOE, as a federal agency, serves as the CERCLA Lead Response Agency for removals and response actions at its facilities. Under Executive Order 12580, and the revised NCP, the Secretary of Energy is appointed to serve as the Primary Federal NR Trustee for natural resources on, over, or under land that DOE manages. This situation is further complicated in cases where DOE may be responsible for a release and the subsequent injury to natural resources under its jurisdiction (or in cases where the releasee may have caused injuries to natural resources under the jurisdiction of other Trustees). DOE's mandate to function in multiple roles does not, however, reduce the standards to be met for each particular role. For example, DOE's NR Trustee responsibilities are not intended to supplant the requirements for the Remedial Investigation/Feasibility Study (RI/FS) process mandated for CERCLA Lead Response Agencies. (1)

Under this agreement, an NRDA discussion group has been created. Preliminary discussion among the Departmental staff and State personnel indicates that the states perceive that the Department does not take the natural resource damage issue seriously. The states have also indicated the need for funding to perform their natural resource assessments. During 1996, this group will propose a pilot in conjunction with a DOE operations office, and state and EPA regulatory personnel to develop a plan to maximize existing funding to enhance NRDA projects. Additionally, this group will examine success stories and will create a "Lessons

Learned" report to inform both state and DOE personnel of the NRDA process.

Alternative Dispute Resolution Discussions

The Administrative Dispute Resolution Act (ADRA), 5 U.S.C. 571 et seq., authorized and encouraged federal agencies to employ consensual methods of dispute resolution. Under the ADRA, each federal agency is required to designate a senior official as a dispute resolution specialist, establish a policy addressing the agency's use of alternative dispute resolution (ADR), review contracts and grants for appropriate inclusion of ADR clauses and provide for department-wide training on ADR.

DOE established an Office of Dispute Resolution in the Office of General Counsel in 1995. The Department's interim ADR policy statement was recently published. It emphasized the Department's commitment to the use of ADR as a management tool to prevent or minimize the escalation of disputes, and to resolve disputes at the earliest possible stage. One focus of the Department's ADR efforts is on the prevention of disputes, or at least early intervention, i.e., before litigation has been initiated. (2) The Department recently drafted proposed environmental compliance agreement dispute resolution language. This proposed language has been shared with NAAG members and it is hoped that through meetings and discussions, a streamlined dispute resolution process will be developed in 1996. Staff will also propose a workshop session conducted jointly with NAAG to highlight the benefits of the ADR process.

State Laws and Regulations Workshop

A recent analysis of environmental enforcement actions issued against the Department indicated that over 90% of the actions were violations of the Resource Conservation Recovery Act (RCRA). Section 3009 of RCRA allows states to enact their own programs as long as they are at least as stringent as the federal program. Further analysis of the violations revealed that over 90% of the RCRA violations were issued by state enforcement agencies. To reduce the number of violations against the department, it has been proposed to develop in conjunction with NAAG, a State Laws and Regulations workshop in 1996 for DOE and Contractor personnel to highlight the differences between state and federal laws.

Workgroup Meetings

The Spring, 1996 workgroup meeting will be held in Albuquerque, New Mexico with a tour of the Los Alamos National Laboratory. Due to anticipated budget constraints, the location of the Fall, 1996 meeting has not been determined; however, in order to maximize participation among DOE HQ and Field staff, the rotation of meetings among operations offices is planned.

DOE Environmental Issues Bulletin

Publication of the DOE Environmental Issues Bulletin will continue. In addition to highlighting specific DOE facilities, up-to-date information on the Department's downsizing initiatives will be prominently featured. The states will also have the opportunity to voice their concerns/opinions through this vehicle which has a circulation of over 300. It is anticipated that this circulation will increase substantially with the availability of this document on the Environment, Safety and Health homepage which can be accessed through the Internet.

Resource Conservation Recovery Act (RCRA)/Atomic Energy Act (AEA) Issues

One issue identified by NAAG and DOE as deserving of joint discussions is the state's request that DOE develop a policy on providing radionuclide information to states upon request. To this end, NAAG provided DOE with a

survey of eight states on their experiences with regard to collecting radionuclide information from DOE. The survey revealed that in a few cases DOE resisted providing radionuclide information to states because the information was classified or DOE did not feel that the states had the authority to request such information.

After discussions on this issue at the Richland workgroup meeting, DOE agreed to review and discuss specific policy language proposed by NAAG and the states. Earlier this year, DOE provided NAAG with a counter proposal in the form of a draft memorandum setting forth principles to be used by DOE employees in voluntarily providing states with requested radionuclide information. The focus of the draft is on voluntary provision of radionuclide information and not issues related to whether the states have the right to require DOE to provide such information. Under the 1995-96 cooperative agreement with NAAG, joint discussions on these issues will continue.

CONCLUSION

As federal agencies have instituted massive environmental restoration programs designed to investigate and remediate contamination at their facilities, there has been a growing desire by those affected by these cleanup efforts (who are referred to as affected stakeholders) to have a greater role in the cleanup and decision-making process. (3) The NAAG members do not represent the typical stakeholder interested in cleanup processes and procedures, rather, the Attorneys General are interested in legal interpretations of federal and state environmental regulations, precedent setting legal opinions, and states' rights to enforce against the federal government and the rights of states to enforce their laws at DOE facilities.

With the goals to identify alternatives to litigation, educate and exchange information while improving compliance, this model of stakeholder involvement has served to open discussions on a variety of legal and procedural issues and opinions. Although consensus is not always reached, by providing a forum for these unique stakeholders to interact with the Department's policy makers and legal staff a successful model has been developed for future interaction with federal and municipal government legal officers.

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INTERACTIVE ENVIRONMENTAL BROADCAST DISTANCE LEARNING FOR INDUSTRIES, GOVERNMENT AND UNIVERSITIES

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ABSTRACT

Budget cuts and increasing demands for a technically competent workforce are problems facing industry and government today. Training requirements must be met in a cost-effective manner. The videoconference format is a proven method that saves money on training and travel costs. The videotapes produced from training programs extend the productivity in any facility. However, designing a training series that meets the needs of a diverse audience composed of scientists, engineers, and management from government, industry, and universities is extremely challenging.

National Environmental Technology Network (NETN), a program of The University of New Mexico's School of Engineering, has a proven track record in developing and producing effective videoconference training programs for industry, government, national laboratories, and universities. To date, nine successful series have been completed: Total Quality Management (two series); Hazardous Waste Management; Waste Minimization and Pollution Prevention; Environmental Risk Management; Radioactive Waste Management; and Mixed Waste Management. The two newest series are Decommissioning and Pollution Prevention. National awards from the United States Distance Learning and the United States Environmental Protection Agency attest to NETN's leadership in providing advanced technical training using distance learning techniques.

INTRODUCTION

Professionals must continue their education throughout their careers to stay abreast of latest technologies. For example, to keep up with developments in their field, engineers require retraining on an average of every four years. The need varies, depending on the precise discipline. Additionally, many technical professionals currently face significant changes in their careers, as their positions shift from defense-related activities to environmental fields.

With budget cuts and increasing demands for a technically competent workforce confronting industry and government today, the emerging training requirements must be met cost effectively. The videoconference format represents a proven method for communicating the most up-to-date information concerning technical, environmental, and management topics. According to Daniel Barron, distance education is a means of "taking quality education to the people who need and want it. Distance education is to instruction what bookmobiles and branch libraries are to reading."(1)

Through videoconferencing, thousands of professionals can receive the most current information and participate in training programs simultaneously, via satellite, without incurring the cost of travel. Each series is structured to stimulate creative interaction and promote a significant exchange of information among technical professionals and stakeholders confronting the increasingly complex environmental challenges of today's world.

Broadcast on C-band, Ku-band, and through the National Technological University (NTU), these videoconference series can be received anywhere in the United States. Most organizations and facilities already have the necessary equipment for receiving these broadcasts. Those not having this capacity can purchase a satellite dish for a minimum investment, a one-time cost for equipment.

Training program videotapes and manuals extend the capacity for training in any facility. Those who participate in the live training sessions have an opportunity to review the material and clarify important points. The programs offer every participant the opportunity to interact through

question/answer sessions, via a toll-free phone number, e-mail, or the use of a fax machine. Questions are accepted before, during, and after the programs. Those who cannot attend the live broadcasts can watch the videotapes and complete the training as their schedules permit. Designing a training series that meets the needs of a diverse audience of scientists, engineers, and managers from government, industry, and universities is extremely challenging. As Reza Azarmsa explains, "Much like any other area of mediated communication, teleconferencing is both a science and an art."(2)

NETN, an international program of the University of New Mexico's School of Engineering, has an exceptional record for developing and producing effective videoconference training programs for industry, government, national laboratories, and universities. NETN is part of an environmental consortium funded by the Department of Energy (DOE). The Waste-management Education and Research Consortium (WERC) is composed of The University of New Mexico, New Mexico State University, New Mexico Institute of Mining and Technology, Sandia National Laboratories, Los Alamos National Laboratory, and Navajo Community College. Members have teamed together to help solve the environmental problems facing our nation, collaborating in three areas: research, education, and training.

Distance Learning at NETN

To date, NETN has produced the following nine successful videoconference series: Total Quality Management (two series, one for service organizations and one for manufacturing); Hazardous Waste Management; Waste Minimization and Pollution Prevention; Environmental Risk Management; Radioactive Waste Management; Mixed Waste Management; and the two newest produced during 1995. The first, Decommissioning, is a six-part series addressing decommissioning within the U.S. Department of Energy's Environmental Management Program. The second series, Pollution Prevention, examines technologies for applications from government installations to cottage industries.

The live, interactive programs produced by NETN have been broadcast to over 167 sites with diverse audiences. The training has spanned the United States, reaching 8,000 participants in 48 states at 100 industry sites, 64 military and government installations, and 32 universities. National awards from the United States Distance Learning Association (USDLA) and the U. S. Environmental Protection Agency attest to NETN's leadership in providing advanced technical training using distance learning techniques. These series have received awards for the past three years, and in 1993, NETN earned an award for "Most Outstanding Distance Education Network" from the USDLA.

How to Produce a NETN Television Series

The process described below is accomplished over a 9-12-month period. To select a topic for distance learning series, NETN surveys the various receive sites, requesting information about their current training needs. When sufficient interest is expressed in one particular area, and the funding for production secured, the topic becomes the focus of the next distance learning series.

Step 1: Experts Meeting

In the early stages of developing a series, NETN conducts an "Environmental Experts" workshop, where environmental professionals meet to listen to speakers and participate in panel discussions and question/answer periods. To simplify information exchange, workshop participants complete surveys for NETN. These surveys ask respondents to describe the

issues they consider critical for professionals in these fields to address; list the concepts that need to be communicated to stakeholders; identify recognized national experts in the fields; and provide names of periodicals, national organizations, and associations important for people involved in the particular area of expertise.

Step 2: Advisory Board Meetings

The next step involves of advisory board meetings, which lay the foundation for the development of the series. National experts involved in previous programs, scientists from the institutions that comprise WERC, and stakeholders are all invited to attend. To provide an overview of a series' production, speakers with past experience in televised training describe the roles and responsibilities of those involved in the process.

Step 3: Identification of Nationally Recognized Program Leaders

As the topics develop, the technical advisors and the advisory board suggest possible program leadersexperts in the field with backgrounds in industry, business, and government. NETN contacts these individuals and those who agree to join the advisory board and NETN staff to form the nucleus of the evolving series.

During the next portion of the advisory board meeting, members begin the planning process by developing a proposed series outline. The outline consists of specific program topics, any particular emphasis to be considered for each topic, and possible case studies.

Step 4: Identification of Presenters from Diverse Backgrounds

The program leaders then select about five people with expertise in their program topics to serve as presenters. The technical advisors understand the overall program content and make sure all aspects of appropriate topics are covered. They also work with the NETN staff and may provide a final review of the programs' instructional manuals.

The individual program leaders recruit nationally recognized experts in the field for their program. They maintain a balance in the program content and sometimes also serve as presenters. The program presenters need good communication skills to deliver their material and to interact with the participants during question/answer periods. Presenters are encouraged to incorporate graphics, videos, humor, and real-life examples to keep their presentations interesting. Each presenter submits charts, suggested readings, test questions, glossary of terms, and a brief biography to NETN, all of which is incorporated in the manuals.

Step 5: Production/Conference Calls

Because NETN seeks presenters from all over the United States, much of the program design occurs during a series of conference calls. Once a plan is established for the individual programs, the program leaders meet with the NETN staff in Albuquerque to define the content of each program and assure continuity for the series as a whole. This not only reduces redundancy, it also confirms that the participants will receive an adequate amount of technical background information. Then, as the scheduled broadcast dates approach, conference calls among those involved in individual programs continue as needed.

Step 6: Instructional Manual Preparation

Meanwhile, the presenters' materials are prepared for the instructional manuals and returned as final drafts to the presenters and the technical advisors for review. The manuals for each program contain the following information: program description, presenters' charts, suggested readings, test questions, presenters' biographies, and various NETN feedback forms.

A reproducible master of each program manual is sent to each receive site prior to the program broadcast. The master is duplicated at the site, and copies are distributed to all participants attending the training. One bound copy of the manual is sent to the site to be used as reference and to augment the site's videotapes of the programs.

Simultaneously, the video producer pulls together all the elements necessary for live broadcast. Scripts are written and case studies are chosen. Arrangements for on-location recording at the sites are made; the case studies are videotaped and edited for the broadcasts. All elements for live broadcast are orchestrated, down to the selection of soundtrack music.

Step 7: Training Facilitators Across the United States (diverse audience)

Prior to the first program's broadcast, NETN conducts a facilitator training program. Experience has shown that site facilitators contribute to the overall success of this televised training when they function as proactive members of the team. The facilitator training program covers a wide range of instructional material. Facilitators learn technical details, such as how to establish a comfortable environment for the viewers and how to check the operation of the equipment. They learn how to find the appropriate audience and publicize the upcoming training at their site. They receive tips about effective ways to interact with series advisors and are encouraged to seek experts within their organization who can enhance the learning experience.

In addition to the televised training, facilitators receive a training packet to serve as a reference during the course of the series. Given this overview of what to expect from the distance education experience and how to make it meaningful for the participants, the facilitator often serves as the foundation for the partnership that develops between the university and the audience.

Step 8: Publicizing/Marketing

Publicizing the training event occurs at two levels. NETN can assist the sites with publicity by developing a customized brochure, news releases, and advertisements. A brief promotional tape is also sent to each receive site. On the broader spectrum, NETN notifies former participants of the upcoming series, writes press releases, and develops detailed brochures for distribution. To help ensure a wide array of participants, the NETN director publicizes the training at trade shows and at national meetings on distance education and environmental issues.

Step 9: Series Broadcasts

One to two weeks after the facilitator training program, the first program is broadcast on alternating Wednesdays. Facilitators are instructed that every program within a series contains a minimum of six presenters from industry, government, and universities with varied scientific backgrounds and expertise. Included in many programs is a segment on the regulations that pertain to the topic being discussed. To increase effectiveness, live presentations are mixed with video that demonstrates the technology involved in the program topic, as well as relevant case studies that illustrate practical uses of the material. Customized workshops and on-site training are also available.

Step 10: Modifying Programs to Feedback and Evaluations

Regular feedback from the participants following each program enables the production staff to constantly improve on the quality of the series. After the final program, the participants complete both NETN Feedback and

Evaluation forms. The facilitator returns these forms to NETN, where a thorough review of the evaluations is completed. This direct input from the receive sites allows the technical advisors, program leaders, presenters, and production staff to improve the quality of the series by capitalizing on what went well, and modifying the programs to make improvements where indicated.

The Keys to Success

The keys to producing effective videoconference series extend into every facet of production. In the preliminary stages, well-defined topics and carefully selected presenters with excellent credibility are essential. Comprehensive instructional manuals enable the audience to follow along easily and record their notes. Question/answer periods with ample time for discussion encourage interaction that facilitates information exchange and lasting partnerships among the sites.

Timing is also an important consideration. Experience shows that short (3 hrs) programs, regularly offered, are the most effective. NETN broadcasts these live videoconferences at regular intervals, for example, on the second and fourth Wednesdays of each month. Finally, the role of the on-site facilitator cannot be overemphasized. Well-prepared facilitators can play a proactive role in distance education, as they bridge the gap between the television studio and the audience and promote an open exchange of ideas and information.

Advanced Technical Training Through Distance Learning

As the workforce shifts from defense to environmental careers, working professionals need environmental information for current and future jobs. They require immediate, high-level training in specific environmental fields. NETN's goal is to assist the U.S. Department of Energy (DOE) in complying with the Congressional mandate of a 30-year remediation initiative.

Interactivity

Advanced technical training and technology transfer represent a proven method of disseminating valuable information to engineers, scientists, researchers, hospitals, universities, and laboratories around the world. Video conferences provide unique networking opportunities, allowing participants to exchange ideas with recognized environmental experts. During the interactive sections, participants are encouraged to ask specific questions that pertain to their particular site. Throughout the series, audience members develop the tools to change directions in their careers, with a strong emphasis on environmental clean-up and future use of a site.

Cost Effectiveness

Transmissions via satellite offer economic advantages, such as savings on travel and workshop registrations. The price per site allows unlimited attendance during the live broadcasts. To prolong the life of the information contained in these series, the sites have access to videotapes of the programs and a bound copy of each instructional manual for their library.

Certificates and Continuing Education Units

DOE currently receives NETN's environmental series in all DOE Operations offices across the US and at DOE Headquarters. This is part of the DOE effort to retrain their workforce from defense to environmental careers and to support environmental information exchange. After each series concludes, participants receive certificates of completion and earn Continuing Education Units (CEUs) through The University of New Mexico.

Endorsements

Recognition and endorsement for this educational effort have been extended to NETN by the American Society of Mechanical Engineers (ASME) and the Hazardous Waste Action Coalition (HWAC). Sponsors include DOE, the U.S. Environmental Protection Agency (EPA), and the Waste Management Education and Research Consortium (WERC).

How Evolving Partnerships Enhance the Training

NETN achieves balance and diversity by seeking presenters from industry, government, and universities. Working closely with experts from such varied backgrounds has enabled NETN to develop strong partnerships among the presenters and the receive sites.

The ongoing interactive programs promote mutual awareness and understanding. Unique partnerships can evolve through a distance education experience. For example, Sandia National Laboratories and Los Alamos National Laboratory subscribed to NETN's first series on Total Quality Management in 1990. Both organizations have participated in every subsequent series, and a mutually beneficial understanding has developed between The University of New Mexico and these national laboratories. Because individuals feel free to ask site-specific questions, participants throughout the nation come to understand each other better, discovering common ground in the search for common solutions.

Technical Aspects of the NETN Series

As mentioned earlier, the technical aspects of a videoconference seriesprogram content and specific emphasisare determined through the input from viewers of previous series, advisory board meetings, and an Environmental Experts workshop.

The Total Quality Management (TQM) series were NETN's first two distance education initiatives pertained to Total Quality Management (TQM). A14-part series offered an engineering/management approach to TQM, and a 6-part series focused on TQM as it applies to service organizations.

Hazardous Waste Management, the next series, consisted of the following programs: 1)Introduction: What Is Waste?; 2) Risks Associated with Hazardous and Radioactive Wastes; 3) Transport Processes Related to Wastes; 4)Waste Form Modification; 5) Site Characterization; 6) Sampling and Analysis; 7) Soil and Groundwater Remediation I Physical/Chemical Processes; 8) Soil and Groundwater Remediation II: Biological Processes; 9) Radiation and Radioactive Materials; 10)Radioactive and Mixed Wastes Management; and 11)Waste Minimization and Series Close.

Waste Minimization and Pollution Prevention, adhering to the logical progression of the content, NETN followed the previous series with an 8-part videoconference: 1) OverviewWhy Minimize Waste?; 2) Where Do We Start Waste Minimization?; 3) How Does Recycling/Reuse/ Reclamation Make Economic Sense?; 4) Are the Right Product/Process Designs Being Addressed?; 5)Solvents and Organic Chemicals; 6) How to Implement Minimization in Metals, Plating, and Electrical Interconnects; 7) How to Minimize "End of Life" Problems; and 8) Planning and Preparing for the Future.

Environmental Risk Management sought to bridge the gap between technical issues and social concerns. Program titles included 1) Risk: Terminology, Concepts, Methods, Applications and Why Risk Communication Is Difficult; 2) How Do We Decide What Is Risky?; 3) Identifying the Gap: Divergent Technical and Social Methods for Risk Characterization; 4) Quantification of Models for Risk Assessment; 5)Risk Characterization:

Synthesis and Communication; 6) Risk Assessment: Communicating Results and Public Perception; and 7) Risk-Based Decision Making: A Final Act?

Radioactive Waste Management was produced concurrently with the Environmental Risk Management series. These programs were 1) An Introduction to Radioactive Waste Management; 2)Interactions Between Radiation and Matter; 3)Decommissioning and Decontamination; 4)Transportation Issues in Radioactive Waste Management and UMTRA; 5) Low-Level Radioactive Waste Management; 6) High-Level Radioactive Waste; 7)Transuranic Radioactive Waste Management; and 8) New and Other Technologies for Radioactive Waste Management.

Mixed Waste Management was broadcast in 1994. Generated by government facilities, research laboratories, hospitals and universities, mixed waste is a major concern in remediation projects, including the cleanup activities on the U.S. DOE weapons complex and old radioactive waste disposal sites. The series consisted of 1) Generation and Characterization of Mixed Waste; 2) Storage of Mixed Waste; 3) Minimization and Treatment of Mixed Waste; 4) Technical Development of Mixed Waste Issues; and 5)Remediation Activities.

Decommissioning, a new 6-part series, was broadcast in 1995, with an introduction to the series with a goal and vision statement from U.S. DOE. The inventory and history of the major decommissioning sites with overviews and the various phases of the decommissioning process are included. Each program consists of technical presentations by experts focusing on decommissioning terminology, problems involving contamination and radiation along with successful approaches, and existing and evolving decommissioning techniques. This series stresses the importance of good planning, pre-job briefings, and dress rehearsals with mock-ups to uncover problems and save time. Participants learn the techniques for dismantling and segmenting equipment and structural members and demolishing structures.

Pollution Prevention, a 5-part series, was also aired in 1995. Pollution prevention is far more powerful, more painless, less costly than continually degrading the environment by just "managing waste." Only through massive educational and awareness programs can the attitudes and values of the people be made sensitive to the need for pollution prevention. It is everyone's responsibility and moral obligation to protect the environment for future generations.

The end-of the pipe approach to waste management is a losing strategy. Striving to be in environmental compliance is not the answer; and managing waste is not only expensive, but poisonous to the environment and to the health of the public. Pollution prevention the front-end approachmakes business sense and is less destructive to the environment.

CONCLUSION

There is a growing need for information about environmental issues throughout the world. Delayed broadcasts and taped programs have been received by ISTEC (a Latin American Network) and tapes used as far away as Estonia.

Communication and participation are critical elements in our national efforts to identify and explore environmental challenges. NETN develops distance education programs that explore technologies for the ultimate protection and preservation of the environment, while sustaining economic growth. We are proud of our role in building partnerships between the public and private sectors in working toward a sustainable global future for the environment.

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Session 35 -- INNOVATIVE RESTORATION APPROACHES
Co-chair: Don Ofte, FERMCO
35-1
LESSONS FROM THE PRIVATE SECTOR ON PERFORMANCE-BASED MANAGEMENT

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ABSTRACT
Implementation of the Government Performance and Results Act of 1993 (GPRA) has provided a unique challenge for Federal Agencies, such as the Department of Energy (DOE) Office of Waste Management (OWM). While performance measurement, as required by GPRA, is new to Federal Agencies, private industry has applied it at all organizational levels to better manage their operations for some time. There has been significant discussion about how the private sector uses performance measures, but there have been very few empirical studies systematically examining their use.
To gather information on comparable private industry practices, waste management industry firms were surveyed through questionnaires and follow-on interviews. Questionnaires were sent to 75 waste management firms throughout the United States and Canada. Twenty-four percent of the firms responded to the questionnaire and participated in the follow-on interviews. The questionnaires were typically completed by vice-presidents or senior financial officers. Information collected from the questionnaire and follow-on interviews provided valuable insight into industry practices in the area of performance measurement. This paper discusses the study results and how they can be incorporated in the DOE OWM performance measures and influence the character of the "critical few" metrics used by senior DOE managers.

INTRODUCTION

The Government Performance and Results Act of 1993 directed Federal Agencies to develop annual performance measurement plans for their program activities. Private industry successfully uses performance measurement at all organizational levels to better manage operations. These private enterprises, particularly those in the waste management industry, are a valuable source of information for DOE OWM on current performance measurement practices. Of particular interest is the character of those "critical few" performance metrics or indicators used by corporate executives, the position which best corresponds to the operational oversight responsibilities of the DOE Deputy Assistant Secretary for Waste Management and the Assistant Secretary for Environmental Management.

To gather information on private industry practices, waste management firms were surveyed, first through the distribution of a survey questionnaire, and then with follow-on discussions after the questionnaires had been returned and reviewed. The survey was sent to 75 waste management firms throughout the United States and Canada. Information collected from the survey and follow-on discussions provide valuable insight into industry practices in the area of performance measurement.

The responding firms represent a cross section of the waste management industry. The core businesses of over 50% of the respondents is waste disposal and/or waste treatment, though the wastes which they manage pose less handling problems than those faced by the DOE. Of the firms which responded, eight (44%) have annual revenues of over $100 million and several core business lines, requiring the chief executive officer (CEO) to manage multiple performance centers, a management complexity similar to that facing the Deputy Assistant Secretary for Waste Management.

SURVEY METHODOLOGY

Although performance measurement is used across all industry sectors, those firms in the waste management business were judged to be in the best position to provide the type of performance measurement practices most useful to the DOE OWM. Candidate firms were identified using a variety of sources, and a screening process. Telephone contacts were made to identify the specific types of services the companies provided and the corporate representative who should receive the survey questionnaire. The survey form transmittal package included a cover letter summarizing the DOE efforts to increase the use of performance measurement practices. Twenty-four percent of those receiving the survey responded. As the survey questionnaires were received, data was compiled and analyzed. This information was reviewed and used during follow-on discussions with the eight largest responding firms. A profile of the participating firms is shown in Fig. 1. The follow-on discussions with the respondents, who were typically vice presidents or senior financial officers, and the corporate CEOs were worthwhile. However, it was evident that though they were interested and supportive of the study, other time demands limited their participation and accessibility.

Fig. 1

FINDINGS

The survey findings are divided into those performance metrics or indicators used by the chief corporate executive, typically the CEO, and those used by other organizational levels. Universally, the performance measures used by the CEO subsume the more detailed measures from other organizational levels.

Overall Performance Measurement Practices

Key results extracted from the questionnaire are:

- 94% of the respondents use organizational goals as the focus of performance measures.
- 78% of the respondents use performance measures to monitor organizational progress toward corporate mission and goals.
- 67% of the respondents share mission statements and goals with employees.
- Smaller firms tend to use fewer performance indicators.
- Performance measurement information is distributed to the executive level at least monthly and often weekly.

No organization used purely financial measures, such as revenue or cost per ton of waste stored or treated. While 17% of the respondents used purely nonfinancial performance measures, such as customer satisfaction or volume of market share captured, the majority used a combination of financial and nonfinancial measures. The performance measurements used at one level in the organization often helped explain the performance results at the next higher level. Examples of typical measures appear in Tables I and II.

Table I

Table II

Senior managers generally rely upon five to seven key financial and nonfinancial indicators and focus on division or product line performance. Below senior management, the variety and number of indicators increases and are predisposed toward quantitative measurements such as unit cost, resources expended, and achievement verses time unit. Middle managers tend to focus on their specific work centers. Performance measures for each work center are defined and collected for consolidation into divisional or product measures and for comparison with similar corporate operations. Such a systematic approach was the basis for the Fiscal Year 1996 OWM performance measures for waste treatment, storage, and disposal.

While the OWM does not generate revenue or operate for profit, the underlying principle of producing quality products for the least possible cost is consistent in both sectors. The respondents use performance measures for monitoring, evaluating, and controlling costs. The performance metrics used for financial measures were similar among the firms surveyed. The key condition for optimum use of performance measurement indicators, which seemed to be taken for granted by the respondents, were effective communications and cost collection, and a quantity tracking system which provided relevant, timely, accurate, and consistent information. Unique financial, cost estimating and collection systems across the DOE complex could make consistent performance data collection difficult. Standard definitions for direct labor, material, and indirect cost pools are necessary for credible complex-wide measures for waste management operations.

Qualitative measures, while fewer than quantitative measures, did also exist at all management levels and tended to focus on customer satisfaction. This is akin to stakeholder interest measures for DOE. Because of the diverse DOE stakeholder population, measurement focus should start with an objective area such as satisfying regulatory issues and as the system matures, proceed toward the more esoteric stakeholder issues. Stakeholder-related measures are important but will require OWM and stakeholder cooperation for mutual acceptance.

Corporate Oversight

Specific chief executive performance measurement practices were solicited through follow-on discussions from the eight largest respondents. This approach validated the questionnaire input and provided the opportunity to discuss how performance measures were used. These discussions were used to validate methodology for developing "critical few" measures for DOE management. The title "critical few" indicates the awareness by senior DOE management that they could not evaluate each performance measure for every EM office, a fact supported by the CEO interviews. The discussions provided significant insight and focus for the OWM "critical few."

While each firm uses a variety of indicators or metrics to monitor performance within the organization, at the highest corporate level, typically the CEO or president, financial/business indicators were primarily and universally used to measure performance. These included corporate earnings, earnings per share, return on investment or assets, total revenue, and total cost. This is the type of information included in annual reports issued by the company and is targeted toward stockholders.

Financial/business performance metrics, in the form of tables, charts, or curves were typically provided monthly. This provided actual versus planned performance through a specified period and occasionally included performance projections. These metrics conveyed "what" had occurred but did not typically provide much analysis or detail. Supplemental information on operational performance at the product or service level was available, however, and provided insight into trends. Consequently, corporate executives could look down into their organizations and pinpoint those areas or operations which had contributed to the successes or problems.

There were no reported instances where aggregating or indexing of dissimilar performance metrics, produced through sophisticated mathematical modeling, was used by the CEO. In one case, the aggregated performance of similar activities, landfill operations, were provided to the CEO as a key performance measure. The differences in the waste handled and landfill sites were normalized, using weighting factors. Variance from the corporate planned cost would be explained by examining the actual volumes handled by each site as compared to the planned volumes for each site.

At several companies, along with key financial performance indicators, the CEO received performance metrics which evaluated customer satisfaction such as: on time deliveries, work in progress, milestones achieved, and other performance measures which they considered significant when evaluating overall performance. The selection of metrics appeared to reflect the overall size of the company, the variety of products and services it provided, and the management style of the CEO.

The CEO feedback supported the position that no single metric could be developed to measure the performance of the DOE Waste Management Program. A single metric would have been especially difficult in the case of the Waste Management Program anyway, because managers must deal with various types of existing waste and an indeterminate amount of waste generated by ongoing operations and decommissioning of closed facilities. Several metrics would be required to measure performance in key program areas. To mirror its private sector counterparts, the DOE Waste Management "critical few" should be composed of metrics measuring production, cost of operations, and customer satisfaction. A program analysis by staff and management should be conducted to determine what measures, reported at the operations level, meet these key program parameters.

CONCLUSION

Private waste management firms generally use performance measures to monitor their operations. Performance metrics used within the organizational levels complement those received by the chief executive by identifying areas which have contributed to overall corporate performance. Performance indicators recognized as significant by senior and executive management generally ranged between five and seven. This suggests that too many performance indicators are either not manageable

and/or useful. From the study results, chief executives use two to three key performance metrics. These high-level performance indicators most often measure financial/business performance. Below the executive level, managers use a combination of financial and nonfinancial indicators which relate to a specific product, process, service, or organizational unit. The objective of this study was to examine private sector performance measurement practices and to identify potential areas where the DOE OWM could benefit. As the DOE moves toward implementing GPRA, the practical lessons from the private sector are helpful in designing the performance measurement system. Key lessons learned from this study are:

Focus on a small number of meaningful measures.

Develop a system where more detailed information is available, if requested.

Establish an effective process for defining and communicating the performance measures within the organization and for relating measures to the overall organizational goals.

Implement a cost collection and quantity tracking system that provides relevant, timely, accurate, and consistent information.

While the DOE neither generates revenue nor is operated for profit, performance measures can monitor operations execution and efficiency. The cornerstone to these measures is a consistent system for measuring output and collecting costs. The ability of the existing systems to meet the criteria of consistency will have to be evaluated as part of the performance measure development effort. Additionally, the "critical few" program parameters must be clearly defined, and they should contain production, cost of operations, and customer satisfaction measures. Additional interactions with private sector waste management firms, both at the headquarters level and in the field, can be beneficial as the DOE continues to improve the overall effectiveness of waste management operations.

35-2

OPTIMAL PROJECT SELECTION TO MINIMIZE LIFE CYCLE COSTS

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ABSTRACT

Sound economic methods are required to selectively fund projects that address significant environmental restoration, deactivation, decontamination and decommissioning, surveillance and maintenance, and waste management challenges. The funded projects must offer the greatest environmental and health risk reduction, significantly reduce sources or inventories, generate the highest public stakeholder acceptance, or ensure that worker safety is maximized for the least cost.

This paper presents a proven approach that identifies the optimal selection of projects so that the total Life Cycle Cost (LCC) is minimized for all selected projects. Annual budget constraints, risk reduction requirements, and multiple project combinations are considered. Scenarios based on annual funding levels, desired relative risk reduction, and project cost factors are examined. The approach was developed and is currently being implemented at Oak Ridge National

Laboratory (ORNL) Environmental Restoration Program Characterization and Remedial Decisions Office.

INTRODUCTION

Environmental restoration and waste management (ER/WM) program managers need to choose the optimal number of projects to fund during a given Fiscal Year (FY). The total Life Cycle Cost (LCC), i.e., the sum of the annualized costs of all the funded projects, must be minimized.

At the same time, ER/WM managers must deal with other major constraints. Projects must be selected that maximize the reduction in environmental and health risk. Budgets cannot be exceeded within the specified planning horizon. Some projects, or a combination of projects, must be funded for a variety of programmatic, regulatory, public stakeholder or technical reasons. In some cases, if one project is funded, another project cannot be funded for similar reasons.

STATEMENT OF THE PROBLEM

The critical challenge facing ER/WM managers is:

"It is a fact of life that we have limited dollars. What projects should be funded that demonstrate success?"

The problem statement is:

What is the best selection of $K < N$ projects that minimizes total LCC, meets or exceeds a specified level of risk reduction, satisfies available budget and LCC constraints, and meets project combination constraints within a given planning horizon?

The term "risk reduction" corresponds to the decrease of human health and environmental risk. Risk is the probability an individual will develop cancer (1). Inventory or source-term reduction, stewardship, public acceptance, or worker safety could be substituted for risk reduction in the problem statement.

An alternative way to express the problem statement is:

What is the best selection of $K < N$ projects that maximizes risk reduction, is less than a specified total LCC, meets available budget constraints, and satisfies project combination constraints within a given planning horizon?

This is a rearrangement of the objective and the primary constraint of former problem statement. The objective is to maximize risk reduction subject to the primary constraint of minimizing total LCC. The constraints for available budget and project combinations within the planning horizon remain the same. The solution to both ways of expressing the problem is the same. (2)

REVIEW OF CURRENT APPROACHES

Three approaches are currently used for ER/WM project selection. The benefits and the limitations of each approach are presented in Table I.

The purely economic approach assumes cost is the only evaluation criteria. Although this approach is simple, it is rarely used in a standalone manner. The total current year cost and the LCC for all candidate projects are determined. A funding line is defined. Projects are rank-ordered from minimum LCC to maximum LCC. Projects are selected based upon minimum LCC until the funding line is reached for the current year costs. All projects that fall below the funding line are discarded until the next planning horizon's budget exercise.

Multiple Criteria Decision Making (MCDM) uses a quantitative metric to prioritize the projects independent of cost and then applies the purely economic approach. One common metric is a weighted linear additive function. Weights are assigned to decision variables such as public

acceptance, risk reduction, worker safety, or regulatory compliance. Each project is "scored" for these variables, and the overall weighted score is computed. Projects are prioritized by this overall score. The economic approach is then used to select projects for funding.

Effectiveness-Cost Analysis prioritizes projects using an effectiveness-cost (E/C) ratio. The effectiveness measure may be expressed in dollars, or it may be an MCDM-like quantitative metric. Cost is LCC or cumulative annualized outlays for some planning horizon. Projects with the highest E/C ratio are selected for funding until the funding line is reached. The remaining projects are not funded.

Unfortunately, project combinations are examined after the first cut of project selection is accomplished. Several iterations in the project selection process are required, and these are usually accomplished in a heuristic, brute-force, manner.

There are obvious benefits of each approach. There are also significant limitations. The greatest limitation is that a quantitative metric is always required for the MCDM approach and Effectiveness-Cost analysis. The project scores and the weights of the decision variables must be calibrated and repeatable. If this is not the case, the project selection credibility is questionable regardless of how reliable or repeatable the subsequent application of the pure economic approach.

Table I

THE OPTIMAL PROJECT SELECTION APPROACH

The optimal project selection approach employs an operations research principle known as mathematical programming. Mathematical programming is a method that deals with the allocation of resources among competing activities (2). This type of approach is useful for a variety of ER/WM problems including identification of transportation routes for waste shipments to disposal facilities, personnel scheduling, and, as the topic of this paper, project selection. It is assumed that any project being considered for funding meets some minimal acceptable level of stakeholder acceptance and worker safety.

The optimal project selection approach is summarized in Table II. The problem statement is formulated as an integer mathematical programming problem. All results are deterministic, i.e., the formulation cannot handle random variation in the objective function coefficients or the constraints. The decision variables are binary, namely, projects are either selected or not selected. The objective function is linear, and the coefficients of the decision variables are the total LCC for each project. There are several of constraints: 1) all decision variables are either 0 or 1, 2) a linear combination of projects and associated risk reduction is required to be greater than some threshold value, 3) a linear combination of projects and planned budget for each project for one or more separate FY is required to be less than a specified value, and 4) a series of logical rules associated with project combinations must be met. The logical rules associated with the project combination constraints is presented in Table III. The computational approach is implemented using Microsoft Excel, Version 5.0a.

Table II

Table III

The details for optimal project selection are presented in Table IV. There are six major actions that must be performed:

- Describe projects in terms of cost and risk reduction
- Describe the planning horizon and associated budgets

Identify scenarios in terms of budgets, desired risk reduction, and allowable project combinations

Identify optimal projects in terms of projects than minimize total discounted LCC and meet specified budgets, desired risk reduction, and allowable project combinations constraints

Evaluate optimal projects in terms of rank-ordered scenarios

Select optimal projects based upon the results of the evaluation and associated sensitive analyses and scenario evaluations.

The optimal project selection approach does not include management challenges such as changes in mission, client or ER/WM re-organization, market share loss or gain, or crises and emergencies.

Table IV

Candidate projects are first described in terms of cost and risk reduction. Cost is expressed as both a discounted and an undiscounted LCC. For example, if the undiscounted LCC cost for a project is $25M, the discounted LCC at an 8% discount rate over 15 years may be $21.5M. Risk reduction is expressed as a percentage of total risk reduction that would occur if the project were successfully implemented based upon the project's geographic domain. For example, the current human health and environmental risk for a project may be 1.0E-03. If the project were successfully completed, the risk would be less than 1.0E-04 that would represent a linear risk reduction of 90%. (As discussed earlier, inventory or source-term reduction, stewardship, etc. could be used instead of human health and environmental risk).

The planning horizon and budgets are then described. The planning horizon is defined to be those periods (in years) for which the optimization is applicable. Up to three planning horizons may be examined. The optimization approach requires one planning horizon to be the maximum duration for all projects. The first FY is strongly recommended as the second planning horizon, and it is suggested the third planning horizon may be either two, three, or five years. The budget for each project is determined from the undiscounted LCC. The total budget for all projects is the sum of the undiscounted LCCs for all projects. Annual budgets for the projects for FY (the FY1 and FY2, FY3, or FY5) are expressed in current year dollars.

Scenarios are developed to compare optimal project selections. There are no strict rules for scenario identification other than scenarios for the optimization can only be expressed in terms of the decision variables of budget, risk reduction, and project combinations. For example, a reasonable Baseline scenario is: maximum funding levels and budgets are available, risk reduction is the highest, and project combinations represent current planning thought. A Worst Case scenario can be developed in which expected funding levels and budgets are minimal and risk reduction is very low. The worst case project combinations explicitly depend on user perceptions. Currently, the model can examine as many scenarios as desired.

The optimal project identification is determined using data from the previous actions. Using mathematical integer programming, optimal projects are identified for the Baseline and Alternate scenarios. All projects identified result in the minimum total discounted LCC and meet risk reduction, budget, and project combination constraints.

Optimal projects are evaluated in terms of the rank-ordered scenarios. This is because a sensitivity analyses of the constraints is not meaningful since project selection has been formulated as an integer

programming problem. The rank ordering of the scenarios may be based upon 1) likelihood (i.e., probability that a scenario will occur), 2) utility (i.e., the relative worth of the scenario to program challenges), or 3) whatever ordering makes sense to decision-makers.

Optimal projects are selected based upon the results of the project evaluation. The optimal project identification and evaluation provides a recommendation for funding a selected set of projects based on cost and some measure of project effectiveness; however, there are many management challenges that are not included, nor attempted, in the optimal project selection approach. As such, the final selection always rests in the hands of the decision-maker.

ILLUSTRATION OF APPROACH

An example that is representative of project selection activities at ORNL is presented in this section, Table V. Ten projects are candidates for funding. Each project is an Environmental Restoration (ER) project, a Decontamination and Decommissioning (D&D) project, or a Surveillance and Maintenance (S&M) project. Project duration ranges from one to 15 years. The undiscounted LCC of each project is $25M. Risk reduction for each project ranges from 1% to 100%, where large values of risk reduction are preferred to small values.

Project duration ranges from one to 15 years, hence this is the maximum planning horizon. Two other planning horizons are considered, FY1 and FY2. The discounted LCC is the total annualized Net Present Values of capital and operations and maintenance costs using an 8% discount rate. Budgeted current year costs for each project are identified for FY1, FY2, and FY3.

Table V

Seven scenarios are developed to compare optimal project selections. These are presented in Table VI. As the shading of the entries in table darkens, the scenarios get progressively worse. The scenarios are subjectively rank-ordered by likelihood from most likely to least likely. No numerical likelihood estimates are used to create the rank-ordering. The Baseline Scenario considers the maximum funding level of $250M over 15 years with full funding for FY1 and FY2. Projects 2, 9, and 10 must be funded. This is denoted as "2, 9, 10" in Table VI. Disjunctive project constraints are 1) Project 4 or Project 5 can be funded, but not both, and 2) Project 7 or Project 8 can be funded, but not both. This is denoted as "(4,5) (7,8)" in Table VI. No conjunctive constraints are considered in any scenario. Risk reduction is required to be at least 75%.

Scenario 1A is the same as the Baseline Scenario except that (1) the maximum funding level is reduced to $200M, and (2) there are no disjunctive project constraints. Scenario 1B is the same as the Scenario 1A except that 1) risk reduction is required to be at least 70%, and 2) the disjunctive project constraints "(4,5) (7,8)" are present.

Scenario 2A represents 1) the maximum funding level is reduced to $200M, 2) FY1 funding is cut by 10%, 3) FY2 funding is cut by 20%, 4) Risk Reduction is required to be at least 65%, and (4) there are no disjunctive project constraints. Scenario 2B is the same as Scenario 2A except that the maximum funding level is reduced to $150M.

Worst Case Scenario 1 represents 1) the maximum funding level is reduced to $150M, 2) FY1 funding is cut by 10%, 3) FY2 funding is cut by 20%, (4) risk reduction is required to be at least 60%, and 4) the disjunctive project constraints "(4,5) (7,8)" are present.

Worst Case Scenario 2 represents 1) the maximum funding level is reduced to $150M, 2) FY1 funding is cut by 20%, 3) FY2 funding is cut by 20%, 4) risk reduction is required to be at least 60%, and 5) the disjunctive project constraints are not present.

Table VI

The optimal project identification is presented in Table VII for each scenario. Several observations are provided:

Although the Baseline Scenario considers maximum funding level of $250M over 15 years with full funding for FY1 and FY2, risk reduction cannot exceed 77%. This is due to the presence of the disjunctive project constraints.

All projects identified result in the minimum total discounted LCC. The discounted LCC ranges from $142M when the full $250 funding is available to $121M when only $150M is available.

All projects meet risk reduction constraints. The maximum risk reduction possible is 77%. When funding is at the worst case level, risk reduction ranges from 61% to 65%.

All projects meet FY1 and FY2 budget constraints for full funding availability. When the full $106M is available for FY1, budget requirements range from $76M to $92M. When the full $32M is available for FY2, budget requirements range from $18M to $28M.

All projects meet FY1 and FY2 budget constraints for reduced funding availability. When $90M is available for FY1, budget requirements range from $87M to $90M. When $80M is available for FY1, budget requirements are $71M. When $25M is available for FY2, budget requirements range from $14M to $22M.

All projects meet project combination requirements. Although there are only a few project combination requirements, most problems may contain several hundred such constraints.

Table VII

Optimal projects are evaluated using the rank-ordered scenarios. This is illustrated in Table VIII. Risk reduction is as presented earlier. Slack represents the difference between the constraint requirement and the result of the scenario optimization. As the shading of the entries in table darkens, the outcomes of the scenarios get progressively worse. Several observations are made:

The Baseline scenario seems reasonable until FY2 slack is examined, i.e., $8M may be too little slack for the ER/WM program.

Scenario 1A offers greater slack ($14M) for FY2 than the Baseline ($8M). Scenario 1B offers twice the slack for FY1 ($30M) than the Baseline ($14M). There appears minimal difference in risk reduction between the Baseline and Scenarios 1A and 1B.

Although both the Worst Case scenarios look bad, the benefit of some slack in FY1 and FY2 may offer some hope to the ER/WM manager.

Table VIII

Optimal project selection is made based upon the evaluation results. The Baseline and Scenarios 1A virtually equivalent. Scenario 1B offers slack (read "insurance") in FY1 but a gamble for slack in FY2. Scenarios 2A and 2B are to be avoided. The worst case scenarios offer fiscal challenges to the decision-maker, but, like Scenarios 1A and 1B, there is some slack present that could be judiciously managed. As stated earlier, there are other management challenges not included in the optimal project selection approach.

CONCLUSIONS

A proven approach to identify projects combinations that minimize total LCC and meet annual budget constraints, risk reduction requirements, and multiple project combinations has been presented. Scenarios based on annual funding levels, desired relative risk reduction, and project cost factors have been examined. Optimal projects to be selected have been evaluated and recommended.

ACKNOWLEDGMENTS

The mathematical optimization approach for project selection was developed for the Oak Ridge National Laboratory (ORNL) Environmental Restoration Program Characterization and Remedial Decisions Office. The approach supports ER/WM, D&D, and S&M project identification, evaluation, and selection.

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PROGRAM MANAGEMENT: THE KEYS TO A SUCCESSFUL ERWM CONTRACT

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ABSTRACT

This paper identifies the many opportunities presented in the planning and initial implementation of a large Environmental Restoration and Waste Management (ERWM) task order contract. The U.S. Department of Energy-Oak Ridge Operations (DOE-ORO) ERWM Remedial Design Contract with Foster Wheeler Environmental Corporation is used as a model for exploring these topics. The program has been highly successful and has obtained from DOE outstanding marks for achieving quality, responsiveness, timeliness, and subcontracting goals. Potential traps are identified in this paper so that they can be recognized and avoided or mitigated. Teaming and subcontracting are also addressed.

INTRODUCTION

Program management for a large Environmental Restoration and Waste Management (ERWM) task order contract is both a skill and an art. Unlike project management, program management deals with global and political issues, with both client and home organizations, as well as with day-to-day operations. Program management requires up-front planning and nurturing, for no contract matures successfully by itself. This paper identifies the many opportunities presented in the planning and initial implementation of the contract. Potential traps are identified so that they can be recognized and avoided or mitigated. Teaming and subcontracting are also addressed. The authors rely on years of program management experience to explore such questions as the following: Can you have an integrated team? What needs to be done before you sign your contract? Do you know who your client(s) is(are)? Have you incorporated the relevant, especially any new, procurement strategies?

The U.S. Department of Energy-Oak Ridge Operations (DOE-ORO) ERWM Remedial Design Contract with Foster Wheeler Environmental Corporation is

used as a model for exploring these topics. This ERWM Programwhich focuses on the Oak Ridge National Laboratory, the Y-12 Weapons Plant, and the K-25 Site (formerly the Oak Ridge Gaseous Diffusion Plant) in Oak Ridge, Tennessee; the Paducah Gaseous Diffusion Plant in Paducah, Kentucky; and the Portsmouth Gaseous Diffusion Plant in Portsmouth, Ohiois in its fifth year with over 225 task orders. The program has been highly successful and has obtained from DOE outstanding marks for achieving quality, responsiveness, timeliness, and subcontracting goals.
Planning is a term that is overused when discussing management philosophies. However, strong, competent initial planning has proven critical to the success of this large ERWM task order contract. Most companies begin planning for such large contracts well before they are published in the Commerce Business Daily. Marketing groups develop teams, alliances and strategies to win these contracts. This effort may cover a period of several months and incur costs into the hundreds of thousands of dollars. As the proposal becomes formulated, it sets the stage for the operations and management of the contract. This effort cannot be done in a vacuum by the marketing groups. It is imperative that the operations managers, especially the ones who are designated in the proposal as key staff, play an active part in the formulation of the teams, alliances and strategies.

TEAM STRUCTURE

The structure of the team is the first aspect to be developed. Questions that must be asked include the following: Can your company show strong expertise to cover all aspects of the Request for Proposal (RFP)? Can your company provide adequate resources to support the depth of the contract proposed? Does the RFP outline subcontracting goals or targets for small businesses (SBs) and small disadvantaged businesses (SDBs)? Does your company have the key resumes required to sell the team? After answering these questions honestly, additional issues need to be considered: If you are putting together a team, who should be on it and how will it be structured? Any team members should be chosen based on specific expertise that they bring to the project. Homework that should be done on these potential team members, whether in-house or subcontractor, includes ensuring that they are in good standing with your potential client. If more than one large company is to be part of this team, it is very important that the two companies complement each other. It is also important that SB and SDB firms complement the prime contractor wherever possible. Track records of working with these companies should be considered. Remember that you may be stuck with someone with whom you do not want to work over the long haul. Like a marriage, these relationships often can be difficult to dissolve.

Integrated Team Structure

After selecting your teaming partners or subcontractors, you must choose one of the two main organizational structures: a strong prime contractor with separate subcontractors or a strong prime with an integrated team of subcontractors. If the nature of the work under the contract will allow you as the prime to discretely subcontract pieces of work, a strong prime with separate subcontractors would be the best approach. If discrete packages cannot be broken out, an integrated team approach is best as an organizational format.

An integrated team consists of a prime contractor whose responsibility is to manage, control, and provide technical expertise and other support to the program. Under the DOE-ORO ERWM Remedial Design Contract, Foster

Wheeler Environmental is the prime contractor with staff in the following key positions: Program Director, Deputy Program Director, Operations Manager, Engineering Manager, Program Controls Manager, Quality Assurance Manager, and Health and Safety Manager. This structure enables the prime to maintain control and provides the client with one entity with which to work. Senior positions such as project managers, lead engineers, and lead scientists are filled by the most competent individuals regardless of company affiliation.

To help assure that your integrated team will function smoothly, it is ideal to have the team collocated under one roof. Foster Wheeler Environmental's ERWM Contract has six contractors and subcontractors under one roof. It seems that by their nature, engineers, scientists, technicians, etc., are task oriented. It has been Foster Wheeler Environmental's experience that once individuals are assigned to a particular task/project and are collocated to that project, they become project team members foremost and their company affiliation becomes secondary. Integration in that manner has been achieved rather seamlessly. The organizational challenges come not at the team member level but at the supervisory level, especially if there are supervisors at the same level, representing different companies. Supervisors will tend to place billability interests for their company employees above the interest of other teaming companies. This tendency must be monitored and managed by the prime contractor's senior management. The key to managing this situation is to have up-front agreements with written guidelines for all to follow. On the ERWM contract one subcontractor that does not have its own local office occupies offices under this roof that have been set aside for its corporate functions. Other than this, it is virtually impossible to tell for which company individual staff members work.

Integrated Team Personnel

As these teams are assembled, another major opportunity arisesthe selection of key individuals. The key members of the organization must be able to work and to live together. The key staff must be chosen based not just on their technical and managerial expertise and their abilities to sell the team, but also on their personal traits. Can they work together over long hours, days and weeks in the same cramped quarters? Can they leave their egos at the door for the benefit of the program? Can they be on long-term assignment if required? Technical and managerial superstars may not be ideal team members.

As the prime contractor, an important responsibility is your concurrence concerning key individuals selected from your subcontractors. Remember that they must operate as an integral part of your team. If you have an integrated team, it is also important that you have a subcontractor manager/supervisor located with you on whom you can rely to support you and get things done within his/her own company.

If your team is composed of SB and/or SDB subcontractors, the following considerations must also be taken into account: Can the SB/SDB supply the staff required without hiring new personnel? Is the SB/SDB a viable firm or just a body shop? What is the SB's/SDB's professional reputation? What degree of commitment has the owner/president given you? As the prime contractor, you may have to establish goals for SB/SDB participation. As with your own staff and the staff of your large subcontractors, it is important that you concur with the individuals offered by your SBs/SDBs to support your contract. In these days of tight and dwindling budgets,

it is very important that everyone on your team carries his/her own weight.

Once your program has started and you have selected your initial cadre of staff, a likely scenario is for your client to bring you additional work that is going to increase your staff by over 50%. What do you do? If you are unsure that this upsizing is permanent, you may want your subcontractors to supply the required staff. Whether or not the additional staff comes internally or from your subcontractors, it is very important that you maintain control over your team. Whenever possible, concur with all new hires or transfers into your program. Remember that many of the individuals offered to you may be offered because their managers view this as a viable way to get rid of performance problems. You must be careful whom you accept.

CLIENT INTERFACE

A crucial task as you develop your proposal is to make sure that you know who your client or clients really are. Foster Wheeler Environmental's ERWM Contract, for example, was held by the DOE-ORO Project Management Division of DOE Construction and Engineering. DOE's Office of Environmental Restoration and Waste Management determines the ERWM work to be done and controls the budgets to implement the work. The Maintenance and Operations Contractor, Lockheed Martin Energy Systems, oversees all work done on the DOE-ORO reservations. Energy Systems transferred funding from its accounts to the DOE Project Management division so that Foster Wheeler Environmental's DOE Contracting Officer's Representative had the funding in place to release work to the Foster Wheeler Environmental team. Although the contract was with one entity, several groups and organizations played major roles in the day-to-day operation of the program.

Contract Negotiations

After the notice of intent is received, you will begin contract negotiations with your new client. You must know to what you are entitled and to what the client is entitled. If this is a government contract, use the Federal Acquisition Regulations (FARs) as the basis for your negotiations. The FARs protect both the government and the contractor. If your preplanning is adequate, you will have in hand written agreements with your subcontractors. These good faith agreements will need to be modified as your prime contract is finalized, but they give you the necessary latitude in negotiations with the client. In addition, new procurement strategies that are being popularized must be evaluated for relevance, including the cost plus incentive fee and fixed price strategies. Due to Cabinet-level interest in procurement reform, you must take an aggressive approach to these new procurement scenarios.

Client Communication

During the initial months of your program, it is imperative that frequent communications be held with your client(s), even on a very informal basis. One of your primary goals is to support and to be responsive to your client in order to ensure continuing and, hopefully, new work. One way to improve communication is to become very active in the community. Not only does that provide for excellent public relations, but it also provides access to information that is vital to your existence, such as what your competitors are doing and how are you being perceived in the community and by the client. This is where you will first hear warning signals of any problems with your team. Caution is merited because

perceived problems can have effects that are just as real as the effects of real problems and must be dealt with immediately.

Particularly as your program starts to produce deliverables, it is important that quality and responsiveness go hand in hand. Initially your reputation is very fragile or nonexistent. What your team does in the first several months will set the tone for the next several years. Adequate preplanning and key staff involvement will establish your reputation with the client as a team that operates in a responsive mode, rather than in a reactive mode. It is virtually impossible to change initial impressions. In fact, your contract may not last long enough to change those impressions.

One common mistake made by contractors hired as "experts in their field" is to place too much emphasis on their internal ability to determine what the client needs for any given project. More often than not, the client has a good grasp of the problem at hand and the required solution. The contractor's expertise is required to effect that solution within the allocated resources (budget, schedule, etc.). Today's client is usually very well informed and can be expected to provide resource constraints. The operational phase of a program can be expected to be a continuous process of tailoring work and deliverables to effect the predetermined solution within the resource constraints while at the same time managing no compromise in quality.

CONCLUSION

In summary, the success of an ERWM task order contract is directly related to the preplanning and initial operational involvement of the key individuals on the team. These key individuals should be involved in all phases of structuring the team. Senior staff involvement and their review of all deliverables are crucial to ensuring the quality, and providing adequate control, of the product delivered to the client. Frequent communication, both with the client and with others in the community, is critical to the survival of your contract. Planning to incorporate these recommendations, drawn from the experience and success of the DOE-ORO ERWM Program, into your program will be a first step towards assuring that your contract meets with similar success.

ROCKY FLATS ACCELERATED SITE
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ABSTRACT
An Accelerated Site Action Project (ASAP) is being developed at the Rocky Flats Environmental Technology Site to address the fundamental problem at Rocky Flats, that the current nuclear material stabilization and environmental cleanup activities are too slow, too uncertain in outcome, and too costly. A draft Phase I ASAP plan was developed to shatter certain cost and schedule paradigms and described a single, feasible alternative among many for achieving the most rapid and economical stabilization of the Site. This feasible alternative provided for interim closure of the Site and would make the Site nearly risk free for onsite and offsite populations and enable alternative uses of most of the Site's 6500 acres. ASAP Phase II, currently in progress, builds on the Phase I work to develop and evaluate additional alternatives based on technical and stakeholder input. The ASAP represents a step change in the mindset and management approach to achieve the vision of the future for the Site.

INTRODUCTION

An Accelerated Site Action Project (ASAP) is being developed at the Rocky Flats Environmental Technology Site (Rocky Flats or "the Site"), to radically decrease the Site risks and increase land availability for potential other uses as compared to the Site's previous course of action. A draft plan document dated October 9, 1995 was prepared as Phase I in the process to accelerate the closure of the Site. The document described a single, feasible alternative among many to achieve the most rapid and economical stabilization of the Site. This feasible alternative provided for interim closure of the Site and would make the Site nearly risk free for onsite and offsite populations and enable alternative uses of most of the Site's 6500 acres.

The ASAP Phase I feasible alternative began to bracket what is possible at Rocky Flats before the next phase of planning began. This initial feasibility was necessary to shatter certain cost and schedule paradigms at the Site that, if left unchecked, could have prevented or significantly lengthened the eventual cleanup. ASAP Phase II, currently in progress, builds on the Phase I work to develop and evaluate additional alternatives based on technical and stakeholder input while continuing to seek the optimum mix of solutions to create an integrated project for Site closure.

PROBLEM DEFINITION AND PROJECT STRATEGY

The fundamental problem at Rocky Flats is that the current nuclear material stabilization and environmental cleanup activities are too slow, too uncertain in outcome, and too costly. The Department of Energy (DOE) Plutonium Vulnerability Study identified the Site as having the highest-risk plutonium facilities in the nation, and these facilities are located within 50 miles of the Denver metropolitan area's 2 million people. Compounding this problem is that even with past high funding levels, the Site has had difficulty making meaningful progress toward cleanup. In July 1995 a new contractor operating under a new performance-based contract took over operations at Rocky Flats. Now just as the Site is poised to make progress, the budget is falling to levels that allow for little expenditure on mortgage and risk reduction in the face of high nuclear facilities baseline safety costs. The projected outyear funding profile cannot address DOE's commitments for plutonium and waste treatment and stabilization, or environmental cleanup.

At the heart of ASAP is a shift in management focus to achieve tangible and meaningful results for the funds expended. The Site's new management strategy is based on the following elements:

To establish a unifying vision of an interim state for the Site that will simultaneously reduce risks and budget outlays and that can be achieved in the professional lifetime of the people working at the Site

To continue to seek ways to achieve early removal of plutonium and waste from the Site

To enable DOE, the Contractor, and the workforce to bring the Site to a stable, interim closure state at the earliest possible date, with additional closure actions accomplished after this date at a lower overhead rate and higher efficiency than the current Site work structure allows

To challenge current strategies for environmental restoration, waste management, and plutonium stabilization and storage to achieve risk reduction and land-use value for much lower costs and with faster schedules

To recognize that the march of time represents the greatest cost at the Site. The Site has spent over $700 million per year in the recent past with little highly valued progress. The annual baseline for keeping the plutonium facilities safe and stable is about $400 million. Therefore, every month of inactivity or indecision on a path forward is costing taxpayers more than $30 million. This lost opportunity cost, which was simply accepted in the past, must be factored into all future decisions.

To aggressively challenge existing baseline activities and costs, including both DOE and environmental regulatory burdens

To view and manage Site activities as projects to better align DOE, the Contractor, and the employees, and to increase accountability for scope, schedule, and cost

THE INITIAL FEASIBLE ALTERNATIVE

The ASAP Phase I was intended to develop an integrated alternative to test the hypothesis that with unrestricted funding the Site could be substantially closed in eight years for around $5 billion. The key features of the Phase I feasible alternative, Plutonium and Waste, Land Use, and cost and Schedules, are described below.

Plutonium and Waste

The plutonium and containerized transuranic waste will be in safe, interim storage awaiting the earliest possible shipment from the Site.

Land Use

The following land uses would be enabled by this alternative:

The outer 5,000 acres of the Buffer Zone would support unrestricted use, including open space.

An inner 1,000 acres of the Buffer Zone would meet standards for use as restricted open space.

Of the remaining 500 acres associated with the industrial area or landfills, 300 acres would be cleaned up to allow future industrial or commercial development, if desired. The 200 most contaminated acres, including the current plutonium processing area, would be safely closed with a long-term barrier cap with long-term monitoring to ensure cap performance and integrity. A groundwater diversion system and passive reactive barriers would be installed to protect offsite ground and surface waters.

Bulk low-level cleanup and demolition waste would be placed under the cap.

Most buildings, except those deemed by local stakeholders to have a future economic value, would be demolished or covered by the cap.

The feasible alternative would not compromise the ability to clean up the entire Site in the future.

Cost and Schedule

As shown in Fig. 1 this plan can be accomplished for about $6 billion, compared with the current Baseline Environmental Management Report (BEMR) estimate of more than $22.5 billion. However, it requires larger annual budgets through the year 2002 than are currently anticipated. Alternatively, preliminary estimates indicate that the work represented by this feasible alternative could be accomplished by about 2015 for about $10 billion with a slightly more aggressive funding scenario than is currently planned. Figure 2 depicts several different funding scenarios being explored. Funding variations greatly impact the schedule. The critical path items are plutonium processing (stabilization), the final decommissioning of Building 707 and its support buildings, and the placement of the barrier cap over the Protected Area.

Fig. 1

Fig. 2

By the end of 2003 (or 2015), the Site population could drop to less than 300 from the current figure of more than 5,000. The annual operating cost could be less than $40 million, down from more than $600 million currently. The remaining facilities would be configured so that the final closure cost, and the demolition of the interim plutonium and waste storage facilities, would cost less than $200 million. In ASAP Phase II the technical logic, schedule, and costs for several different alternatives are being studied. Preliminary analysis indicates that bringing the 1,500 acres that are currently not designated as residential-capable to that standard would cost an additional $10 billion and take at least an additional decade to complete.

There will also be additional costs, yet to be estimated, associated with the final disposition (shipment) of stored plutonium and waste. These outyear (beyond 2003) costs could be several hundred million dollars depending on ultimate disposal costs and criteria.

KEY ISSUES TO BE RESOLVED

ASAP Phase I has elicited extensive interest among DOE and non-DOE stakeholders. Comments have been provided at meetings with DOE Headquarters, regulators, and citizens' groups. A number of issues have been identified, many of which are currently being addressed in the ASAP Phase II work. Some of these key issues to be resolved include the following:

Considering the logical array of alternatives that address most stakeholder concerns, and determining aggregate stakeholder priorities.

Evaluating methods to expedite plutonium and transuranic waste shipment from the Site.

Considering onsite disposal of low-level and low-level mixed wastes.

Achieving a fundable alternative. It is not clear that the current alternative, even with its dramatic cost and schedule savings, will be funded in preference to a project of longer duration.

Achieving consensus on the strategies for plutonium and waste storage and facility decommissioning.

Determining the level of plutonium and waste processing, consistent with national interests, that should be done before the materials are placed in potentially long-term storage.

Establishing a prudent planning horizon for the possibility of long-term storage of plutonium and waste.

Determining the optimum remediation or stabilization strategy for soil and groundwater to identify the cost-benefit tradeoffs.

Determining the appropriate authorization basis and safety controls necessary to balance safety and efficiency in proceeding with plutonium, waste, and decommissioning activities.

STAKEHOLDER INVOLVEMENT AND PROJECT PRECURSOR EVENTS

The ASAP concept, which had its formal beginning on August 1, 1995, had many important precursor events that contributed to the idea in many ways. The ASAP incorporates many of the features of previous work at Rocky Flats and represents an integrated compendium of the major issues and agreements needed to achieve acceptable closure of the Site eventually.

In 1993, the Site mission changed from nuclear weapons production to cleanup.

In 1994, DOE responded to the mission change by issuing a request for proposals to procure a new performance-based integrating management contractor for the Site to carry out the new mission.

In early 1994, the Site began negotiating a new cleanup agreement with the US Environmental Protection Agency (EPA) and the State of Colorado to reflect the fact that the previous agreement, the Interagency Agreement (IAG), did not reflect the mission change.

In 1994, the Defense Nuclear Facilities Safety Board issued two important recommendations (94-1 and 94-3) dealing with plutonium stabilization and storage at the Site.

In 1994, with input from various stakeholder groups, a Strategic Plan was issued that focused on nuclear material stabilization, waste management, and environmental restoration.

In early 1995, several public meetings were held regarding the then-proposed solar ponds remediation plan, which involved construction of a Corrective Action Management Unit (CAMU) waste facility at the solar ponds.

In early 1995, several important interactions with regulators and stakeholders were held to review the path-forward options for the Site. One of the most important of these was the March 4, 1995 summit. At the summit there was a consensus to place a higher priority on risk reduction by stabilizing plutonium than on environmental remediation. Onsite disposal was also discussed by many attendees as a way to cut costs to enable more risk reduction. Another important event was the April 1995 regulatory summit, at which similar conclusions were reached.

In early 1995, Kaiser-Hill was selected as the new contractor to carry out the Site mission. Kaiser-Hill assumed responsibility for the Site on July 1, 1995.

In June 1995, the Future Site Use Working Group issued a consensus opinion that the Buffer Zone should generally be maintained as open space and that the Industrial Area should be for industrial use. No onsite residential use was anticipated.

A series of meetings were held in the summer of 1995 with the state of Colorado, regulators, and many stakeholders regarding the alternatives for plutonium storage at the Site, including the possibility of constructing a new vault. The possibility of long-term storage of plutonium at the Site was addressed.

Information related to the ASAP Phase I plan was shared with a broad cross-section of stakeholders in many different private and public forums. Participants' key areas of concern align very well with the alternatives being analyzed in the ASAP Phase II. In general the stakeholders require that:

Everything is being done to remove the plutonium and waste from the Site at the earliest possible time.

The plutonium is stored in the safest possible configuration.

The implications and decisions are clearly outlined and openly made.

There will be an adequate stakeholder involvement program.

The tradeoffs among cost, schedule, and final cleanup criteria have been properly balanced.

Future land use options, including both dedicated open space and economic conversion, have been accounted for and enabled.

TECHNICAL SUMMARY

The ASAP Phase I document presented the feasibility analysis that was performed to determine whether an accelerated closure of Rocky Flats is possible. The analysis was structured into six major tasks: 1) Plutonium Consolidation and Stabilization, 2) Waste Management, 3) Facility Decommissioning, 4) Interim Closure, 5) Site Infrastructure, and 6) Implementation. Just as the ASAP is a logical integration of many precursor events, so a number of the technical tasks discussed below are continuations of currently planned activities, such as major plutonium stabilization.

Plutonium Consolidation and Stabilization

Highly enriched uranyl nitrate (HEUN) solutions would be bottled and shipped off the site.

Plutonium (Pu) metal and oxides will be packaged to meet DOE-STD-3013-94 in double, welded stainless-steel containers.

Pu solid residues will be processed to meet safe, long-term storage criteria. Where possible, residues will only be repackaged and managed as Transuranic (TRU) waste.

Pu liquid residues will be moved from their current containers and stabilized for long-term storage.

The Pu and residues to be managed as Special Nuclear Material (SNM) will be stored in a newly constructed storage facility (vault) after consolidation for staging in Building 371.

Pu and residue processing will be conducted primarily in Building 707.

Pu pits will be packaged in approved shipping containers.

The estimated cost for this activity is $800 million, of which about $150 to $200 million is for a new passive plutonium vault. The critical path schedule for both residue and Pu stabilization is expected to continue through 2001. Continued planning and analysis are needed to shorten the required schedule. The post 2003 operating costs are estimated to be $20 million.

Plutonium consolidation and stabilization issues that remain to be resolved include the following:

The onsite Pu storage location (new vault, Building 371, or other alternative)

The implementation methods to meet the criteria for safe, long-term storage of residues

Schedule compression, including technology choices for residue stabilization

The quantity and types of materials to be shipped to Waste Isolation Pilot Plant (WIPP) or other locations

Waste Management

A new hardened TRU waste storage facility would be constructed. This facility may store up to 3,000 kgs of plutonium within the waste matrix of approximately 20,000 drums.

Bulk Low-Level Waste (LLW), Low-Level Mixed Waste (LLMW), and remediation and decommissioning waste would be disposed of in an onsite disposal facility located in the Industrial Area (probably in the Protected Area). Some waste may be shipped offsite and most waste will be disposed in such a way that will facilitate future offsite disposal if it becomes cost-effective.

Waste treatment would be accomplished only as necessary, regardless of regulatory imperatives to ensure safety for long-term storage or disposal.

Waste would be stored temporarily in existing buildings outside the Protected Area while long-term storage capacity is constructed.

All landfills and waste facilities would be closed by the end of 2003. The cost for this alternative is about $550 million with about $100 million required for the new hardened TRU waste storage facility. This new facility would be expected to be operational in the year 2000. Long-term Operations and Maintenance costs after 2003 are estimated to be about $3 million per year.

Some waste management issues for further consideration include:

Onsite as opposed to offsite storage/disposal of all waste types

Treatment criteria for long-term storage or disposal for all waste types and regulatory alignment with criteria

Lower-cost options for storing TRU waste other than in a hardened facility

Facility Decommissioning

Most buildings of the 425 facilities at the Site are not radiologically contaminated and will be dismantled except for those deemed economically valuable (e.g., National Conversion Pilot Project).

The major plutonium buildings and Building 881 will be partially dismantled with the lower-level portions entombed. This would involve removing significant contamination and then filling the basement areas with demolition debris, entombing the basement with material such as impervious clay slurry, and covering it with a landfill cap.

Many of the plutonium buildings must remain operational for several years in order to consolidate and stabilize plutonium and waste. The approach will therefore be to remove administrative and ancillary buildings first in order to clear space and to level the workload. Major plutonium facilities, such as Buildings 371 and 707, will be the last to be completely decommissioned.

Current planning indicates most facilities can be decommissioned by the end of FY02 at a total cost ranging from $1.5 billion to $2 billion.

The remaining facility decommissioning issues to be addressed include:

Further refinement of building sequencing and further development of a detailed logic, safety authorization basis, and cost estimates for all buildings

Ensuring onsite and offsite safety during decommissioning

Determining the cost-benefit tradeoffs regarding the degree of contamination to be left with the building rubble or placed in a disposal cell

Defining the regulatory process for decommissioning

Determining the workforce composition and skill mix

Interim Closure

The potential exists for unrestricted use (from a contamination perspective) of 5,000 of the Site's 6,500 acres. An additional 1,000 acres would be suitable for use as unoccupied open space, and 300 acres would be suitable for industrial reuse. The remaining 200 acres would be placed under a landfill cap, which includes the current landfill (OU7), the 800 old uranium processing area, and the plutonium processing area (Protected Area).

Groundwater and surface water would be protected to national standards for water leaving the Site by upgradient diversion and downgradient passive reactive barriers.

About 40 of the 173 individual hazardous substance sites (IHSSs) that are high or medium ranked from a risk perspective would be remediated. The remaining 133 Sites have low enough contamination and risk levels that they can be cleared without further action.

The cost for this activity is about $400 million with about half the cost for IHSS remediation and half the cost for landfill caps and water management. Long-term monitoring will cost about $3 million per year after 2003. The final cap is on the critical path schedule, will take about two years to complete, and extend about six months after the last building (Building 707) has been fully decommissioned.

Interim closure Issues to be resolved include:

Acceptability of the land use restrictions of the feasible alternative and the standards that apply to those land-use criteria. Additionally, there are many who favor a 'greenfield' site cleanup that would cost another $10 billion or so to achieve.

The methods to be used for surface and groundwater control (e.g., reactive barriers) and water quality standards;

Integration with the issues to be resolved for waste management and facility decommissioning regarding onsite disposal of waste and decommissioning materials;

The design of the landfill cap to ensure long-term integrity; and

The impacts of excavating and placing the more than 2,000,000 cubic meters of material needed for the landfill cap.

Site Infrastructure

The feasible alternative requires a site infrastructure to support the remaining plutonium and waste storage facilities and about 300 total staff.

Relocate most infrastructure offsite. This would be accomplished by using public or commercial utilities to provide utilities services directly to buildings. Sewage would be managed onsite in a small lagoon or septic system.

Most emergency and health services would be contracted for offsite, except for the nuclear material protective force.

All office and support facilities would be located offsite to reduce the demand for expensive onsite infrastructure.

Cost of development of the infrastructure for commercial service to the Site and conversion of other site infrastructure to the new configuration is estimated at about $70 million. The post 2003 annual operating costs are estimated to be $12 million. There are no critical path schedule items in this conversion.

Site infrastructure issues to be resolved include:

Verifying the desirability of this fairly radical reconfiguration and the ability and willingness of the commercial and public utilities and emergency services to serve the Site

Timing for implementation (sooner or later) of individual components

Implementation

Regulatory Alignment - The regulatory structure and process needs must be aligned to accomplish the Site closure mission. Currently, the Site is regulated as an operating production facility, which creates regulatory road blocks to an expedited cleanup that would not exist at a typical Superfund site.

Workforce Restructuring - DOE and the contractor will need to coordinate realignment of required work, skills mix, and retraining to provide the most productive use of the workforce. A human resource plan must consider the inevitable downsizing of the workforce.

Stakeholder and Political Alignment - A scope, schedule, and funding package needs to be developed that meets the consensus needs of both the funders (i.e., Congress, DOE/HQ) and the beneficiaries (e.g., Colorado and the nation at large) of the project.

Site Productivity - The Contractor and DOE need to continue to evaluate procedural and motivational methods to increase the Site's productivity and cost, scope, and schedule accountability.

Projectizing - The implementation plan calls for projectizing the Site, as indicated by the word "project" in the ASAP. Projectizing has some significant structural and process implications for DOE, the Contract, and the Contractor.

NEXT STEPS

This paper primarily summarizes the Phase I of ASAP. Adjustments in the FY96 operating activities are currently being made to accommodate the most basic features of the strategy. Phase II of ASAP, to be accomplished by early February 1996, will develop and evaluate alternatives to address the key policy choices and issues described at the beginning of this paper. This process will also look at outyear (beyond 2003) alternatives to ensure that the final Site strategy is represented and will include rigorous stakeholder and regulator involvement. It is expected that ASAP Phase II will facilitate mutual understanding of the key issues and allow DOE to arrive at a best solution during Phase III that balances competing needs and concerns.

Once the overall integrated ASAP plan has been developed and necessary consensus reached, key decisions and issues will be segregated to determine decision pathways and time frames. A plan will also be developed for more detailed stakeholder involvement in the alternatives in those second-tier decisions.

CONCLUSION

The Accelerated Site Action Project represents a step change in the mindset and management approach for driving the Rocky Flats Environmental Technology Site to closure (Fig. 3). It radically alters the vision of the future for the Site, and moves the reality of Site closure into the personal career time frame for most of the Site workforce and local stakeholders. Most important of all it shatters the cost and schedule paradigms for closure of DOE nuclear sites that say closure must take many decades, and demand budgets that the Congress may never appropriate. ASAP presents the argument that the closure can be done quickly, safely, economically, and in a manner that greatly reduces the risks posed to the Site workforce and surrounding communities. ASAP will be neither easy to

implement nor without controversy, but it offers a feasible solution to achieve the closure of the Rocky Flats Environmental Technology Site and thus conclude a major chapter in the nation's nuclear weapons legacy.
Fig. 3

35-5
HOW TO ACCELERATE THE FERNALD REMEDIATION

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ABSTRACT

The Fernald Environmental Management Project is unique among Department of Energy (DOE) sites by virtue of successful efforts by the Fernald Environmental Restoration Management Corporation (FERMCO) and DOE-Fernald Area Office (FN) in securing a stakeholder-assisted final site closure vision and all Record of Decisions (ROD) or Interim RODs required to set the stage for final remediation. DOE and FERMCO have agreed in principle on a Ten Year Plan which accelerates all activities to remediate the site in approximately half the target schedule.

This paper presents the path that led to the current Ten Year Plan, the key elements of the plan, and the implementation strategies.

PATH TO THE TEN YEAR PLAN

The Fernald site is divided into five operable units (OUs) pursuant to the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) process. RODs are in hand for all five OUs. These RODs summarize the results of the Remedial Investigation/Feasibility Study (RI/FS) reports and detail the selected remedial actions. The five OUs and the primary remedies are:

OU1 - Waste Pits: excavation, drying and shipment to an off-site disposal facility.

OU2 - Other Waste Units: excavation and disposal in an on-site facility.

OU3 - Former Production Area: Dismantlement of structures and disposal on and off-site.

OU4 - Silos: Vitrification of the waste and disposal at the Nevada Test Site.

OU5 - Environmental Media: excavation of soil and disposal in an on-site facility; pump and treat groundwater.

With the remedies chosen and the RODs in place Fernald is in a position where it has the technical scope, schedule and budget defined for each Operable Unit. The first step in the development of a site remediation plan was the integration of the schedules of the five Operable Units into one comprehensive overall plan, with the duration of the work for the five OUs optimized and with a defined critical path. Once the schedule was defined, the scope and costs were reviewed to put them all on a consistent technical basis and to identify efficiencies which were realized by the integration of the OUs.

Initially, two unconstrained funding scenarios were developed: a seven-year plan and a ten-year plan. The seven-year plan contained many of the same assumptions as the ten-year plan, however, it assumed that some of the DOE Orders, requirements and restraints were removed. Funding for these plans peaked at $351 million and at $331 million for the seven and

ten-year plans, respectively (unescalated). The site at the end of either of these plans was completely remediated except for the continuing pump-and-treat operation needed to complete remediation of the aquifer. Nothing was left on site except for the disposal facility, the wells and the treatment facility.

These plans were provided to DOE Headquarters and to stakeholder groups. When challenged by DOE Headquarters to determine how quickly the site could be remediated if the funding from FY97 on was set at $276 million in FY96 dollars and the seven-year assumptions were used, the current Ten Year Plan ($276 Million Case) was developed.

At the time the Ten Year Plan was completed, Fernald was projecting a 25-year remediation schedule and a cost from FY96 to completion of $5.8 billion (escalated), in accordance with the then-current target budgets. The cost estimate for the Ten Year Plan is $3.4 billion. Its shortened schedule and substantial cost savings made it an attractive proposal. In addition, Fernald will be the first DOE site of major consequence to be totally brought to clean closure. Successful implementation of the Ten Year Plan will set benchmarks and pave the way for similar successes at other major DOE facilities simply by showing it can be done if the right approach is used.

Key to acceptance of the Ten Year Plan were the substantial cost savings, the acceptance of the plan by the regulators and public stakeholders and their active support of the plan with DOE and Congress. The regulators involved are the US and Ohio EPAs. Public stakeholders include the Fernald Citizens Task Force and the citizens' group Fernald Residents for Environmental Safety and Health (FRESH). Ohio Congressmen and Senators also supported the adoption of the Ten Year Plan.

Once the Plan was accepted by DOE-HQ, Fernald was requested to undertake detailed FY96 and FY97 planning to support it. The plans for FY98 to FY05 were also requested, at a planning package level of detail. These plans will be completed in February 1996 and will provide the baseline for Fernald's final remediation .

KEY ELEMENTS OF THE TEN YEAR PLAN

Figure 1 illustrates the Fernald site in the years 1995 through 2005. The progress of decontamination and dismantlement (D&D) of site facilities and the construction of the on-site disposal facility can be clearly seen. In the year 2005, all that will remain at Fernald is the disposal facility and a groundwater pump-and-treat system. The final land use has not been determined, except that residential or farming use will not be permitted. The final land use will be determined by DOE and the stakeholders with additional public input.

Fig. 1

Figure 2 depicts the overall schedule and indicates the critical path activities. Once the remaining nuclear materials are removed and the safe shutdown activities (removal of in-process material from lines, tanks and equipment) are completed, the critical path consists of D&D of the buildings, excavating the soil and underground foundations, piping, and utilities and building the on-site disposal facility.

Fig. 2

Figures 3 through 10 provide the key indicators or performance measures for the remediation of Fernald. They are:

Fig. 3

Fig. 4

Fig. 5

Fig. 6
Fig. 7
Fig. 8
Fig. 9
Fig. 10

TEN YEAR PLAN IMPLEMENTATION STRATEGY

Several changes have been made or are being implemented to streamline project operations, procedures or requirements to allow the Ten Year Plan to proceed within its schedule and budget constraints. They include projectization of the work, streamlining the DOE requirements which apply to the work, the use of innovative ways of working to achieve accelerated results, and the adoption of the performance-based fee approach.

Projectization

The five Operable Units as originally defined included all remediation within their physical boundaries. For example, OU4 included both vitrification of the silo materials and the remediation of the underlying soil. However, now that all RODs are in place, it is more efficient to handle OU4 soils under a Soil Remediation Project and to have this Project manage the soils from the other four OUs as well. The Soil Remediation Project will define the excavation parameters, contract for the excavation, and select the parameters that verify that the soil remaining is below the action levels. They also have the burden of defending these items to the regulators. There are other items within the five OUs that could be more efficiently handled by a single Project as well, such as the D&D of the remediation facilities and off-site shipment of waste to a commercial disposal facility. Therefore, these items were transferred to single Projects after consultation with the regulators. Five Projects were set up, each one containing the primary work in the OU, with the peripheral work transferred to the relevant Project. The five Projects are:

- Waste Pits Remedial Action Project
- Facilities D&D Project
- Fernald Residues Vitrification Plant Project
- Soil Remediation Project
- Aquifer Restoration Project

In addition to the scope movements, the projectization of the OUs was completed by changing their internal organization to a well defined project structure instead of a functional structure, defining various subprojects and naming project managers responsible for the subprojects and projects. In addition, the support groups Cost/Schedule Control, Procurement and Document Control were reorganized and aligned to this project structure.

DOE Requirements

The Fernald site was the first DOE site to have it Standards/Requirements Identification Documents (SRIDs) approved. These SRIDs define the DOE Orders and Requirements that apply to the Fernald Project. After completing the basic document, FERMCO has been active in streamlining these requirements to shorten schedules and reduce costs. Examples of these requirements are the request for waiving the requirement for a Performance Assessment for the on-site disposal facility, by showing equivalence through the CERCLA process, and waiving the requirement for disposal of waste in DOE off-site facilities. Eliminating these two requirements allows the shortening of the schedule and substantially reduces costs. FERMCO is actively pursuing the waiver or reduction of

numerous other requirements, many of them administrative in nature and applying more to an operating facility than a remediation site. The inclusion of standard commercial procurement practices is one of the areas being pursued.

Innovative Ways of Working

Fernald is exploring new ways of working on site that maximizes the use of off-site expertise. Recently Chem-Nuclear Systems Inc. (CNSI) was contracted to stabilize 6,000 gallons of thorium nitrate as a CERCLA removal action on-site. By using an off site subcontractor with experience in the treatment of significant quantities of thorium nitrate, a fully-trained work force and a proven "off-the-shelf" mobile treatment system were brought together as a unit, resulting in a highly successful remedial action. Use of experienced technicians to operate the system expedited training and increased safety. The entire project was completed less than six months after FERMCO contracted with CNSI to perform the work. Project costs were reduced by over 40 percent and the schedule accelerated by one year from the original plan which was based on design, construction, and operation of a treatment system in-house.

The accelerated schedule was possible because FERMCO used CNSI's Nuclear Regulatory Commission (NRC) field license to meet DOE requirements for handling radioactive material. DOE requirements and NRC requirements implemented by CNSI during the project were shown to be equivalent for project management systems; the quality assurance program; environmental, safety and health programs; training; and conduct of operations. Demonstrating that DOE and NRC requirements were equivalent provided the following benefits:

Eliminated review of CNSI engineering, design, quality assurance, procedures, and training programs during readiness assessment. CNSI's experience and attainment of an NRC license demonstrated their ability to conduct the operation.

Use of an "off-the-shelf" mobile treatment system tailored to meet the requirements of thorium nitrate treatment eliminated extensive design documentation and reviews and reduced the planning and design phase by approximately 10 months from the original plan.

Operation and control of the solidification process equipment was solely CNSI's responsibility under their NRC license. The project site was designated an "NRC compound" for processing thorium nitrate. CNSI's license requirements applied in this compound in conjunction with the DOE requirements. Extensive coordination was required to ensure that both DOE and NRC requirements for radiological protection were met.

The FERMCO on-site union personnel worked along side CNSI technicians in the NRC area. It was the first time such joint operations had been conducted at this DOE nuclear facility. To address the union issues, roles and responsibilities of FERMCO and CNSI were established early in the planning process, with support from Industrial Relations and input from the union. These roles were incorporated into the subcontract. Ultimately, the relationship was successful because CNSI made the union workers an integral part of the project team and listened to their suggestions and concerns for safety and the operation. The team produced outstanding productivity and safety records with operations completed without safety incidents and with radiation exposures below expected levels.

Another area of innovation under consideration at Fernald is the privatization of some of the project activities. Two that are being

considered now are the privatization of OU1 and the treatment of mixed-waste residue from production processes. Benefits expected to be derived from these efforts are reduced up-front cash flow and thus funding requirements to the project as the contractors will amortize their capital costs over the term of the work; utilization of off-site expertise in allowing contractors maximum flexibility applying their skills and methods to solve the problem; and lower overall costs due to the conversion of the efforts from cost reimbursable to fixed price/unit rate efforts. Studies are underway to work out the funding, contractual, technical, regulatory and other considerations of privatization. Decisions will be made in the first half of 1996 on whether to pursue the privatization of these two efforts.

Performance-Based Fee

Fernald was the first site to adopt the contract reform principles of performance-based fee contracting. In this concept, as described in detail in Ref. 1. the basic objectives are set for each six month period in alignment with the Ten Year Plan and include hard milestones linked to key deliverables where quality, safe work is required to maintain the schedule and soft objectives set to measure progress on procurement, safety, stakeholder involvement and other measures. By aligning both DOE and FERMCO personnel to the fee plan, a win-win situation is produced. Both DOE and FERMCO benefit if excellent performance is achieved in accordance with the fee plan. The use of this plan will keep the achievement of Ten Year Plan in focus and a high priority for all parties.

CONCLUSION

All of the prerequisites are in place to allow the Fernald site to proceed at an accelerated pace of remediation. The basic elements of the Ten Year Plan are laid out and are being detailed. Performance Measures have been selected to monitor the progress, and changes in project structure, organization, and requirements are being implemented to allow the Ten Year Plan to succeed.

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35-6

INTEGRATION OF EM ACTIVITIES AT THE INEL A SYSTEMS ENGINEERING APPROACH

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ABSTRACT

This paper describes the status of the Idaho National Engineering Laboratory (INEL) systems engineering approach to integrating Environmental Management (EM) activities. It provides a detailed description of how a team of technical and environmental program experts was assembled to examine integration opportunities and solutions to manage EM programs across the INEL. Using a conventional systems engineering approach, the team selected a preferred path that treats and stabilizes waste and material for disposal, accomplishes maximum volume

reduction of waste and materials for disposal, prepares the appropriate waste and material for shipment to the Waste Isolation Pilot Plant (WIPP) and the deep geologic repository, and minimizes the total cost by doing the work in the near-term rather than deferring it. Further, it details the work that has been accomplished to streamline EM activities at the INEL to meet the Department of Energy (DOE), Lockheed Idaho Technologies Company (LITCO), and stakeholder objectives in managing EM programs. Preliminary results of this integration effort identified savings of approximately seven billion dollars over the life-cycle of the program, when compared with the FY 1995 BEMR submittal. Additionally, this integration effort identified an alternative that moves the waste and materials from Idaho within a time frame acceptable to stakeholders and maintains acceptable risk levels. This activity has received wide acceptance at the INEL and within the State of Idaho, and has captured the attention of DOE-Headquarters and other DOE sites.

Integration activities have continued since the original March 1995 effort. The most recent product is the FY 1996 BEMR submitted to reflect changes to the INEL EM programs. This submittal includes reduced budget targets, except for FY 2001 and 2002, and the recent settlement agreement between the DOE and Governor Batt of Idaho. The accelerated schedules required by the settlement agreement generate a cost curve that spends more money from FY 2002 through 2015 to achieve significant savings over the life cycle of the INEL EM program, while simultaneously fulfilling the intent of the agreements and reducing costs to the targets. (See Fig. 1)

Fig. 1

WHAT IS THE PROBLEM WE ARE SOLVING?

The INEL must treat, dispose, and otherwise manage 405,000 cubic meters of waste, 700 cubic meters of spent nuclear fuel, and other materials in a time of declining budgets and increasing regulatory requirements. This could prevent the INEL from achieving final disposition of wastes in a time frame that is acceptable to stakeholders and from meeting the requirements of the settlement agreement between the DOE and the State of Idaho.

Historical funding mechanisms fostered compartmentalization or "stovepiping" of EM activities. This stovepiping restricted integration and synergism between the activities and impeded efforts to identify cost-efficient, effective solutions to managing EM programs across the INEL.

WHAT IS THE INEL'S SOLUTION TO THE PROBLEM?

In March, 1995 the INEL assembled a team of technical and environmental program experts to examine the problem. Using a conventional systems engineering approach, the team selected a preferred path, the Full Treatment Alternative, that:

- Treats and stabilizes the maximum amount of waste and material for disposal,
- Accomplishes maximum volume reduction of wastes destined for repositories,
- Prepares the appropriate waste and material for shipment to the WIPP and the deep geologic repository, and
- Minimizes total costs by doing work in the near-term rather than deferring it.

The team integrated waste streams and materials with treatment processing, treatment technologies, and facilities. All EM wastes and

materials were considered. The scope of this effort included all waste management, environmental restoration, technology development, facility transition, and infrastructure activities planned for the INEL. For all alternatives considered, risk was maintained within acceptable limits.

HOW DID THE INEL ACHIEVE THE SOLUTION?

Systems engineering, practiced successfully by the Lockheed Martin Corporation, was used. It emphasizes the management of change and the careful control of requirements, decision criteria, and costs. A computer model was developed for this task to allow comparison of alternatives. This combination of systems engineering and technical expertise, as illustrated in Fig. 2, produced an integrated solution.

Fig. 2

The systems engineering staff developed an eight-phased approach to developing the alternative set. The phases consisted of:

1. Requirements Analysis and Data Gathering
2. Functional Analysis
3. Scenario Development
4. Alternatives Development and Evaluation
5. Trade Studies and Systems Analysis
6. Refinement of Alternatives
7. Presentation of Results
8. Alternatives Verification and Post-Workout Activities

Joint DOE and contractor teams initially had difficulty tackling the requirements definition process because systems engineering requirements are somewhat different than requirements as usually defined in the DOE culture. Some resistance persisted through the identification of functions. But once the requirements and functions were displayed by waste stream and each team reviewed other teams work, team members began to accept the process. They quickly recognized the value of increasing their understanding of the interconnectedness, similarities, and differences between EM programs. Additionally, they began to understand that having clear requirements that aligned with work activities facilitated the identification and elimination of non-essential activities and the defense of essential activities. During scenario development it became apparent that the group was convinced of the value of the workout. (See Fig. 3)

Fig. 3

Breaking traditional ways of thinking was especially challenging. Systems engineers used innovative methods to involve team members in creative thinking. Various approaches to instilling paradigm shifts were attempted, with some success, but intervention from executive management finally got things moving. The teams, entrenched in tradition, needed to be encouraged to consider eliminating work and jobs, as well as to discover ways to relax perceived requirements while maintaining acceptable risk levels. As opposed to the former views of cutting costs by slipping schedules, teams began to modify scope and decrease associated costs. This was finally achieved in mid-December, just in time to meet the deadline for the BEMR submittal.

WHAT IS THE PATH FORWARD AT THE INEL?

At the INEL, technical integration across the programs is largely achieved. The INEL will continue to follow the systems engineering approach to managing EM programs. At the present time, the path forward is simply a modification of the original effort, an approach known as the Full Treatment Alternative. This alternative maximizes treatment and

minimizes final waste form volume for shipment to deep geologic repositories. It does not rely on the granting of a No-Migration Determination for WIPP. And finally, it has a far lower cost than traditional approaches to environmental management.

The path forward, as originally defined, continues to undergo improvement. A team is currently refining a defensible requirements and cost baseline. This activity is being accomplished in preparation for the budget submittal in April, 1996. Integrating the systems engineering approach continues to yield increased opportunities to streamline costs and work activities.

WHAT ARE SOME OF THE DIFFERENCES BETWEEN DOE AND AEROSPACE INDUSTRY SYSTEMS ENGINEERING APPROACHES?

All steps of the Systems Engineering Process are applicable to the DOE. Many parts of the process are infused into the DOE EM Culture, as formerly required by DOE Order 4700.X, Project Management System, and currently implied within DOE Order 430.1 Life-Cycle Asset Management. But since managing waste and materials is substantially different from building a rocket or an airplane, adjustments were made to accommodate the unique situation.

A Systems Management Approach, rather than a Systems Engineering Approach should be used. Systems Management combines Systems Engineering, Program Management, and Program Controls in an interactive attempt to instill discipline into managing programs. This approach still follows traditional Systems Engineering, but infuses the cultural aspects of the system with new rigor through the identification of requirements and functions. The cultural aspect of the DOE makes it resistant to the systems approach. Dynamic environments, role confusion and diffusion, and lack of trust between contractors, and between contractors and federal employees makes implementation especially challenging. (See Fig. 4)

Fig. 4

Efforts to "systems engineer" activities at other DOE sites have met with varied acceptance. Conventional approaches require the systems engineer to design modifications to systems based on specifications received from other functional organizations, and to turn back the final product once the acceptance testing is complete. In the DOE system, specifications are often non-existent for the human systems which require engineering. The most successful strategy identified to date is to have the systems engineers work directly with the program staff, both contractors and federal employees, to facilitate them through the process.

Lessons learned indicated that adjustments must be made when tailoring the traditional aerospace approach to the DOE systems. Some of these required adjustments are described below. Systems Engineers should:

Be sensitive to the requirements and concerns of the stakeholders,

Capitalize on DOE and contractor knowledge in the components of the systems process,

Avoid forcing the approaches or terminology onto processes,

Provide operational definitions of terms so that concepts may be accurately translated in the DOE language,

Teach or train individuals on the systems engineering concepts in a way that the benefits are obvious,

Be flexible in applying rigid approaches to a "soft, fuzzy" culture, and

Be prepared for confusion and rejection, before enlightened acceptance is realized.

Applying systems engineering within the DOE is really human systems engineering, and is less well defined than traditional systems engineering. Creating culture change is not an easy task, so continued emphasis should be placed on integrated systems management. The work of the EM Integration activities will become the INEL EM baseline. The challenge is to infuse that baseline into the way of conducting EM business at the INEL. Additionally, many non-EM systems engineering applications are underway at the INEL. These activities will continue as Lockheed systems engineers become more effective in supporting DOE requirements.

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PREPARATION OF A COMPREHENSIVE, LIFE-CYCLE WASTE MANAGEMENT PLAN FOR THE ROCKY FLATS ENVIRONMENTAL TECHNOLOGY SITE

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ABSTRACT

At the U.S. Department of Energy's Rocky Flats Environmental Technology Site (RFETS), planning efforts are keenly focused on achieving meaningful progress toward cleanup, decommissioning, and site closure. In the face of declining operating budgets, the site's Integrating Contractor, Kaiser-Hill Company, launched the Accelerated Site Action Project (ASAP). The ASAP proposes to accelerate interim closure of the site so that risks to the environment and surrounding communities can be greatly reduced and lands made available for other beneficial uses.

Rocky Mountain Remediation Services (RMRS) is helping to meet the challenges of the ASAP through preparation of the RFETS Life-Cycle Waste Management Plan (LCWMP). The LCWMP produces a waste management strategy for implementation with the lowest life-cycle cost, consistent with the overall direction and guidelines of the ASAP. The plan assures that waste management systems, facilities, and services at RFETS are configured to satisfy operational mission needs in the most cost-effective manner.

The LCWMP adopts a systems approach, utilizing a defined set of end-state requirements (i.e., disposal site waste acceptance criteria) to derive the preferred path for each individual waste stream. The large number of wastes at RFETS are classified into distinct groupings, defined by common source, bulk material, concentration of contaminants, and/or downstream treatment requirements. The LCWMP provides a comprehensive view of all wastes currently in inventory and those projected from future activities such as 1) consolidation, stabilization, and long-term storage of special nuclear material; 2) remediation of environmental releases; and 3) decontamination and decommissioning of former production and infrastructure facilities. The proposed routings for individual waste streams are analyzed within the context of the overall waste management system to ensure that an appropriate combination of existing and new facilities is planned for treatment, storage, and disposal.

At the core of the waste management planning effort is the application of analytical models to 1) simulate the flow of materials through storage,

treatment, and final disposal locations; and 2) estimate the life-cycle costs, for comparison purposes, of numerous different waste management alternatives. The models assist waste planners in rapidly assessing the technical and economic feasibility of different options in a repeated effort to refine and optimize the waste strategy. An ongoing review allows continuous reduction of the main risks and uncertainties associated with the strategy. Allowance is also made to accommodate any newly identified wastes.

Preparation of the LCWMP is allowing an integrated, comprehensive waste management strategy to emerge. With a clearly defined waste strategy, RMRS and the Integrating Contractor can justify and control the site's waste management baseline. Work projects can thus be integrated with other major programs and support operations at RFETS to assure the success of the ASAP.

INTRODUCTION

The Rocky Flats Environmental Technology Site (RFETS), formerly named The Rocky Flats Plant, is part of the U.S. Department of Energy's (DOE) vast nuclear weapons complex. The facility is located on approximately 6,550 acres in northern Jefferson County, about 16 miles northwest of Denver, Colorado. Constructed in 1952, Rocky Flats was in production for almost 40 years, manufacturing nuclear weapon components from materials such as plutonium, uranium, beryllium, and a variety of stainless-steel alloys. When the production mission was officially terminated in 1992 the site entered a period of transition, focusing future efforts on safe storage of special nuclear material; cleanup of environmental contamination; facility decontamination and decommissioning; waste management; and conversion of facilities for economic development. The major challenge to the Waste Management Program was whether systems and facilities were prepared to handle the types and amounts of waste that could be expected from such a major shift in site mission.

Today the site's goals are established on achieving final closure at reduced fixed and operating costs over previously baselined budgets. The site's Integrating Contractor, Kaiser-Hill Company, has launched an Accelerated Site Action Project (ASAP) with the intent of rapidly stabilizing materials and facilities at the site; reducing risks to the environment and surrounding population; and increasing the availability of land for other stakeholder-desired uses.

This paper presents the approach taken by RMRS to develop a Life Cycle Waste Management Plan (LCWMP) and discusses the importance of this document to achieving the site's current mission and goals. Also presented are elements of the waste management strategy, key waste issues at Rocky Flats, and potential solutions currently under consideration.

PURPOSE AND ROLE OF THE LCWMP

The concept of a life-cycle approach to waste planning was part of RMRS's strategy when awarded the contract for environmental remediation, waste management and decontamination and decommissioning in July, 1995. The techniques used at RFETS are based on previous parent-company (British Nuclear Fuels plc) success with a comprehensive strategy for intermediate level wastes at the Sellafield site in the United Kingdom.

The fundamental problem for the LCWMP is to develop an optimized strategy for managing the large variety and volumes of waste expected from future site decommissioning and closure activities. The plan will lead to an optimized configuration of waste management systems and facilities to satisfy operational mission needs. To be successful, it must blend

technical feasibility, life-cycle costs, and risk acceptance consistent with the overall site strategies and goals set by the ASAP. The LCWMP will provide the in-depth analysis necessary to support the ASAP and compare the costs of alternative approaches for each waste stream currently in storage or expected from future generation.

Additionally, the LCWMP supports the following RMRS objectives:

- Document a waste management program baseline in terms of scope, requirements, timing of events, and costs
- Develop a waste management system model with which to perform cost-benefit analyses on different waste management options
- Apply a "systems" approach to waste management using final waste form requirements to drive the processing and handling strategies for individual waste streams; assure that solutions to individual streams are sensible within the context of the overall waste management system
- Assess treatment, storage, and disposal (TSD) alternatives for each waste stream (i.e., technical approach, timing, life-cycle costs, risks, and uncertainties)
- Identify waste management system and facility deficiencies
- Produce a high-quality strategic planning service to support the Integrating Contractor and the ASAP

PROJECT APPROACH

The importance and anticipated level of effort demands that the Life Cycle Waste Management Plan be established as an individual project. The general approach consists of preparing a project plan; establishing a project team with responsibilities matrix; allocating a separate budget source; and placing the effort under the watchful eye of a senior management review group. Since the LCWMP is such a large-scale integration effort, accommodating plans and strategies of numerous organizations throughout the site, the composition of the project team is crucial to promoting communication across organizational boundaries.

The core team comprises the project manager from the Strategic and Integrated Planning department; support staff from Programs/Operations Support and Technology Applications; and a program lead advisor from the Integrating Contractor. This team is responsible for collection and synthesis of all pertinent waste management data; review of waste processing capabilities; interface with the site"s waste generators; and operation/analysis of system and cost models. The core team develops and issues proposed waste strategies and assures integration with site cleanup, decommissioning, and other risk-reduction activities. They are also responsible for identifying the principal risks and for maintaining an overview of the waste strategy.

Proposed waste strategies are distributed for review and revisement to waste storage, treatment, and disposal operations groups, who will become the eventual owners of the strategy and sponsors of key projects. Their involvement in the concept and scope of projects that arise from the waste strategy is important.

LIFE-CYCLE PLANNING PROCESS

Strategic planning at RFETS is proceeding on several levels. The Integrating Contractor is directing the Accelerated Site Action Project (ASAP) which will define the overall path forward for the site and the conditions under which site closure will occur. The ASAP is being developed through a phased approach to allow analysis of multiple site scenarios and full dialogue with regulatory agencies and other interested stakeholders. Decisions made through the ASAP will form the framework for

a more detailed waste management life-cycle cost analysis provided by the LCWMP. Figure 1 presents an overview of the life-cycle planning process and relationship between planning efforts. RMRS and the Integrating Contractor are working jointly to develop consistent data sets and assumptions for use in the ASAP and LCWMP.

Fig. 1

The process used to develop the LCWMP is explained in the following sections.

Gathering and Maintaining Waste Management Data

Collection, synthesis, and maintenance of adequate waste management data is a time-consuming process in life-cycle planning. Key information required for the analysis includes the status of on-hand waste inventories; waste generation forecasts; assessment of onsite treatment, storage, and disposal capabilities; knowledge of TSD options at other Federal and/or commercial facilities; and waste management costs.

Waste generators are requested to prepare and submit future waste forecasts for their individual programs. It is important that waste planners work closely with the generators to understand the bases and assumptions behind these estimates. Waste projections are a potential source of uncertainty in the life-cycle analysis and they have a tendency to fluctuate based on internal or external conditions imposed upon the programs (e.g., changing program budgets or differing cleanup standards set by a regulatory agency). Also, waste planners, with their understanding of waste processing and disposal criteria, can suggest waste minimization techniques or methods to improve waste packaging efficiencies.

System Modeling

Several analytical models are used to assist planners in visualizing the relative differences between options and the sensitivities of underlying assumptions. The first of these models combines annual waste generation projections with other waste management data such as on-hand inventories; capacities of individual storage units; present and future waste treatment capabilities; waste characterization and certification rates; and shipping/disposal capacities. This model is spreadsheet based and controlled by numerous links between worksheets and graphical output. The data are modeled to simulate the annual waste flow balance and to identify potential deficiencies (either in capacity or capability) or under-utilization in any area of the RFETS waste management system. Once a baseline scenario is analyzed, input parameters are varied to simulate other waste management scenarios or to test the sensitivities of assumptions.

The second model derives the life-cycle costs of each planning case. Its function is to differentiate the economics between optional paths for each waste streams or to project the life-cycle cost of the Rocky Flats waste management system as a whole (depending on system configuration and projected operating lifetime). RMRS has chosen to use the System Cost Model (SCM), developed for the DOE by a joint effort between Lockheed Idaho Technologies Co. and Morrisson Knudsen Corporation. This model has direct application to waste management life-cycle cost analyses and has been used by DOE-Headquarters in a number of prior studies. RMRS will be tailoring this model to reflect site-specific conditions and plans.

The advantage of using system models is that it allows the disposition of a complex arrangement of waste streams to be evaluated over time. These models comprise a flexible planning tool with the means to rapidly

respond to changing site conditions and/or assess modifications to planning assumptions.

Analysis and "Optioneering"

This part of the planning process begins when a specific site scenario or case is presented for evaluation. The objective of this step is to perform iterative analyses of waste management options in a continuous attempt to drive down the total costs of the waste management system. With each iteration, the waste streams contributing most to the system life-cycle cost are identified. Then treatment, storage and disposal (TSD) options are developed and tested through the life-cycle cost model. The relative differences in cost, risk, and/or uncertainty between options are documented.

At this stage, knowledge of the physical, chemical, and radiological characteristics of the waste streams, and understanding of the appropriate waste acceptance criteria is of key importance. Each waste stream option must satisfy the acceptance criteria of the intended receiving facility.

The advantage of this approach is that allows waste planners to focus on optimizing a detailed waste management strategy for the preferred ASAP alternative. The waste strategy functions as a "living" document, subject to continuous review and refinement.

WASTE MANAGEMENT STRATEGY AND KEY ISSUES

The overall policy for the management of RFETS wastes is storage onsite followed by treatment in preparation for either offsite or onsite disposal. The strategies for individual waste streams within this policy are considerably more complex and driven by specific technical, economic and risk factors. The extent and approach to waste treatment and onsite disposal are still under review at this time; however, as the LCWMP continues to evolve, the objective is to ultimately declare a strategy for all waste streams from the point of generation to final disposal.

RFETS generates and/or stores six major categories of radioactive and nonradioactive waste. Radioactive types include plutonium residues, transuranic (TRU), and low-level waste (LLW). The nonradioactive categories are hazardous, municipal solids and other regulated wastes such as asbestos-containing material, polychlorinated biphenyls (PCBs), and infectious medical waste. The radioactive wastes have their mixed waste counterparts, i.e., those waste forms that are both radioactive and bear a hazardous constituent or characteristic. The only major category that will not be generated in the future is residues, which are byproducts from weapons production and were once held in reserve for plutonium recovery.

Annual waste generation is currently ranges between 1,500 and 2,000 m3. As cleanup and decommissioning activities accelerate, large quantities of waste are expected. Annual volumes may total as high as 50,000 m3 per year from 1998 through 2002. Over the next 10 years, the largest sources of waste will stem from processing/stabilization of residue materials, remediation of contaminated areas throughout the site, and decommissioning of former production facilities.

Waste Storage

One of the key waste management problems at RFETS has been the ability to supply adequate storage space to support continuing operations. RFETS has accumulated over 20,000 m3 of waste in storage , mostly due to lack of access to appropriate treatment and disposal facilities. Wastes are stored in a variety of locations throughout the site and under a variety

of different conditions. Radioactive and nonradioactive waste is currently stored in many former plutonium production buildings; in steel framed, warehouse-style buildings; in cargo containers; and in large tents on asphalt pads.

The primary issues relating to life-cycle planning for waste storage are:

- Determination of future storage capacity requirements: this is a complex function of waste generation rates and the capability to permanently dispose of such waste
- Storage facility configuration: ensuring an appropriate and cost-effective mix of facilities are available to safely store wastes of a variety of plutonium concentrations; i.e., from less than 1 g/drum to greater than 200 g/drum
- Storage time frames: longer storage requirements favor the construction and operation of new storage facilities vs. utilization of existing buildings

Waste storage strategies for the next 2-3 years are targeted at increasing operational efficiencies and reducing overall storage costs. Waste Management is consolidating storage units as much as possible and reducing the quantities of waste stored in former plutonium processing facilities. New capacity to cover near-term storage needs is being added by retrofitting existing facilities that were idled when the site's mission changed. An example of this is Building 440, formerly used to manufacture safe, secure transport trailers, is under conversion to a low-level waste storage and shipping facility.

Strategies to cover long-term waste storage requirements continue to be reviewed at this time. In general, the style of facilities used to store low-level and most transuranic waste will be pre-engineered, Butler-type buildings due to their relatively low cost of construction and operation. Specific plutonium content limitations will be placed on these facilities to ensure the safety and exposure protection to workers and the offsite public. It is probable that, because of high plutonium loadings, some transuranic wastes will require storage in a significantly more hardened and protective facility. This building would be a reinforced concrete structure with high-efficiency particulate air filtration.

The storage capacity requirements for radioactive wastes are dependent on projected rates of generation and the availability of disposal sites under the ASAP site alternatives. Options range from utilization of existing facilities to the consolidation of all wastes into new, large storage buildings.

Waste Treatment

Current waste treatment strategies are published in three primary documents: the RFETS Sitewide Waste Water Treatment Strategy; the RFETS Proposed Site Treatment Plan, which addresses low-level and transuranic mixed wastes; and the Site Integrated Stabilization Management Plan , which covers liquid and solid residues. Many wastes at RFETS require various forms of treatment to 1) remove conditions that could compromise the integrity of the waste container and/or jeopardize worker safety; 2) satisfy Resource Conservation and Recovery Act (RCRA) Land Disposal Restrictions (LDR) treatment standards prior to disposal; 3) meet certain waste acceptance criteria imposed by the disposal facility; 4) provide for more efficient (and cost effective) waste packaging; or 5) provide a means to recycle/reuse the waste material.

The treatment requirements for various RFETS waste categories are shown in Table I. These requirements reflect current planning scenarios and the respective disposal site waste acceptance criteria.

Table I

The expected future cost of solid waste treatments has prompted a reevaluation of the expected benefits to be derived from such activity. Much of the low-level mixed (LLM) waste inventory at RFETS could be considered low risk to human health and the environment if disposed of properly. Waste treatment strategies are still under review at this time; however a variety of options are under consideration including no treatment; minimal treatment to LDR standards based on the inherent risks of the waste; and full treatment of LLM waste to LDR. The minimal treatment approach reviews the physical and chemical characteristics of each waste and proposes treatment for only those streams that represent a significant health hazard to the public if disposed of untreated.

As treatment requirements for TRU and TRU mixed (TRM) wastes are established by the WIPP WAC, two primary options are up for consideration. The first option would propose minimal treatment of TRU/TRM waste forms to facilitate safe storage onsite for a period of 20 to 30 years. The second option would offer treatment of current and future TRU/TRM waste to guarantee availability for shipment to WIPP as resources allow. With the second option, there is an economic incentive to combine TRM and LLM treatments in common treatment systems, if the decision is made to locate these facilities at RFETS.

The site's process waste water treatment facility, located in Building 374, has been in service since 1978. This facility is operating beyond its original design life and several systems are either inoperable or marginally operable. Major upgrades are required to restore its original capacity and capability. Following an evaluation of the site's waste water treatment systems, RMRS proposed an alternative strategy for a low-cost, safe, and versatile waste water treatment system for the future. RMRS recommended that existing precipitation and evaporation systems in Building 374 be closed in favor of smaller, more cost-effective units. This strategy will save approximately $58 million in planned capital expenditures and approximately $60 million in operating and maintenance costs over a 10-year period.

Waste Disposal

Life-cycle waste management planning assumes that all wastes at RFETS are eventually placed in approved, final disposal facilities. While municipal solid waste is the only type currently disposed onsite, a number of onsite and offsite options are under consideration for other waste categories. The overall waste disposal strategy is still under review at this time but could incorporate both onsite and offsite disposal facilities. Disposal options, as well as long-term, monitored retrievable storage, continue to be evaluated with regulatory agencies and affected stakeholders.

Proposed onsite disposal options include waste emplacement in either concrete-lined cells or in conventional-style landfills. Either facility would be designed to handle LLW/LLM waste; small quantities of specialty wastes such as radioactive PCBs and asbestos; demolition debris; and other municipal solid wastes. These facilities would be permitted and designed to meet RCRA Subtitle C standards in terms of protective measures and environmental controls installed to prevent contaminant migration. RFETS is currently constructing a new municipal solid waste

landfill which is expected to meet the site's needs for the foreseeable future.

RFETS is currently utilizing offsite disposal facilities for low-level and low-level mixed wastes and will likely continue this practice in the future. Table II lists the current and potential disposal destinations for wastes along with acceptance dates used for planning purposes.

Table II

SUMMARY AND CONCLUSIONS

This paper presents the approach taken by RMRS to develop a Life-Cycle Waste Management Plan and discusses the status of an emerging waste strategy at the RFETS. The planning effort is currently in progress and initial results of life-cycle cost modeling should be ready by late March or early April, 1996. Future technical papers will explore the outcome of the waste strategy in considerably more detail.

As Rocky Flats takes on the challenge of accelerating site closure, it is imperative that the waste management system be ready and capable of supporting the effort. RMRS is meeting the challenge through a life-cycle approach to waste planning and operations. This methodology will provide continuous reduction in the unit cost of waste operations, freeing up the site's limited operating budget to concentrate on high priority, risk-reduction activities.

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CONTRACT REFORM: IT'S WORKING AT FERNALD*

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ABSTRACT

DOE's contract reform initiatives at Fernald and the performance-based system DOE is now using to evaluate FERMCO are key elements to the current and future success of DOE and FERMCO at Fernald. Final cleanup of the Fernald site is planned for completion by 2005 per an accelerated 10-year remediation plan which has been approved by DOE and endorsed by the U.S. EPA, Ohio EPA, and the Fernald Citizens Task Force. Required funding of approximately $276 million plus inflation annually for 10 years to accomplish final cleanup is now being considered by U.S. Congress. Contract reform initiatives and modified performance measurement systems,

along with best business practices, are clearing the path for the expedited cleanup of Fernald.

MODIFIED CONTRACT BOOSTS "ON-THE-GROUND" REMEDIATION

When DOE and FERMCO signed a significant modification to the contract in July 1994, it represented the first significant action under DOE's contract reform initiatives nationwide. In effect, it was the beginning of what is now referred to as performance-based contracting.

The modified contract represented a significant departure from the management and operating (M&O) type contract that DOE has traditionally awarded at other sites. DOE wanted a contractor that was project-focused. The modified contract provides FERMCO a financial incentive to manage the environmental remediation process as efficiently as possible. Unlike M&O type contracts, this contract requires FERMCO to accept financial responsibility for its actions at Fernald, including any fines or civil penalties that might arise from FERMCO's own negligence. In return, FERMCO is granted more authority to make aggressive decisions about remediation methods.

Rapidly shifting political and business environments have brought about a number of changes in the post Cold War era. These changes are driven by dramatically reduced federal budgets that require work to be completed faster, better, and at less expense to taxpayers. DOE and FERMCO have formed an effective partnership that is resulting in streamlined environmental remediation efforts at Fernald. To be sure, contract reform is an integral part of DOE's response to President Clinton's mandate to "reinvent government."

PERFORMANCE-BASED FEE A POWERFUL MANAGEMENT TOOL FOR DOE

The goal of both DOE and FERMCO was to make the bonus fee performance-oriented, creating a system that would establish objective, quantifiable criteria as the basis for fee determination. The centerpiece of the contract modification was the replacement of the old Cost Plus Award Fee (CPAF) model -- widely used in the DOE system at the time -- with a fee plan based on quality and performance.

Under the new performance-based fee system, FERMCO and DOE agree upon a set of specific, measurable project goals for every six months. FERMCO earns bonus fee only when it exceeds those goals; satisfactory achievement of Performance Objective Criteria (POC) by itself is simply expected and no longer earns any fee.

NUMERICAL GRADING SYSTEM

Each of the three major performance areas are given ratings of excellent, satisfactory or unsatisfactory. Unsatisfactory performance results in a negative numerical rating; satisfactory performance results in a numerical rating of zero, and excellent performance receives a positive numerical rating. Each six-month evaluation period, FERMCO's performance-based fee depends on the sum of the numerical ratings. If FERMCO receives an overall rating of zero, no performance-based fee is earned. If a positive numerical rating is earned, FERMCO receives a performance-based fee. If a negative number results, FERMCO receives no performance-based fee and has to pay back some of the fee the company previously was paid.

The performance-based fee system provides an opportunity for DOE evaluators to align project objectives with FERMCO managers at the beginning of each six-month fee period. FERMCO is then rewarded on its ability to meet those objectives. This is a prime example of a win-win situation for both DOE and FERMCO.

For example, early last year DOE placed heavy emphasis on neutralizing more than 200,000 gallons of uranyl nitrate hexahydrate, or UNH, which basically is uranium dissolved in nitric acid. It was an intermediate compound in the former uranium recovery process at Fernald. UNH had been an ongoing concern for Fernald management, regulators and stakeholders, primarily due to the poor condition of the tanks in which the material was stored. In the first half of FY 95, FEMRCO missed a milestone for startup of the UNH processing. Successful completion of the UNH neutralization project became part of FERMCO's performance-based fee plan for the second half of FY95. FERMCO assigned a special team to the task, and the UNH project was successfully completed several weeks ahead of the regulatory deadline.

While the performance-based fee system is designed to reward FERMCO for achieving excellent performance, it can result in forfeiture of base fee if FERMCO fails to meet minimum performance requirements. Funds available under the performance-based fee structure are divided into two main categories: General Contract Performance Criteria and Milestone Completion.

GENERAL CONTRACT PERFORMANCE CRITERIA

General Contract Performance Criteria are elements of the Fernald work scope that directly affect our mission of a safe, least-cost, early cleanup of the site. These criteria are further subdivided into three main sections: Safety and Health, Environmental Management and Administrative Management. The three levels of performance are "excellent," "satisfactory" and "unsatisfactory." The objective measures which determine the performance level, and the amount of fee, are assigned to a particular criterion.

For example, the DOE Fernald Area Office may consider shipping one thousand drums of mixed waste to be excellent performance, for which FERMCO would receive two percent of the available fee. Shipping only 800 drums of mixed waste would reflect satisfactory performance, but FERMCO would receive no fee. Shipping any fewer than 800 would be considered unsatisfactory performance, and FERMCO would have to refund two percent of the performance-based fee back to DOE. FERMCO could be required to refund up to 25 percent of its base fee for unsatisfactory performance.

While most of the General Contract Performance Criteria contain these objective performance measures (excellent, satisfactory or unsatisfactory), a few of them have been reserved for DOE's subjective evaluation of FERMCO's performance. In these cases, DOE provides a description of the criterion without any specific performance measures attached to them.

MILESTONE COMPLETION

The Milestone Completion section includes activities that are considered high-risk, and most of them are included in the Amended Consent Agreement between the DOE and U.S. EPA. Milestones have performance measures for "achieved excellence" and "did not achieve excellence" because nearly all of the milestones are based on deliverable dates. That is, FERMCO either met the due date or it did not. FERMCO is not required to refund fee to the DOE for failing to achieve excellence under the Milestone Completion section, but FERMCO is still potentially liable for fines levied by EPA for missed deadlines.

PERFORMANCE AREAS, CRITERIA AND MILESTONES USED FOR EVALUATION

It is important to keep the Fernald mission statement in mind when reviewing the major performance areas negotiated between DOE and FERMCO to assess FERMCO's performance against established criteria:

Together, DOE and FERMCO are committed to protecting human health and the environment through the safe, least-cost, earliest, final cleanup of the Fernald site, within applicable DOE order, regulations and commitments, and in a manner which addresses stakeholder concerns.

FERMCO's performance is assessed every six months of the fiscal year, covering periods from October 1 through March 30 and April 1 through September 30. FERMCO is evaluated on three elements of the mission statement: 1) Safe Cleanup, 2) Least-Cost, Earliest and Final Cleanup, and 3) Addressing Stakeholder Concerns.

Under the Safe Cleanup performance measure, FERMCO is expected to significantly reduce the probability of accidents, exposures or releases from occurring. While DOE recognizes that an effective Safety and Health program cannot prevent all accidents, exposures or releases, FERMCO's Safety and Health programs are expected to identify, categorize and control deficiencies in a timely manner to prevent them from happening again.

In addition, FERMCO's Radiological Protection Program is evaluated on how well it keeps exposure and contamination incidents to workers and the public at a minimum. Therefore, employees must be aware of workplace hazards and safety precautions that can prevent accidents from happening. That is why Safety First initiatives and the Voluntary Protection Program are heavily weighted in evaluating FERMCO's performance.

The Least-Cost, Earliest, Final Cleanup performance measure address the Project Management Control System, which is used to integrate technical, cost and schedule performance data. Success of removal actions and waste packaging and shipping also are considered under this heading. This category also considers FERMCO's ability to improve procurement processes and develop technology programs.

FERMCO also is evaluated on its ability to address stakeholder concerns. FERMCO is expected to comply with requests from the public in a timely manner, and to provide information about project plans at the site to employees and the public. Programs evaluated under this heading include the envoy program, media relations, employee communications, support of the citizens task force, and the success of the Community Relations Plan which is designed to inform stakeholders about public involvement opportunities.

CRITICAL FEW PERFORMANCE MEASURES

DOE and FERMCO recently agreed on a list of "critical few" additional performance areas to be measured. FERMCO will be evaluated under the Program Progress heading for its ability to conduct legacy waste characterization and low-level waste shipping; minimize effluent discharges to the Great Miami River from wastewater treatment systems; achieve design milestones and implement field work in support of EPA-approved Records of Decision, and remove holdup material from equipment and lines under the Safe Shutdown program. FERMCO also is expected to demonstrate a Return on Investment by reducing maintenance and electric utility costs and transferring or disposing of excess government property; measure stakeholder opinions and views about Trust and Confidence in a process to be conducted by independent third parties with results reported to DOE; and meet scheduled commitments for rebaselining

the 10-year remediation plan, upgrading the Progress Tracking System list and successfully completing Project Tracking System milestones.

WHERE WE ARE TODAY

The nature and extent of contamination has been defined, and the selected remedies have been well researched and they're supported by DOE and its stakeholders.

We have a plan and now we are baselining our schedule milestones and developing detailed cost estimates for each of the five areas targeted for remediation at Fernald. It is an aggressive plan designed to expedite cleanup and significantly reduce or eliminate costs associated with overhead, landlord activities, and building maintenance. A concerted effort is being made to spend as little money as possible supplying heat,electricity and maintenance support to buildings targeted for demolition.

We have realized significant cost savings, cost avoidances, and greater efficiency by aligning technical project needs with regulator and stakeholder interests, thereby reducing the obstacles that drive costs up. The modified contract incentivizes FERMCO to produce "on-the-ground" remediation progress. FERMCO also has strategically aligned project organizations which are completely focused on planning and executing final remedial activities. The project concept works well at Fernald, because start-and-finish timelines exist and the work scopes are easily measured.

DOE and FERMCO developed this accelerated cleanup plan in conjunction with federal and state regulatory agencies and Fernald stakeholders to achieve the most cost-efficient, aggressive approach to cleanup without compromising safety principles or regulatory guidelines. This can be accomplished within 10 years. The focus has been to create a dedicated project team that will be disbanded in a programmed manner as each phase of the job is finished.

Under the accelerated cleanup plan, as work progresses we will continually collapse and consolidate radiologically contaminated zones. They will get smaller and smaller and fewer in number until there are no more.

The U.S. EPA, Ohio EPA and Fernald stakeholders agree with DOE and FERMCO that an accelerated cleanup approach is well-timed, doable, and fiscally responsible, saving taxpayers an estimated $2.7 billion in escalated funds. We are moving forward in an aggressive manner.

Session 36 -- IMPROVED TECHNIQUES FOR LIQUID LLW

Co-chairs: Clint Miller, PG&E;
Lloyd McClure, Lockheed-Martin Idaho Technologies Co.

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AN OVERVIEW OF RECENT DEVELOPMENTS IN THE USE OF CHEMICAL ABSORBING MEMBRANES FOR RADIONUCLIDE SEPARATIONS

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ABSTRACT

The U.S. Department of Energy (DOE) will rely extensively on efficient chemical separations technologies to treat radionuclide containing waste left behind from nuclear weapons production. The waste varies in composition over extreme ranges of pH from acidic to basic. A new approach uses technology which has been developed to incorporate almost any sorbent particle into membranes and then fabricate these into versatile, highly efficient chemical absorbing cartridges.

This is particularly attractive because the process is based on selective sorbent technology instead of traditional wet chemical extractions, chemical precipitations or large ion exchange columns. It provides the capability of removing contaminants down to detection levels at high flow rates, if necessary in the presence of high levels of radiation, concentrating the contaminants while generating a minimum of secondary waste. The full system is projected to have a lower capital cost and smaller footprint than conventional technology. In many ways, the process is as simple as filtering the liquid through a cartridge.

DISCUSSION

In early 1989, 3M introduced this technology as an alternative form of solid phase extraction (SPE), which is used to extract dissolved species from water (1)(2). The sorbent is loaded into a web or membrane which is used in a filtration-like process.

Figure 1 shows a typical example of Emporem membranes in which inert PTFE fibrils are used to hold the sorbent particles. Several classes of materials have been successfully incorporated into the 3M membranes, including commercial organic ion exchange materials, inorganic adsorbents, unique zeolyte structures and elaborate macrocycles. The resulting membranes are characterized by high separation efficiencies, radionuclide loading, fast flow rates and kinetics and physical ruggedness. A radiolytically resistant (up to 2,000 megarad) material may be used as the membrane matrix. Because the membrane is very densely packed (up to 95% particle by weight), with small (5-25 mm), high surface area particles the flow rate can often be 10 to 100 times greater than in standard column ion exchange processes while achieving equal extraction efficiencies. Particles of this size would result in unacceptable back pressure if used in columns. Channeling, or wall effects, which can be a severe limitation for columns, is absent in membranes and offers another reason why high flow rates are possible. This membrane separation technology allows the use of a number of known, high performance, chemical adsorbing powders, which previously could not be put into a useful engineered form because of their small particle size (3). It provides the ability to concentrate materials into much smaller volumes than alternative technologies.

Fig. 1

Laboratory scale experiments at Idaho National Engineering Laboratory (INEL) (3)(4) showed that strontium and cesium could be selectively removed from actual radioactive acidic waste with extremely high efficiency even in the presence of relatively high concentrations of other ionic species. The membranes used were 22 mm diameter, 0.5 mm thick filter-type disks with either IBC Advanced Technologies, Inc., Superligm particles for the removal of strontium and lead, or crystalline titanium phosphate (PhTiA) (5) particles for the removal of cesium. The waste composition is shown in Table I.

Table I

Table II summarizes the test results and Figs. 2 and 3 show the breakthrough curves describing the separation of cesium and strontium respectively .
Table II
Fig. 2
Fig. 3
When the membrane is fabricated into a pleated cartridge large volume separations are possible. Cartridge capacity varies according to sorbent type and mass and the composition of the process stream.
In August 1995, Empore cartridge technology was tested at the West Valley Nuclear Services site. The goal of the project was to test the system for removal of technetium (as pertechnetate) and cesium from waste water. The cartridges used were 4" pleated cartridges containing carbon/Aliquat 336m and potassium cobalt hexacyanoferrate for, technetium and cesium removal, respectively. The flow rate used was 0.1 gallons/minute and the test continued for about eight days, or a volume of almost 1,500 gallons. To summarize the results, the technetium cartridge was loaded to a 50% breakthrough after about 90 gallons (see Fig. 4) and the cesium cartridge ran for 40 hours (about 250 gallons) with no detectable breakthrough of cesium. The cesium in the feed was present at 1,200 pCi/L, while the detection limit for the cartridge effluent was about 2 orders of magnitude lower. These results are particularly interesting because the feed had already been processed through traditional ion exchange columns for cesium removal.
Fig. 4
In September 1995, the technology was tested at Idaho National Engineering Laboratory, Test Area North (TAN) (6). The purpose of the demonstration was to evaluate the performance of the cartridges on injection well water which had already been processed through ion exchange columns.
Figure 5 is a schematic of the equipment used at INEL. The flow rate was 0.25 gallons per minute, (i.e. 1,250 gallons were processed over the 80 hour demonstration), using a feed whose composition is shown in Table III. Species to be removed in this case were strontium and cesium using 4" cartridges of, respectively, sodium titanate and potassium cobalt hexacyanoferrate, (Table IV).
Fig. 5
Table III
Table IV
The presence of non-radioactive strontium in the groundwater resulted in a much higher concentration of total strontium, an important consideration since there is no measurable recognition of sorption for different isotopes of the same element. The high total strontium concentration, coupled with absorption of calcium and magnesium, resulted in Sr breakthrough of 50% (breakthrough =C/Co *100%) after approximately 800 gallons of water had passed through the Sr absorber. Concentrations of Sr, Ca, and Mg in the effluent are shown as a function of time in Fig. 6. Calcium and Mg, present as divalent cations at substantially higher concentrations than Sr in the TAN groundwater, compete with Sr for the active sites in the adsorption medium.
Figure 6 indicates that both Ca and Mg breakthrough occurs more rapidly than the Sr breakthrough and implies that Sr is preferentially adsorbed in the sodium titanate cartridge. Analysis of radio-strontium, 90Sr was performed independently on several samples of the effluent during the

course of the experiment. Analytical results for 90Sr are also shown in Fig. 6. The analytical results for total and radio strontium are in excellent agreement and serve to indicate consistency of the two data sets. Note that the "leveling" of the Sr curve at approximately 55% breakthrough may be attributed to displacement of Ca or Mg by Sr, or it may be an unrealistic nuance of the curve-fitting routine. Irrespective of the cause, the Sr curve would eventually climb to a breakthrough of 100% if enough water had been pumped through the cartridge.

Fig. 6

Cesium showed no detectable breakthrough for the for the entire demonstration, with a detection limit of 9 pCi/L. Again it is important to note that this water had already been through the groundwater treatment facility.

CONCLUSIONS

Strontium, cesium and technetium have been successfully removed from radioactive waste using a new and innovative technology. Laboratory scale experiments at Idaho National Engineering Laboratory (INEL) showed that strontium and cesium could be selectively removed from actual radioactive acidic waste with extremely high efficiency even in the presence of relatively high concentrations of other ionic species. Larger scale technology demonstrations, removing trace quantities of technetium (as an anionic species) and cesium from process water at the West Valley Nuclear Services site in New York state and removing cesium and strontium from previously processed injection well water at INEL show the feasibility of applying the technology on an industrial scale.

Additional large volume technology demonstrations are planned involving wastes of different characteristics in terms of contaminants, pH and ionic strength and using a variety of sorbents.

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LIQUID RADWASTE PROCESSING WITH SPIRAL WOUND REVERSE OSMOSIS

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ABSTRACT

Two different reverse osmosis systems were investigated in this work. The first was a 50-element plant-scale system that is used to treat 2200 cubic metres of AECL liquid radwastes annually. It uses thin film composite (TFC) membranes and operates at an applied pressure of 2760 kPa, with a fixed crossflow of about 40 L/min. The other system uses the same thin film composite membranes for waste processing, but is a 2-element pilot-scale system. It is operated at pressures ranging between 1500 kPa and 7000 kPa, at a fixed crossflow of 55 L/min.

The average lifetime of the thin film composite membranes in the plant-scale processing application at AECL is about 3000 hours. After this service life has expired the rejection efficiency (for total solids) declines rapidly from 99.5% to about 95% as the membranes become impaired from chemical cleaning procedures that are required after each hundred cubic metres of waste are treated. The permeation flux for the plant-scale system decreases from about 2.2 L/min/element to below 0.5 L/min/element at the end of the membrane's useful service life.

The plant-scale membrane elements, fouled by an assortment of chemicals including calcium phosphate and various organics, were successfully regenerated by exposing them to a three-step chemical cleaning procedure (in the pilot-scale system), using detergent, HCl, and an alkaline-based cleaning with EDTA. The 3-step procedure was successful in elevating the flux from 0.5 L/min for the spent membrane, to 1.2 L/min after cleaning. The 1.2 L/min post-cleaning flux could be maintained provided that the crossflow velocity remained high.

The DF for cesium for the plant-scale system (at the operating pressure of 2760 kPa), decreased from about 100 when the membranes were new, to about 30 after they were replaced. After cleaning the fouled membranes with the pilot-scale system, the DF for cesium increased from about 30 to 50, if the applied pressure to the system was increased from 1500 kPa to 5500 kPa. By comparison, the strontium DF increased for the fouled membranes at the operating pressure of 2760 kPa, from about 1000 (when they were new), to about 4000 for the spent membranes. The strontium DF was unaffected by the applied pressure. The increase of strontium DF is believed to be due to the exchange of strontium with deposited calcium on the fouled membrane.

INTRODUCTION

In the mid 1970's, AECL, at the Chalk River Laboratories (CRL) site, built a Waste Treatment Centre (WTC) for treating low-level solid and aqueous liquid wastes. The objective was to demonstrate processes for converting Canadian Deuterium Uranium (CANDU) wastes to a form suitable for disposal. The liquid waste streams are effectively volume reduced by a combination of continuous crossflow microfiltration (MF), spiral wound reverse osmosis (SWRO), and tubular reverse osmosis (TRO) membrane

technologies (Fig. 1). Backwash and chemical cleaning wastes from the membrane plant are further volume reduced by evaporation. The concentrate from the membrane plant is ultimately immobilized with bitumen using a thin-film evaporator in a shielded cell. The ability of the MF/SWRO technology to remove impurities non-selectively makes it suitable for the treatment of radioactive effluents from operating nuclear plants, with proper membrane selection, feed characterization, system configuration, and system chemistry control.

Fig. 1

Currently, there are two streams routinely treated at CRL. One originates from the Decontamination Centre (DC waste) and the other collects waste from the Chemical Drain (CD waste) system. The compositions of the two streams are given in Table I. A total of about 2200 m3 of blended liquid (low to intermediate-level) waste are currently treated by the membrane plant annually. The current overall volumetric recovery of the two stage (concentrate-staged) reverse osmosis train employing SWRO and TRO is 96.6%. For a 35 m3 batch of waste treated there is 0.7 m3 of backwash concentrate and 1.17 m3 of TRO concentrate sent to the evaporator for further volume reduction in a small evaporator, and subsequent immobilization with emulsified bitumen in a thin-film evaporator. The bituminized product from the facility occupies 0.175 m3. The immobilization of the concentrate with emulsified bitumen has been described in a companion WM'96 paper (1). The overall volume reduction of the fresh feed through the integrated plant (employing both membranes and evaporation) is about 250.

Table I

SPIRAL WOUND REVERSE OSMOSIS SYSTEM

Reverse osmosis (RO) is a technology which is well established for the production of potable water from brackish water or seawater. It has been successfully used to produce high-purity water for the electronics, pharmaceutical, and power industries. Because of its versatility to remove ionic impurities, particulates and colloids, organics, microorganisms and pyrogenic material from water, RO has attained a prominent role in water purification (2).

In spiral wound reverse osmosis treatment, feedwater containing dissolved and suspended solids is pumped into the system at a desired feed pressure greater than the osmotic pressure of the solution. The feed stream is pumped into a pressure vessel containing one or more membrane elements connected in series. The feedwater then flows into the channels between the membrane sheets. These feed channels are composed of a plastic netting which breaks up the flow into small turbulent areas above the membrane surface.

Immediately above the membrane surface a concentration boundary layer forms whose thickness depends on feedwater ionic strength, particulate level, and flow in these small turbulent areas. The water and ions are transported by a solubility-diffusion process to the permeate water carrier. The purified water or permeate is recovered at atmospheric pressure. The remaining water, dissolved solids and particulates form the reject stream. The pressurized concentrate or retentate is dropped to atmospheric pressure through a back pressure regulating valve, immediately downstream of the system.

The performance of an RO membrane is usually described in terms of permeate flow, or "flux", contaminant rejection efficiency, and volumetric recovery. Permeate flux refers to the amount of flow across

the membrane per unit area, at a particular operating pressure and dissolved solids concentration. The flow of water across the membrane is proportional to the effective pressure (applied pressure minus pressure drop minus the osmotic pressure of the solution). Increasing the applied pressure will increase the permeate flow without increasing the solute flow.

Rejection is the relative change in contaminant concentration from the feed stream to the permeate stream. RO membranes are not absolute barriers, and some small percentage of the solute (typically about 0.5%) does pass through the membrane. The amount of solute transport is a function of the membrane type and is proportional to the differential concentration across the membrane.

Spiral wound RO membrane types can be broadly classified as cellulosic or noncellulosic. Cellulose acetate membranes are still widely used because of their resistance to fouling, and their low cost. They are, however, easily damaged by bacterial attack and have relatively low rejection efficiencies. Noncellulosic membranes, such as the thin film composite membranes used at CRL, can operate over a wider pH range and exhibit high solute rejection efficiencies. The SWRO configuration achieves a large specific surface area per unit volume, which is typically 1000 m2/m3. This can be compared to 165 m2/m3 for plate modules and 335 m2/m3 for tubular modules.

The ratio of permeate to feed in a RO system is referred to as volumetric recovery, or simply, "recovery". To achieve high recoveries (up to 85% currently employed at CRL), it is necessary to stage the concentrate stream or recycle the concentrate for reprocessing. This is normally accomplished in a tapered system design. For instance, a two-stage system may have four pressure vessels in the first stage, feeding two vessels in the second stage. The tapered configuration compensates for feed flow loss by permeation, therefore maintaining optimum cross-flows in both stages.

Plant-Scale SWRO System Description

The CRL plant-system is a three-stage 5:3:1 tapered system with 10 cm diameter by 6 m long pressure vessels. Each pressure vessel contains six membrane elements. The system is fed with a Goulds 3333 multi-stage centrifugal booster pump. Typical feed crossflows are maintained at about 40 L/min, with an inlet pressure of 2760 kPa.

Pilot-Scale SWRO System Description

The pilot-scale system is equipped with a 5 mm cartridge filter upstream of the two high pressure pumps, both of which can deliver a maximum of 7000 kPa. The pressurized feed flowing at about 55 L/min is directed to one of two membrane pressure vessels, each containing two 10 cm diameter by 100 cm long membrane elements. The purified permeate stream exits the vessel, then passes through a flowmeter, and can be rejected from the system or recycled back to the feed tank. The majority of the retentate (concentrate stream with 99% of the contaminants in the feed), recirculates back to the suction side of the high pressure pump to maintain the high crossflow rate. The remainder of the retentate is directed back to the feed tank. The operating pressure of the system is manually controlled by adjusting the backpressure control valves on both retentate streams leaving the vessel. A system is provided to flush the vessel and piping with clean water after an experiment. An onboard cleaning tank is also provided if chemical cleaning of the membranes is required.

The primary difference between operations of the pilot-scale and the plant-scale system was the effective crossflow rate through each vessel; for the pilot-scale system it was 55 L/min/vessel, while for the plant-scale system it was about 40 L/min/vessel. A more subtle difference between the units is the actual processing method to achieve the desired volumetric recovery of between 85 - 95%. For the plant-scale system retentate was recirculated to the feed tank, and a small bleed fraction was continuously removed. Filtrate was continuously added to the feed tank from the MF system, and so the operation was continuous. By comparison, the pilot-scale rig was operated in a batch mode of operation where a given volume of fresh MF filtrate was volume reduced to the predetermined volumetric recovery.

PERFORMANCE OF THIN FILM COMPOSITE MEMBRANES FOR LIQUID WASTE PROCESSING

Normalised Permeation Flux

The permeation flux from an RO system is a function of several variables including temperature, pressure, and pH. The plant-scale system used 50, 40-inch elements in nine pressure vessels, which were staged as 5:3:1. The permeation flux data obtained was normalised to an applied pressure of 2.76 MPa and a feed temperature of 25oC using a method given by Bukay (1984) (3). The observed permeation flux (OBF) was normalised for temperature using a temperature correction factor (TCF) obtained from the membrane manufacturer, an effective pressure, and for the total number of elements using Eq. 1.

Eq. 1

From a processing perspective it is desirable to maintain a permeation rate of between 0.80 and 1.0 L/min/element at all times, so that the throughput of the 50-element plant-scale system matches the filtrate production rate of about 42 L/min from the MF system situated upstream. In the continuous mode of operation, downtime of equipment is minimized and maximum utility is made of the equipment.

Figure 2 shows the normalised permeation flux for the plant-scale system after liquid processing was initiated. The data shown is a continuation of normalised permeation flux data reported in a paper presented at WM'95 (4). Membranes were changed at about 3800 hours and again after 7000 hours of operation, when the NPF decreased below 0.25 L/min/element. By comparison, the NPF for the pilot-scale system, which is operated at higher crossflow velocity, is consistently at 2.2 L/min/element. The scatter of the data in Fig. 2 is not due to errors in measurement; rather, it is the result of permeation flux declines during a given run, and the subsequent recovery after a chemical cleaning with an appropriate solvent. After a chemical cleaning it was possible to have up to a 100% improvement of normalized permeation flux or more, although this improvement of performance could not be maintained.

Fig. 2

When the membranes were replaced after 7000 hours of operation (Fig. 2), several precautions were taken to minimize the large flux loss observed at 3800 hours. These included: i) operating at a volumetric recovery 5% lower than for the previous set which was processing to 85%; ii) using recommended anti-scalant chemicals in the feed solutions and; iii) cleaning at more regular intervals. Yet, none of these preventative measures appeared to be of much benefit in the resulting loss of throughput that was observed. Regular chemical cleaning cycles appeared to decrease the pressure drop across the membrane plant, but had less impact on improving permeation flux than similar chemical cleanings

performed on the previous membrane set. After about 7000 hours of cumulative service the membranes were replaced due to i) the deterioration of membrane rejection efficiency, and ii) inefficient plant operation due to the frequency of equipment shut-downs from excessive chemical cleaning requirements.

Rejection Efficiency of TFC Plant-Scale Membranes

Another performance indicator that can be used to assess the RO membranes is the overall removal efficiency of conductive ions. The many aggressive chemical cleanings that were required to restore the permeation flux (Fig. 2) may have had a detrimental effect on the membrane integrity, resulting in the considerable loss of membrane rejection performance. Physical abrasion of the polyamide rejecting layer over time may also have been a contributing factor.

A plot of the conductivity rejection efficiency versus the elapsed processing time in the plant-scale system is shown in Fig. 3. The feed conductivity in the present application ranged between 100 and 1000 mS/m, while the permeate conductivity varied between 0.6 and 20 mS/m. Figure 3 is a continuation of rejection efficiency information reported previously at WM'95 (4). The membranes were replaced after 3000 hours because the flux decreased to very low permeation throughputs (Fig. 2), and the rejection efficiency decreased to values below 95% (Fig. 3). A rejection efficiency of 95% indicates that the thin film composite polyamide membranes have degraded to a point where they were no longer useful for the specific processing.

Fig. 3

After the membranes were replaced at 3000 hours (Fig. 3), the rejection efficiency increased back to 99.5% overall. The rejection remained at about 99.5% until about 7000 hours of operation, when the overall efficiencies again declined back to about 95%, and the membranes were subsequently replaced. For those runs between 3000 hours and 7000 hours where the removal efficiency was lower than 99%, leakage took place of excessive nitrate and sodium ions into the permeate following a chemical cleaning. Monovalent ions have low rejection efficiencies (5). Since the plant feed would have been temporarily overloaded with monovalent ions, this situation resulted in a reduced overall rejection efficiency. After the cleaning chemicals were flushed from the system, the rejection efficiency was restored to about 99.3% overall.

Cause of Flux Decline in Plant-Scale System

The flux decline with time observed in Fig. 2 at 3800 hours and 7000 hours, is the result of concentration polarization and/or surface fouling. (6) Surface fouling occurs when there is deposition of submicron particles on the surface, as well as crystallization and precipitation of smaller solutes. It is manifested when rejected solids are not transported from the surface of the membrane back to the bulk stream. In general, there are five types of fouling: namely membrane scaling, fouling by metal oxides, device plugging, colloidal fouling, and biological fouling (7). The different types of fouling frequently occur at the same time and can influence each other. Unfortunately, the interactions between the types of fouling are poorly understood.

The chemical scale on the fouled CRL reverse osmosis membranes is comprised primarily of aluminum, silica, calcium, phosphorous, and to a lesser extent iron and sulphur (4). Calcium hydroxylapatite and octacalcium phosphate scale have been identified as a major fouling species. Clay and aluminum silicate-based scale is thought to account for

the initial large flux decline associated with concentration polarization, and represents the scale which is the most difficult to remove by standard chemical cleaning procedures. Precipitates approaching 10 mm in diameter have been observed on the surface of the membrane, and these precipitates are hard to dissolve in even the most concentrated acids. This is further supported by the observation that alkaline cleaning chemicals at pH 12 are the most effective for permeate flux restoration where silica solubilizes to silicic acid (8). The precipitates in the 50 membrane elements of all three SWRO stages are similar in chemical composition.

While operating, thin film composite membranes take on an anion charge on the surface. This causes cationic foulants, such as aluminum and ferric hydroxides, along with cation coagulant polymers, to be attracted to it. Further, because of the high flux rates on thin film composite membranes, fouling occurs faster and is more noticeable than with other membranes (9). Moreover, in the CRL plant-scale application the volumetric recovery is maintained at about 85%. This high recovery leads to increased solute concentrations at the boundary layer, leading to more concentration polarization and a higher deposition of fouling substances.

Performance of Pilot-Scale System

The primary difference between operations of the pilot-scale and the plant-scale system is the effective crossflow rate through each vessel; for the pilot-scale system it was 55 L/min/vessel, whereas for the plant-scale system it was about 40 L/min/vessel. To compare the performance of the two systems, the fresh feed to the pilot plant was the same 50:50 blend of the CD/DC waste streams. The pilot-scale system was run in parallel with the plant-scale system, and both of the systems were targeted to achieve 85% volumetric recoveries overall. By comparison with the plant-scale system, which was only able to maintain 0.5 L/min/element (at the start of a processing campaign), the pilot-scale system was able to sustain a permeation rate of 2.7 L/min. During the course of a run, the flux typically decreased from 2.7 L/min to about 1.7 L/min, depending upon the final volumetric recovery that was sought. However, the loss of flux during a run was always reversible (after flushing the system with permeate), indicating it was due to concentration polarization, and not irreversible fouling. If the system was placed in total recirculation at any time during the run the permeate flows would not decrease further.

Regeneration of Spent Plant-Scale Membrane Elements with Pilot-Scale System

Since the performance of the pilot-scale system was so much superior to the plant-scale system for CRL liquid waste processing applications, a test was undertaken to evaluate the cleaning effectiveness for spent plant-scale membranes in the pilot-scale system. The spent membranes were removed from the plant-scale system at 7000 hours (Fig. 2), and two were chosen randomly for cleaning tests.

A chemical cleaning campaign of a fouled plant-scale thin film composite membranes was carried out using a three-step procedure at an applied pressure of 2760 kPa (the same operating pressure as the plant-scale system). The fouled membrane was taken from the first stage of the three-stage system. During the first cleaning step a detergent formulation (1 wt.%) was used to remove surface oil and grease. In the second step, HCl (at a pH of 2 and ambient temperature), was employed for the dissolution of metals. In the third step, NaOH (at pH of 12) and 2 wt.% EDTA was used to complex metals and remove silica deposits by dissolution.

A 73.5% improvement of permeability was observed after cleaning with the detergent. A total of 0.8 kg of oil and grease was removed from the element with the detergent cleaning process. After the second cleaning procedure with HCl, a further improvement of 14% was observed. After the third step where chemical cleaning was carried out with NaOH and EDTA, an incremental improvement of 20% was noted for the element. The overall improvement for the element based on the three-step cleaning sequence was 137%; the water permeability increased from 0.48 L/min before cleaning to 1.14 L/min after cleaning. The same cleaning tests were carried out on another fouled element, and the results could be replicated.

A complete regeneration of the fouled membrane elements was not achieved. The permeation flux for a new element was about 2.2 L/min (Fig. 2), and hence the pilot-scale cleaning procedure was capable of restoring about 50% of the membrane's original performance. The cleaned elements could be reused in the plant-scale system at a throughput of 1 L/min, however, because this would match the filtrate production from the MF system situated upstream (Fig. 1).

Following the 3-step cleaning procedure, a series of three further experiments were carried out in which blended CD/DC MF-filtered waste was processed in the pilot-scale unit to evaluate the long term benefit of the cleaning methods. Results are shown in Fig. 4, for the flux curves before cleaning and after cleaning (represented by Tests 1,2,3). In the first experiment a batch of CD/DC waste was passed through the system with a 5 micron prefilter in place. The feed was reduced to a final recovery of 90%. After flushing the system with permeate at the completion of the run, the same test was replicated (Test 2). The purpose of Test 2 was to determine if the permeate flux after a pure water flush could be restored to the same initial value (which had previously been observed for CD/DC waste processing). Finally, a third test was carried out in which the 5 micron prefilter was removed from the system to determine if a coarse filter was required upstream of the membranes at the higher crossflow velocities.

Fig. 4

At a volumetric recovery of 90% in Test 1, the permeation flux decreased to 0.88 L/min. The flux decline curve was identical for Test 2, indicating that the performance curve was reproducible. It also provided evidence that the chemical cleaning procedures had been effective in displacing scaling materials. In Test 3, where the 5 micron prefilter was removed from the system, the permeation rates at all volumetric recoveries mimicked the previous two tests. The results of Test 3 indicate that a coarse 5 micron prefilter is not required upstream of the RO system.

Pure Water Permeability after Spent Membrane Regeneration

A water permeability test was carried out before and after the chemical regeneration procedures, so that comparisons of the cleaning efficiency could be made on the basis of membrane permeability. Pure deionized water was passed through the element at various applied pressures ranging between 1500 kPa and 5500 kPa, and the permeation flux was recorded as a function of the operating pressure.

The results of the permeability tests for the two different membrane elements are shown in Fig. 5. Prior to the chemical cleaning procedure, the flux for element #1 varied between 0.25 and 0.9 L/min/element, as the pressure increased from 1500 kPa to 5500 kPa. After cleaning the flux for element #1 increased to between 0.6 and 2.2 L/min/element over the same

pressure range. A similar improvement of pure water permeability was observed for element #2 after the cleaning procedures. The pure water permeabilities for both elements after the chemical cleaning procedures were approximately the same (at all pressures), even though element #1 was more fouled. This would indicate that the membranes were cleaned to the maximum possible extent by the three-step cleaning procedure, which is about 50% of the new membrane's performance.

Fig. 5

Effect of Pressure on Decontamination Factor

Tests were carried out to evaluate the impact of the chemical cleaning procedures on the removal efficiencies of cesium and strontium. The results are plotted in Fig. 6 in terms of a decontamination factor (DF), rather than a rejection efficiency for clarity on the figure. The decontamination factor (DF) is defined by Eq. 2.

Fig. 6

Eq. 2

It was necessary to use stable isotopes of cesium and strontium instead of radioactive isotopes to determine the DF of each accurately. This was required since the resolution of the analytical instruments were not sufficient to detect radioactive contaminants reliably at low concentrations in the permeate. The results of these tests showed that the DF before cleaning for cesium was 30 (at 1500 kPa), and increased to 50 (at 5500 kPa) before cleaning. However, the cesium DF for a new element was about 100, which indicated that there had been some permanent loss of rejection after its 3000 hour service life was over. There was no increase of cesium removal after the three-step cleaning procedure with the pilot-scale system. The tests demonstrated that the aggressive chemical cleaning carried out did not impair the membrane'a ability to remove contaminants.

The DF for strontium increased from about 1000 for a new membrane element to roughly 4000, after the membrane had been exposed to about 3000 hours of liquid waste processing. The strontium DF was relatively independent of both the applied pressure and the chemical cleaning procedures. The increase of DF after waste processing can be rationalized by noting that the dominant scale on the plant-scale membranes is calcium phosphate. It is probable that strontium replaces the precipitated calcium on the fouled membrane (as strontium phosphate), liberating calcium in the process. Evidence of this phenomenon can be gleaned from the contact beta radiation fields on the fouled membranes. Following their removal from the plant-scale system at 7000 hours (Fig. 2), the fouled membranes had contact beta fields that sometimes approached 20 Rads/h. The high beta field is evidence of Sr-90 deposits; there are no other abundant pure beta emitters in the CRL waste streams.

CONCLUSIONS

Membrane replacements are required after approximately 3000 to 4000 hours in the plant-scale liquid waste processing application at AECL. After this period, the average permeation flux for the 50-element plant-scale system decreased below 0.3 L/min/element, and the bulk conductivity rejection efficiency declined rapidly from 99.5% to 95%. By comparison, with the pilot-scale system, where there is 50% higher feed crossflow, it is possible to maintain a permeation flux of about 2.2 L/min/element.

The fouled CRL membrane elements could be regenerated by exposing them to a three-step chemical cleaning procedure (at higher crossflow in the pilot-scale system), using detergent, HCl, and an alkaline-based cleaning

solution containing EDTA. The 3-step procedure was successful in boosting the flux from 0.5 L/min for the fouled membrane element to 1.2 L/min for both elements that were cleaned.

The decontamination factors for cesium and strontium were not affected by the aggressive cleaning procedures in the pilot-scale system. The DF for cesium increased from about 30 to 50 as the applied pressure increased from 1500 kPa to 5500 kPa. However, there was no variation of the cesium DF after the cleaning procedures were completed. For strontium, the DF varied between 3000 to 4000, and was unaffected by the applied pressure. The strontium DF did not change after the chemical cleaning. In addition, the strontium DF increased from 1000 (for the new element) to 4000 for the fouled membrane. The increase is believed to be due to the exchange of strontium from solution with deposited calcium on the fouled membrane surface.

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ACKNOWLEDGEMENTS

Funding for the experimental program was provided through the CANDU Owners Group Low-Level Radioactive Waste Working Party 49. The authors express their gratitude to process operators in the Waste Treatment Centre and to K. Sullivan for their efforts in operating the equipment and collecting data.

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ABSTRACT

Advanced Liquid Radwaste Treatment processes can be defined as those technologies which provide substantial solid radwaste reduction and improved water quality when compared to conventional processes such as demineralization and evaporation. Nine Mile Point - Unit 1 (NMP1) has successfully processed over 6 million gallons of outage and non-outage generated floor drain water through an advanced treatment system since January of 1995. Operating results show that advanced systems are capable of processing widely varying waste water streams while lowering product water impurities and solid radwaste generation rates. This paper relates full scale operating experiences encountered since startup, as well as, feed and effluent chemistry results. Information provided in this paper will specifically benefit two groups of nuclear facilities: 1) those operating under a zero-liquid discharge scenario, where restrictive product water quality standards are imposed, and 2) those facilities that require less than minimum detectable activity (<MDA) concentrations of gamma emitting isotopes in the water discharged to the environment. Specific operational techniques used to maintain Total Organic Carbon (TOC) concentrations under 50 ppb and post UV-IC anion concentrations under 5 ppb are discussed. Advanced systems show promise for aiding nuclear facilities in meeting solid radwaste, fuel integrity, ALARA, and offsite dose goals.

INTRODUCTION

Changes in social and political affairs have placed increased emphasis on concerns for the environment resulting in new laws, rules, and regulations. Improvements made in analytical instrumentation have had a great impact on analysis limits, with smaller and smaller concentration of impurities detected and evaluated in water returned to the reactor. And, competition from independent power producers dictate cost cutting and expense minimization in every aspect of radwaste operations - from operator overtime to radwaste disposal. These factors drive Nine Mile Point (NMP) to continually research and utilize advanced systems for the treatment of liquid and solid radioactive waste.

Plant Processing History

NMP - Unit 1 is a 610 MW Boiling Water Reactor that operates under a "zero-liquid" discharge philosophy. No water has been released from this facility in over four years due to good water management practices and an effective water treatment program. A forced circulation evaporator was utilized for processing the floor drains from start-up until 1992 when a portable, granular based filtration and demineralization system was installed. The change from evaporation to demineralization provided the following improvements: 1) lowered radwaste volumes 2) increased product water purity 3) lowered operator/maintenance personnel exposure and 4) lowered overall operating cost. In fact, due to the shut down of an

electric fired steam boiler that heated the evaporator, NMP1 set all time electrical generation records. Instead of using the electricity for heating water, it was put on the grid and sold to customers.

In 1994 a new, advanced technology was conceived, cooperatively developed by Nine Mile Point, and tested offsite. After offsite testing provided substantial promise for increased performance and benefits THERMEX was installed at NMP1. This advanced system reduced waste generation rates by over 500% when compared to the demineralizer system. The system has also maintained high product water purity levels despite encounters with high conductivity and high TOC feed water conditions. A second THERMEX is currently being installed at NMP-Unit 2 with start-up scheduled for March of this year.

Goals of the Advanced Processing System

The first goal of the new system is to reduce liquid treatment system solid radwaste generation rates by over 500%. With the uncertainty of disposal site access, NMP1 wanted to extend the life of the on-site solid radwaste storage facility. Replacing the floor drain and equipment drain systems with a more advanced system would increase the life of the radwaste storage facility by a factor of 5 or 25 - 30 years.

The second goal is to process waste water feeds with widely varying concentrations of impurities while maintaining ultrapure product quality standards. NMP's zero liquid discharge philosophy dictates that all water generated within the plant has to be returned for recycle. Lake water in leakage and chemical intrusions into waste water collection tanks periodically caused significant processing problems for the conventional treatment systems. In Table I, "Product Water Quality Requirements", the product water quality standard for the advanced system are compared to those imposed on the demineralizer when it went into service in 1992.

Table I

An ancillary goal of the system was to reduce the concentration of organo-anions that were recycled to the reactor. Standard ion chromatography (without ultraviolet oxidation as a pre-treatment), showed that anions exiting the floor drain demineralizer system were < 5 ppb. After ultraviolet oxidation, the anion concentrations in samples of product water would increase into the tens of ppb. Reducing the amount of anions returning to the reactor will increase fuel integrity and minimize corrosion, thereby reducing overall utility/plant life cycle costs.

The final goal of the system is to reduce onsite processing costs. Niagara Mohawk's cost reduction program dictates budgetary justification all change requests.

PILOT TESTING

Since the advanced process was a new development, pilot scale testing was performed to ensure that equipment installed onsite at NMP was appropriately sized and specified to treat the plant specific waste stream. To minimize costs and reduce the complications associated with testing at a nuclear power plant, two 1,200 gallon samples of floor drain water were shipped from both Nine Mile Point Units to Chem-Nuclear's licensed pilot test facility in Barnwell, SC. The objectives of the pilot test program were to:

1. Obtain operating and performance data necessary to verify full scale system performance. (This will verify the claims made earlier related to improved product water quality and reduced radwaste volumes.)
2. Develop radiological and chemical characterization of typical waste water at various points in the system through monitoring and analysis.

3. Aid plant personnel in developing an understanding of how the equipment operates.

The pilot system is 1/25th scale (2 gpm) and has all the components provided with the full-scale equipment.

Pilot Sample Characteristics

Multiple samples of waste water from other nuclear facilities had been processed through the pilot system prior to the NMP1 sample. To minimize the amount of equipment changes, NMP1 was considering the use of deep bed filters that were already onsite and in use with the portable floor drain demineralizer system. Accordingly, to simulate full scale operation, the test sample was taken from the effluent of the deep bed filter. The concentration of contaminants in the NMP1 waste water was 15-30 umho/cm in conductivity. The sample from NMP1 was cleaner than that from other facilities for several reasons: 1) A few of the other plants obtained their samples by flowing through systems that are not typically used for water treatment. In doing so, a large amount of suspended solid material was added to the sample; 2) The NMP1 sample was filtered through a deep bed filter to aid in simulation of the full scale system. This significantly reduced suspended solids concentrations; 3) The sample from NMP1 was taken from a waste collector tank that contained ion exchange resin. Turbulence associated with recirculating this tank prior to sampling caused the waste water to be mixed with the ion exchange resin, thus lowering the tanks conductivity.

Pilot Processing Results

Physical and chemical results from the operational testing were reviewed by Chem-Nuclear and NMP1 personnel immediately after the tests were completed. The test results were then used to: 1) Custom tailor the system design to specific plant needs; 2) Predict full scale advance processing efficiency; and 3) Final radwaste volumes. Results from the pilot testing are presented in Table II, "Pilot Scale Test Results". The higher than expected concentration of organics can be attributed to an insufficient rinse down of the pilot scale system prior to pilot testing. The pilot system is chemically cleaned and preserved between sample to maintain membranes within their baseline or "clean" performance specifications. Additional rinsing to remove residual chemicals prior to startup would have likely reduced the product TOC. The volume reduction factor was extremely high due to the low conductivity and small amount of solids present in the waste sample.

Table II

Positive results from the pilot scale sample provided justification for implementing the full scale onsite advanced treatment system.

ONSITE OPERATIONS

After completing the 10CFR50.59 safety evaluation and operator training, the full scale advanced system was installed and operations began in January 1995 prior to refueling outage number XIII. Feed and product values for conductivity, pH, and TOC are monitored on a daily basis. The following sub-sections provide additional information on each of the monitored parameters.

Feed Conductivity

Conductivity has averaged 76 mS/cm since start-up. Figure 1, "Feed Conductivity", shows that conductivity averaged 25 mS/cm prior to the outage. The outage lasted from mid-February through mid-April. Outage related maintenance activities caused lake water and decontamination solutions to be put into the floor drains raising the conductivity to

over 100 mS/cm. Now that the outage is over conductivity values are beginning to trend down. One disadvantage of having a resilient water treatment system is that personnel tend to put miscellaneous materials down the drains that would have normally been segregated out and treated through a separate process.

Fig. 1

Feed pH

Feed pH has been neutral to slightly alkaline. During the refueling outage the pH range from 4 to 10. These spikes were likely caused by rinse solutions entering the floor drain after an advanced system chemical cleaning. It should be noted that the flush solutions were very dilute - the conductivity of the flush solutions was lower than the average conductivity experienced during the same time period. No operational difficulties were caused by the pH ranges experienced to date.

Feed TOC

The feed TOC concentration averaged 18 ppm primarily due to unusually high values encountered at the end of March. The high values were possibly caused by a water soluble cationic polymer that overflowed from a clarification tank. Figure 2, "Feed TOC", does not account for oils that were fed to the advanced system several times during the outage. Even though oils entered the system and high concentrations of soluble TOC's were encountered, product quality was not compromised. In fact, the advanced system was the only treatment system at NMP1 that was capable of making water of an acceptable quality for recycle to Condensate Storage Tanks (CST's). During the TOC transient also known as the "swarf event" both floor and equipment drains were processed through the advanced system. If the advanced system had not been onsite water would have likely been released because the plants conventional filter/ demineralizer could not remove the organics associated with the swarf. The oils and TOC did affect the operational performance of the advanced system, it had to be shut down and chemically cleaned after the transient waste water was processed in order to remove high membrane differential pressure.

Fig. 2

Product Conductivity and pH

Conductivity trends are provided in Fig. 3. Product conductivity averaged 0.063 mS/cm or almost 16 MOhm and product pH averaged 6.29. Several of the values reported were determined through the use of a dip cell. These values tended to have higher conductivities and lower pHs due to the reaction of the product water with carbon dioxide in the air to produce carbonic acid. The measurement of pH for high purity water can easily be changed by as much as 1 or 2 units by only trace amounts of CO_2 and other impurities.

Fig. 3

Product TOC

Many organics are non-ionic and highly soluble in water - making them extremely difficult to remove. The TOC concentrations in the product water from the NMP1 advanced system have remained amazingly stable considering that the system has been challenged with feed TOC concentrations of over 600 ppm. To date, the product water has averaged 63 ppb TOC. Since the end of the outage TOC values have ranged between 20 and 50 ppb. Figure 4, "TOC - Feed Vs. Product", shows that the product TOC values are not necessarily dependent on the feed TOC concentration.

TOC removal efficiencies depend on the molecular make-up of the contaminants. Large ionic organics are easily removed and non-ionic low molecular weight organics are difficult to remove. The processing of miscellaneous soluble non-ionic organics, such as, ethylene glycol and hydraulic fluid resulted in several TOC concentrations over 100 ppb.
Fig. 4
Radionuclide Decontamination Factors
Advanced systems offer those facilities that discharge waste water significant promise for the reduction of curies released to the environment. The concentration of gamma emitting radioisotopes have always been at less than minimum detectable activity limits in the product water from the advanced system. Table III, "Decontamination Factors", provides typical concentrations of feed and effluent radionuclides encountered at NMP1.
Table III
Radwaste Generation
The advanced system at NMP1 utilizes multiple process technologies, consequently several types of waste are produced. One waste stream contains ion exchange resin, another is high in dissolved solids, another is high in suspended solids, etc. All of the waste generated at NMP1 has either been shipped to Chem-Nuclear's THERMEX Central Volume Reduction Facility for dehydration or deposited on partially exhausted condensate polishing resin. Since the partially exhausted condensate polishing resin was already considered radwaste, no net radwaste increase was incurred by depositing additional materials on the media.
To date, less than 4 cu-ft of waste has been generated from the advanced system at NMP1. This calculates out to a waste generation rate of less than 2 cu-ft per million gallon of product water. None of the ion exchanges resin contained within the system have exhausted to date; therefore, this rate of radwaste generation is unrealistically low. Based on the waste water characteristics at NMP1, including outage generated water, the projected rate of radwaste generation is 7.8 cu-ft per million gallons of product.
SUMMARY AND CONCLUSIONS
The following conclusions can be developed based on the advanced system test results and the NMP1 operational experience:

Advanced treatment systems that utilize membrane based process components can successfully process waste water from a nuclear power plant.

The industry trends show that the use of an advanced system for the treatment of liquid waste has provided improved water quality and a reduction in radwaste generation when compared to evaporation and demineralization.

Oils and certain polymers are detrimental to the performance of membrane based systems. Prolonged operations with these types of materials will likely result in frequent chemical cleaning.

The rejected contaminants or brine produce a Class A / Type A waste after dehydration.

Conductivity and TOC ranges experienced had little impact on product water quality and overall system performance.

ACTIVE OPERATION OF A MODULOX PILOT PLANT
FOR VOLUME REDUCTION OF ORGANIC

CONTAINING WASTES

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ABSTRACT

A wet oxidation system, ModulOx, for the removal of organic material from radioactive waste has been developed in the UK as a mobile plant concept by AEA Technology in conjunction with Nuclear Electric as part of a four year development project supported by the European Commission and the UK Government (1). The system can effect a significant reduction in waste mass and volume by converting organic carbon to carbon dioxide and water and concentrating the mineral residue prior to subsequent conditioning processes such as drying or cementation. The process operates at atmospheric pressure and aqueous boiling temperature, with organic destruction proceeding by metal ion catalyzed, free radical mediated oxidation with aqueous hydrogen peroxide.

This paper describes the process development and mobile pilot plant design, and presents the results of radioactive operation. Wastes treated by the ModulOx plant include bead ion exchange resins of the type used routinely in PWR/BWR operation, decontamination resins and effluents containing organic chelates such as EDTA, and mixed radioactive sludge.

INTRODUCTION

A variety of radioactive waste streams which contain organic materials are in storage and many continue to be generated. These arise as a result of power reactor operation, fuel reprocessing and decontamination operations, together with contributions from industrial, educational and medical uses of radioactive materials.

Substantial or complete removal of the organic component of radioactive waste can provide one or more of the following benefits :

A significant reduction in the volume of radioactive waste which requires storage, transport, conditioning and final disposal.

Elimination of organic components which may specifically impair safe disposal (toxic chemicals or chelating agents, for example).

Enhanced compatibility of the treated waste with secondary conditioning processes (such as cement encapsulation), producing a conditioned wasteform with better physical or chemical properties for interim storage and final disposal.

Thermal processes, in particular incineration, are the most widely used methods of organics removal from radioactive waste, especially Low Level Waste (LLW). Some organic waste types, notably ion exchange resins, are associated with the production of corrosive by-products which can severely affect incinerator and thermal plant integrity. Complex treatment of the large volumes of radioactive off-gas produced during incineration is usually required, and secondary waste volumes may be large. The volatility of caesium isotopes in particular is a recurrent problem for high temperature processes.

Recognizing that alternative technologies may overcome some of the technical limitations of incineration for several organic waste categories, a non-thermal, atmospheric pressure wet oxidation process using catalyzed hydrogen peroxide has been developed by AEA Technology and Nuclear Electric (2, 3,4).

The ModulOx system does not offer the universal solution to organic radioactive waste treatment optimistically claimed for some thermal technologies. Most plastics, rubbers and oils, together with concentrated organic solvents are not suitable for wet oxidation treatment by this method. Nevertheless, it can be applied to important radioactive wastes such as ion exchange resins, sludges and contaminated liquid effluent which form a significant proportion of organic waste, by volume and activity content, in many countries.

The use of very mild process conditions has allowed the design and construction of a pilot plant which is both modular and mobile, since support systems, power requirements and off-gas treatment requirements are much reduced in comparison to typical thermal technologies. The latter benefit stems from the retention of most of the radioactivity in dissolved ionic or solid forms within the aqueous phase.

This modular wet oxidation system, ModulOx, has been used to process spent radioactive ion exchange resin, solid decontamination waste containing EDTA and citrate, and mixed reactor sludge waste. Preceding the work described in this paper, laboratory and/or non-radioactive plant trials on decontamination solutions and foams, cellulosic waste and organic scintillant solutions have also been conducted (1), as have extensive laboratory and pilot plant trials on toxic effluents (5).

NON-THERMAL, NON-PRESSURIZED PROCESSING OF ORGANIC RADIOACTIVE WASTE

Hydrogen peroxide in aqueous solution can act as a strong oxidizing agent, reacting with nucleophilic organic molecules under appropriate, often mild, conditions. This oxidizing ability forms the basis for its historical use as an industrial bleaching agent.

When trying to oxidize polycarboxylic acids without nucleophilic sites, Fenton found that the reaction with hydrogen peroxide was strongly promoted by the ferrous ion (6). It is generally accepted that in the Fenton system this enhanced oxidizing power is due to the formation of the hydroxyl radical, OH.:

Eq. 1

At ambient temperature and under acidic conditions, the hydroxyl radical has a reported electrochemical oxidizing power higher than any other species except fluorine (7).

With originally insoluble organic polymers such as ion exchange resins, the electronic rearrangements which accompany the initial stages of hydroxyl radical attack have been shown to cause fragmentation by cleavage of benzyl ring systems and alkyl cross-linkages (8). The now hydroxylated resin fragments are more water soluble and their oxidation proceeds in solution. GC-MS analysis of the reaction intermediates present in solution during ion exchange resin oxidation (mixed polystyrene DVB resins) with catalyzed hydrogen peroxide has identified a wide range of oxidized species, including aliphatic acids, ethanol, other short chain alcohols, acetone and primary, secondary and tertiary amines (from anion resin degradation). The large number of different oxidized intermediates found would suggest that the attack on the resin structure is largely indiscriminate.

Ultimately almost all the original organic carbon structure is converted to carbon dioxide, water (from hydrogen abstraction by OH.) and inorganic salts from any functional groups. The overall reaction is therefore analogous to incineration and can be represented approximately by the stoichiometry given below.

Eq. 2

where (X) is a functional group

The rate of reaction of hydroxyl radicals with almost all water soluble organics is extremely fast (7), with reported second order rate constants for hydrogen abstraction in the range 107 - 1010 M-1s-1. In comparison the overall second order rate constant for catalyzed hydroxyl radical production from hydrogen peroxide has been reported as 41.4 M-1s-1 at 20oC in the same study, and although the rate rises with temperature free radical production is under most conditions the rate determining step. It is this latter step, the catalytic cleavage of the peroxide bond, which is controlled in the plant scale wet oxidation application by control of hydrogen peroxide addition rate, temperature, catalyst concentration and pH. The reaction rate must be balanced with the rate at which the exothermic heat and gaseous products can be effectively removed by engineered systems. To maintain relatively stable temperatures and to allow the removal of excess water as steam, the process is operated at or near the boiling point of the aqueous system.

DESIGN OF THE ModulOx PILOT PLANT

The design specification for the pilot ModulOx unit required a plant capable of safely treating radioactive waste, in particular ion exchange resin, containing up to 370 GBq m-3 as 60Co. Another important requirement was the ability of the unit to be readily transported from site to site without dismantling. The design throughput for the plant was 50 - 100 litres of ion exchange resin per 8 hour day.

Following an options study and process selection, the ModulOx unit has been built within its own transport container, and uses a semi-continuous treatment process in preference to batch or continuous operation. This option provides advantages in throughput and process efficiency whilst maintaining the engineering simplicity desirable in mobile radioactive facilities. The pilot ModulOx plant is shown in Figs. 1 and 2. The wet oxidation reactor has a volume of 160 litres.

Fig. 1

Fig. 2

The ModulOx unit design allows ion exchange resins, sludges and liquid wastes to be metered into the wet oxidation reactor, provides for concentration of the mineral residue product using evaporation, and additionally includes a gas treatment system consisting of demisting, wet scrubbing and HEPA filtration components.

The steel ISO container enclosing the plant is 2.6m high, 2.4m wide and 6m long. The plant requires only a 415V 3-phase supply for operation, together with a modest volume of service water for chemical make-up and washdown.A remote control and data acquisition unit, which may be sited up to 100 m away from the ModulOx container, is used for the control of all plant systems during processing of radioactive waste.

NON-RADIOACTIVE TESTING

Ion Exchange Resin

This spent resin had been used to deionize water from steam generator blowdown at a nuclear power station, but was not radioactive. Its chemical and physical characteristics (Table I) were broadly similar to typical nuclear grade ion exchange resin.

Table I

A total of 0.52 m3 of settled resin was processed, at a rate of 60 litres per working day, and reaction profile data obtained (9). Following treatment in the ModulOxm plant, the primary product was an alkaline

mineral sludge containing 90-95% calcium sulphate in the solid phase. A summary of processing results is given in Table II.

Table II

Decontamination Liquor Simulant

A solution consisting of 7% EDTA dissolved in a dilute ammonia solution with added iron and copper salts was processed by the pilot plant. Key results are summarized in Table III. The average throughput achieved whilst removing >99% of organic carbon from the solution was 40 l hr-1. (300 l per day).

Table III

RADIOACTIVE OPERATION OF THE ModulOx PILOT PLANT

Examples of radioactive wastes treated during the first phase of active operations are described below. The ModulOx pilot plant is currently being recommissioned in readiness for a demonstration program involving transport and operation in several European countries.

Spent Ion Exchange Resin

The composition of this organic waste type is given in Table IV below. A waste with relatively low levels of radioactive contamination was chosen to provide information on real waste treatment and distribution of active species without the need for remote waste handling.

Table IV

250 l of settled IX resin was processed at a rate of 40 l day-1 . The residue after wet oxidation consisted of a 35% slurry of calcium sulphate and minor amounts of other salts such as ammonium sulphate, sodium sulphate and calcium hydroxide (the last is added for control of pH). The volume of sludge discharged from the ModulOx plant was 116 litres. 96% of the organic carbon originally present in the ion exchange resin had been removed. The small residual amount of organic matter is almost entirely present as an inert solid. Preliminary analysis suggests that it comprises fragments of unfunctionalized polystyrene present in the original resin polymer.

Spent Decontamination Resin

A similar campaign with real waste comprising ion exchange resin used for decontamination of chemical cleaning solution was carried out. The high levels of the chelating agents EDTA and citrate prevent disposal of this type of waste under current UK Waste Acceptance Criteria for LLW.

The original waste composition is given in Table V. In addition to 60Co contamination, 55Fe and 3H isotopes were also present. Treatment of 360 litres of this waste was carried out at an average processing rate of 50 l day-1 . The organic carbon content was reduced by 95% and specifically the levels of EDTA and citrate were reduced from 4200 mg l-1 and 850 mg l-1 respectively to below the analytical detection limit of 50 mg l-1.

Table V

Following solidification in cement, the waste packages produced met UK waste acceptance criteria for LLW disposal at the engineered near-surface facility at Drigg.

SGHWR Reactor Sludge

The waste treated was a mixture of real and simulant sludge material to reduce the radionuclide inventory to contact handleable levels. The three principle isotopes present in the sludge were 60Co, 137Cs and 55Fe. The organic content of the sludge is associated with powdered ion exchange resin (POWDEX). ModulOx treatment removed 94% of the organic carbon content in this waste, at a net throughput of 60 litres per day. Importantly, the ratio of the three main contaminants present in the

primary waste was preserved, at very much lower activity, in the secondary scrubber liquor and distillate produced. This confirmed that caesium isotopes are not volatile under the ModulOx reaction conditions, and that the likely mechanism for activity carryover of these species into the secondary waste is via particulates or aerosol only.

RESIDUE CONDITIONING AND DISPOSAL VOLUMES

Reference Conditioning Processes for Ion Exchange Resins

In the UK, the current favored option for the treatment of LLW and intermediate level waste (12 GBq t-1 bg and/or 4 GBq t-1 a) organic ion exchange resins is direct encapsulation in cement. This produces a solid product and enhanced stabilization of the radionuclide content with regard to leaching. The waste loading of ion exchange resin in cement is limited by the requirements for the mechanical strength and other physical properties of the resulting conditioned wasteform. For cementation plant currently operating in the UK, the waste loading of both LLW and ILW organic IX resin in cement is typically 50% by volume. Each cubic metre of ion exchange resin waste therefore produces 2.0 m3 of solidified waste for storage and disposal .

Cement Conditioning of Primary Wet Oxidation Residue

The effect of variation in the waste loading of the inorganic residue in cement was examined, using samples of primary residue from the inactive plant trials on mixed ion exchange resin.

The wet oxidation treatment itself produces a volume reduction, with 1m3 of mixed IX resin producing approximately 0.46m3 of residue at 35% solids.

Using a mixture of Blast Furnace Slag (BFS) and Ordinary Portland Cement (OPC) in the ratio 9:1 by mass, the maximum waste loading of residue which produced a product with acceptable mixing and physical properties was established. Mixing trials were carried out at 20 litre scale using 13.3 litres (17.3 kg) of ModulOx residue with a total solids content of 35% by mass.

After 2 days, the cemented residue possessed a compressive strength of 2.6 N mm-2, rising to 19.1 N mm-2 after 90 days. The initial set time was 9 hours. Only small dimensional changes were measured in prism samples of the conditioned residue, suggesting that expansive calcium sulphoaluminate phases had not been formed.

Since each cubic metre of mixed ion exchange resin is reduced to 0.46m3 of residue, at this optimized formulation only 0.69m3 of solidified waste would be produced.

In comparison to the reference method of direct cement encapsulation of the untreated IX resin, treatment by wet oxidation prior to encapsulation of the treated residue should reduce the volume of conditioned waste for storage and disposal by around 66%.

Other Conditioning Methods

In other countries, for example Germany, drying of radioactive wastes prior to storage in high integrity containers is an established conditioning option. To ascertain the potential of this approach for wet oxidation residue, small samples of inactive residue, as used in the cementation trials described above, were evaporated to dryness at 105C. This produced a fine mobile powder. The powder was then loaded into an stainless steel die and compressed with a pressure of 19.6 N mm-2 (commercial waste supercompactors exert pressures in the region of 50 - 100 N mm-2). The compressed product consisted of a solid with a density of 1800 kg m-3. From the results of these small scale drying trials, in

which 100 cm3 of residue was converted to 24cm3 of solid dry product, it may be predicted that a cubic metre of ion exchange resin, treated by ModulOxm and then dried and compacted would only produce 0.11m3 of solid waste. If this conditioning method was acceptable as an alternative to direct cementation, the volume of conditioned mixed IX resin waste could be reduced by 95%.

SECONDARY WASTES

Aqueous Distillate

Water is removed from the wet oxidation reactor during processing by evaporation. The treatment of 1m3 of organic ion exchange resin waste typically produces 5 m3 of aqueous distillate, although the aim is to reduce this quantity by a combination of recycling (e.g. using distillate as a waste transport fluid rather than clean water) and improved waste transfer. The organic content of the distillate, associated with low molecular weight species such as ethanol and acetic acid, varied little between different treatment runs, with organic carbon measurements having an average value of 650 mg l-1 .

By measurement of the 60Co and non-volatile b levels present in the distillate following each radioactive treatment using the mobile pilot plant, decontamination factors (DF's) have been calculated, relative to the original radioactive inventory of 60Co and non volatile b in each run. Variation in DF from 5 x 101 up to >5 x 103 were measured, with activity carryover increased for treatment runs where reagent frothing was more prevalent. The first radioactive run using mixed IX resin, with a 'clean' plant, produced a distillate with the highest DF of >5 x 103. Improvements in process control and plant design may be required if direct discharge of distillate produced during ILW treatment is to be always viable. If DF's cannot be increased sufficiently by these methods alone, then decontamination of the distillate using a small IX resin bed may be utilized. This small amount of secondary spent IX resin can be recycled into the oxidation process, and should only increase the total volume of primary waste by 1% or less.

Airborne Discharges

The non-condensable gas mixture produced by the wet oxidation process consists predominantly of carbon dioxide and oxygen, but may potentially contain small amounts of particulate or volatile radioactivity. For each cubic metre of radioactive IX resin waste processed, approximately 550 m3 of gas is generated. After passing through the ModulOx plant packed tower scrubbing column, the gas stream was mixed with ventilation air from the secondary containment outlet and the combined off-gas filtered using a circular High Efficiency Particulate Air (HEPA) filter.

In all the radioactive treatment runs carried out using the mobile plant, the measured decontamination factor for b particulate activity in the discharged gas was greater than 106 and contained no detectable activity of this type above background levels.

In the case of the decontamination IX resin waste, which contained 7MBq of tritium in the free water associated with the waste, approximately 1.5 kBq of tritium was discharged in the off-gas as tritiated water vapor, corresponding to 0.02% of the original tritium inventory in the waste, or a tritium partitioning equivalent to a DF of 2 x 104.

Scrubber Liquor

The liquid recirculated through the packed tower scrubbing column of the mobile plant was initially 10% by mass sulphuric acid solution. For organic wastes which include nitrogenous material, including most anionic

IX resins, ammonia and amines may be discharged from the treated primary residue during neutralization with calcium hydroxide. Ammonium sulphate dissolved in the scrubber liquor is therefore the main product. If necessary the modest amount of spent scrubber liquor generated during operation can be decontaminated in a similar way to the distillate phase described previously, as similar low levels of active contamination are likely to be present in both streams.

CONCLUSIONS

For the treatment of the important group of organic radioactive wastes which include spent ion exchange resin, decontamination effluent, organic sludges and dilute aqueous liquids contaminated with toxic organics, this demonstration program confirms that ModulOx provides an effective option with several important advantages:

Significant reduction in the volume of spent ion exchange resin without the release of SOx , NOx and volatile metals into the off-gas system, at process conditions which cannot form recombination products such as dioxins and furans.

In comparison to direct cement encapsulation of ion exchange resin, pre-treatment by ModulOx can reduce wasteform disposal volumes by around 66%, or by over 90% if the potential of waste drying and compaction as a conditioning method can be realized.

Chelating agents or toxic organics can be removed by ModulOx from solid or liquid wastes to meet specific Waste Acceptance Criteria for existing or planned disposal facilities, or to meet effluent discharge criteria.

Cement encapsulation of primary calcium sulphate residue from ion exchange resin processing has been shown to produce a solid wasteform with good physical properties suitable for interim storage or disposal.

Operation at 100oC, atmospheric pressure and without the use of any highly corrosive chemicals allows the use of an "off-the-shelf" glass lined mild steel reactor, and standard stainless steel process engineering equipment.

The relative simplicity and standardization of the ModulOx pilot system will minimize the design and development requirements for larger mobile, modular units with throughputs for ion exchange resin of up to 1m3 per day.

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ACKNOWLEDGEMENTS
The work described in this paper was financially supported by the European Commission, the UK Department of Trade and Industry, AEA Technology and Nuclear Electric plc.

36-5
CONTAMINATED CHELATE REAGENT DECOMPOSITION TECHNIQUE

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ABSTRACT
Disposal and safe storage of contaminated nuclear waste is a problem of international scope and concern. Probably the greatest concentrations of waste requiring attention resides with the current nuclear superpower, the USA, and former nuclear superpower, the Soviet Union, from their nuclear power and weapons activity. However, there are also concentrations of similar waste in Western Europe and Japan where disposal and storage could be more acute than in the large land masses of the USA and former Soviet Union because of population density, particularly Japan.
In Japan reducing the radioactive contaminated waste volume needing safe storage is being encouraged. Decontaminating LOW LEVEL radioactive waste reduces or eliminates its demand for safe storage opening up space for the more critical wastes, TRANSURANIC and HIGH level radioactive waste. Chemical decontamination of low level radioactive waste is economical and can be safely achieved without creating any mixed waste because the chemical cleaners can be recaptured, refined for recycling or decomposed into nontoxic elements. Since the decontaminated objects may be recycled as scrap or reusable tools the waste stream is considerably reduced.
As a preface to its main topic this paper will touch briefly on a successful method of chemical decontamination being developed in Japan utilizing methylene chloride and CHELATE SOLUTION for decontaminating low level radioactive waste consisting of tools, scaffolding components and other steel and iron items that are used or removed in routine maintenance programs at nuclear power plants.
In particular this paper will examine the decomposition of CHELATE SOLUTION and the reduction of its waste stream to a negligible amount as developed by Genden Engineering & Construction Company in cooperation with Morikawa Industries Corporation, both of Japan.

INTRODUCTION and BACKGROUND

Japan's problem with nuclear generated radioactive waste disposal, reduction and storage is similar to that of the USA. Japan's waste is mostly created by nuclear power plants, giving it less waste volume to manage since they have no military or defense weapon's volume. However, their problem may well compare to that in the USA when you consider that their population, roughly half that of the USA's, inhabit an area the size of California. It may not be too far fetched to think that if everyone's back yard was in California, the USA could very well have an unsolvable nuclear waste storage problem.

Viewing this as an opportunity to contribute to solving a national problem with a parallel opportunity for commercial reward, Morikawa Industries Corporation (Morikawa) of Koshoku City, Nagano Prefecture, Japan, began research and development activity decontaminating low level radioactive metal objects seven (7) years ago. Morikawa has committed considerable of its resources to this research and development over the past several years, and as designers and manufacturers of special machinery for many years they are able to bring special electro/mechanical/pneumatic engineering skills and knowledge to solving these unique problems.

Recently Morikawa entered into a technology agreement with Genden Engineering & Construction Services Company (GESC), an affiliate of Japan Atomic Power Company, which should enhance both companies abilities and efforts in these developments to substantially reduce the quantity of low level waste (LLW) being placed in extended time storage thus contributing to an environment more safe for the Earth's inhabitants.

As a preface to our principal topic this paper will present equipment with mechanical and chemical processes, developed over the past several years by Morikawa and more recently in cooperation with GESC, for decontaminating LLW metal objects.

Following the preface, this paper presents our principal topic, the development, made jointly by GESC and Morikawa, for successfully and effectively removing low level radioactive contamination from various contaminated components that are generated in nuclear power plants by the use of chelate solutions, which after use in decontaminating objects are decomposed by electrolysis into clean water and harmless carbon dioxide gas thus considerably reducing secondary radioactive waste.

BACKGROUND FOR ATTENTION PAID TO, and EMPHASIS ON DECONTAMINATION

With continuous operation of nuclear power plants in Japan over an extended period of time, a considerable amount of low level radioactive waste has been generated, stored in drums or solidified in concrete, all which is kept at power plant sites. Furthermore, components, pipe and fittings, structural parts, tools, etc. replaced, used and/or discarded during annual plant maintenance programs have been continuously increasing in quantity. Dealing with this problem has become "a real headache" at all nuclear power plants.

Two techniques are commonly used in decontamination. 1) Mechanical processes that physically remove contamination from objects and 2) chemical processes that remove contamination from objects by the use of chemical solvents and/or cleaning agents. Although each technique has favorable and unfavorable features, neither can be said to be an effective decontamination method as a stand-alone process.

DECONTAMINATION USING CHELATE SOLUTION

Chelate agent is also known as a rust removal agent, as its molecular structure has a carbolic acid radical, which has the ability to remove matter adhered to the surface of metals enhancing decontamination. GESC and Morikawa, in order to establish an optimum decontamination method using chelate agent, concentrated on developing a technique to decompose the residual solution containing chelate agent into nontoxic carbon dioxide, CO2, and water, H2O, thus yielding permanently harmless residuals. The joint effort has succeeded in establishing an effective decontamination method that safely decomposes liquid waste.

DECONTAMINATION PROCESSES

REMOVAL OF PAINT COATING ON METAL SURFACES

Chemical Dissolution Method

Low level radioactive objects are put into a methylene chloride (CH2CL2) washing tub. The solution is circulated around and pressure sprayed onto the contaminated objects. This process removes oils and fats from the metal surface and also exfoliates the paint coating exposing the metal surface itself. Then the contaminated object is exposed to supersonic washing inside the methylene chloride tub, where supersonic vibration is induced, which removes even microscopic size paint coating. The spent contaminated methylene chloride from this process is vaporized and distilled purifying it for reuse, which also concentrates the contaminated paint and oil sludge for packaging and safe storage or responsible disposal.

Mechanical Plastic Media Blasting Method

For contaminated objects with paint coatings that methylene chloride cannot successfully remove the mechanical plastic media blasting method may be used. The object's contaminated metal surface is shot blasted with plastic beads accelerated to high speed by compressed air, which upon impact physically removes paint coating and rust that still adheres to the surface. Since plastic media impacting with the object causes no deformation or harm to its surface, this method is effective for decontaminating turbine blades, shafts, bolts, nuts, etc. Spent plastic bead media may be safely disposed of by incineration.

RUST REMOVAL

Contaminated objects that have been partially decontaminated by the use of methylene chloride washing and/or plastic media blasting are next placed into a tub containing a rust removal solution, chelate agent or inorganic acid. Refer to Fig. 1, FLOW DIAGRAM: USED CHELATE SOLUTION ELECTRO-CHEMICAL PROCESS, at the end of this paper. Supersonic vibration is applied to the objects immersed in the solution in the tub to assist and enhance rust removal. In the process using chelate agent, the radioactively contaminated object is decontaminated by metal-ionizing its surface. During decontamination the chelate solution picks up metals and other solids becoming contaminated and must be fortified with fresh chelate to maintain the solution at an efficient processing level. The contaminated chelate enters the waste stream, which in most decontamination processes previously applied entombed the entire waste stream in concrete for safe storage disposal.

Fig. 1

DECOMPOSING TECHNIQUE FOR RESIDUAL CHELATE SOLUTION

The residual chelate agent waste solution that has been used in the rust removal process is placed into a reaction precipitation tub where sodium hydroxide is added to precipitate metallic ions as hydroxides, which results in separating metallic ions from the chelate solution. Again

please refer to Fig. 1, Flow Diagram: Used Chelate Solution Electro-Chemical Process.
The chelate solution is transferred to an electrolytic cell where it is decomposed into carbon dioxide and water and where sodium hydroxide (NaOH) is added supporting the electrolysis process and improving reaction efficiency. The discussion immediately below, supported by several graphs (Figures), demonstrates some of the factors influencing EDTA chelate decomposition.
Figure 2 illustrates how electrolysis lowers the EDTA concentration from 10,000 PPM to less than 10 PPM over a 2-1/2 hour processing time frame. The Y axis uses a logarithm scale to obtain a near-straight-line curve. This data was derived using the following test setup and specifications:
Current density: 8.3 amps/dm2
Waste liquid volume: 3 liter.
Anode surface area: 6 dm2
Added chemical: 0.1 mol NaOH/liter
Fig. 2
Figure 3 shows the effect electrolysis current level has on the time to achieve lowered EDTA concentrations. At 20 amperes current 4 hours is required to reduce EDTA concentration to approximately 60 PPM, while at 60 amperes current 2 hours is required to reduce EDTA concentration to approximately 4 PPM. Expressed another way, the electrolysis process at 60 amperes current requires half the time (2 VS 4 hours) to achieve a 93% lower concentration (4 PPM VS 60 PPM) than occurs at 20 amperes current. These data were derived using the following test setup and specifications:
Electrolyte volume: 3 liter.
Electrode area: 6 dm2
Electrolyte: NaOH.
Fig. 3
Figure 4, following on page 6, illustrates the reduction of radioactivity that occurs during the various decomposition processes shown in Fig. 1, Flow Diagram.
Spent or used EDTA solution has a radioactive level of 30 Bq/cm3.
Step 1 Metal Ion Separation/Filtration process reduces used EDTA solution radioactivity to 1.8 Bq/cm3.
Step 2 Electrolysis further reduces used EDTA solution radioactivity to 0.5 Bq/cm3.
Step 3 Dedicated Filtration reduces EDTA radioactive level to 0.08 Bq/cm3.
Step 4, Reverse Osmosis Membrane processing, reduces used EDTA solution radioactivity to a level lower than the natural background or surroundings.
Fig. 4
Following dedicated filtration and before reverse osmosis membrane processing, as an option, the spent partially processed chelate solution may be processed through an ultraviolet ray reactor tower equipped with low voltage mercury lamps. Bubbling air, which becomes saturated with Ozone when it passes near the mercury lamps, flows into the residual chelate solution. Exposure to the ozone rich air oxidizes the remaining organic matter dissolving it.
The sodium hydroxide contained in the water solution that has passed through the ultraviolet reactor tower is then recovered in the reverse osmosis membrane and recycled.

COMPARISON OF CHELATE AGENT DECONTAMINATION METHOD WITH OTHER METHODS

Morikawa compared methods using chelate agent with other chemicals.

Using inorganic acid the processes involved include:

- Cleaning contaminated objects using inorganic acid.
- Concentrating the liquid waste.
- Solidifying waste in concrete.

This method requires 20 man-hours of labor and produces about 1.5 drums of waste needing storage space.

Using chelate agent solution the processes involved include:

- Precipitation treatment.
- Electrolysis treatment.
- Secondary waste treatment.

The chelate agent method requires 4 man-hours of labor producing about 0.04 drums of waste, principally filter media, needing storage space.

As these figures for the competing processes show, the chelate agent method uses 4 man-hours while the inorganic acid method uses 20 man-hours of labor. The Chelate agent method saves 80 percent of the labor man-hours. Also the chelate method generates 0.04 drums of waste while the inorganic acid method generates 1.5 drums. Chelate agent generates less than 1/100 of the waste requiring storage that the inorganic acid method generates.

HOT TEST RESULTS, LOW LEVEL RADIOACTIVE OBJECTS.

Morikawa conducted decontamination hot tests of contaminated objects using processes including Organic Solvent Decontamination, Rust Removal Decontamination and Washing. This test was conducted at Tokyo Electric Power Company's (TEPCO's) Fukushima No.1 Nuclear Power Plant.

The steel tools and objects subjected to decontamination processing numbered 159 as shown in Table I at the end of this paper. For the various tool and steel objects decontaminated in the hot test, data in Table I show the Level of Radiation and Count of Objects successfully decontaminated in the first step. During the first step 131 objects were successfully decontaminated to radiation levels below 50 cpm out of 159 objects processed. The test was not 100% effective in decontaminating all objects because the test equipment consisted of laboratory units not suitable for processing some tools and several steel objects, such as the unacceptable 28 objects having levels over 50 cpm readings following first step decontamination. These 28 objects had heavy welds, thick rust and/or deep cracks. After hand grinding these 28 unacceptable objects were processed again in a second step through the decontamination equipment. These data as displayed in Step 2 at the bottom of Table I show that all previously unacceptable 28 objects dropped to acceptable levels below 50 cpm or below background level readings.

Table I

FUTURE PROJECTS

Morikawa's initial prototype equipment for decomposing contaminated liquid waste Chelate solution has proved to be technically satisfactory. Another prototype machine has been assembled using the same test equipment components and is being shown and demonstrated at nuclear power plants now around Japan.

A basic patent application for the technique of chelate agent decontamination and liquid waste decomposing equipment was issued in 1993. Additional patent applications for peripheral technology and related applications are being filed now. These additional patent

applications are also being filed in the USA and other major industrial countries.

SUMMARY

We believe this decontamination method for steel and iron metal substrates using cleaning chemicals and/or chelate solutions for removal of radiation bearing coatings. corrosion or rust, grease, oils and other solid contaminates where the cleaning chemicals can be purified for recycling/reuse by vaporizing and distilling, and/or where other chemical chelate solutions can be decomposed into nontoxic elements or substances for safe release to the environment, to be an economically viable and responsible decontamination method substantially reducing the volume of low level radioactive waste and mixed chemical waste that requires expensive long term storage thus freeing up space for the more critical storage of transuranic and high level radioactive waste. Not only do these processes avoid long term storage for a considerable volume of material, they make it possible to safely recycle steel and iron scrap through normal primary metal industry channels.

Responsible avoidance of storage for radioactive waste, wherever possible, is not only highly desirable economically, socially and politically, but should be aggressively pursued and could well be, if not already established, as a major nuclear industry objective and policy.

36-6

DRYING OF LIQUID OR SUSPENDED RADIOACTIVE WASTE UNDER ATMOSPHERIC CONDITIONS

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ABSTRACT

The German regulations for the final disposal of radwaste prescribe, that these waste neither contain nor release free liquids, apart from reasonably achievable and unavoidable residual content. In addition, the volume reduction of radwaste is to be the preferred treatment prior to all other methods for conditioning.

Referring to these requirements, a new designed system is now available for drying water-containing, pumpable operational radwaste. The separation of the water from the solid matter is performed under atmospheric conditions with a hot-air heating. The solidified product is generated within the disposal container. The process and the system design offers economical, operational and radiation protection related advantages.

INTRODUCTION

In the Federal Republic of Germany, the legal situation stipulates volume reduction of radioactice waste is to be the preferred treatment prior to all other methods for conditioning. On the other hand, the regulations for the final repository require avoidance of fermentation, rotting and corrosion processes, as well as the generation of radiolysis/hydrolysis gas set off by reactive water (1).

An advantageous and acceptable treatment for meeting these requirements is the drying of the material, especially when it contains mostly water as liquid and is non-burnable.

Liquid waste has already been partially dehydrated mechanically or thermically with the existing systems of the nuclear power station (sedimenting, centrifuging, evaporation etc.). Pre-treated liquid wastes with a high water content are for example Evaporator concentrates and Filter sludges (suspensions of ion exchange resins, filtering aids etc.). However, because of the remaining high water content, this pre-treated liquid waste is not suitable for final disposal, so it must be subjected to thermic residual drying. As a result, a significant reduction in volume will take place.

For this purpose, various techniques have been developed in Germany. The drying system that will now be described offers an advantageous, proven dryer unit which conditions liquid waste in accordance with the regulations for the final disposal of radwaste (2,3).

SYSTEM DESCRIPTION OF HPA DRYER WITH POST-FEEDING

The flow chart of the process is shown in Fig. 1:

Fig. 1

The drying of liquid waste is carried out in standard disposable containers (200-l-drums or 180-l-cartridges). The movement of the containers is carried out with a transport carriage which can be transported, depending on the individual controls activated on the control panel, to the lid removal unit or the lidding unit.

The heart of the process is the drying chamber. After the lid has been removed, the container is transported into the drying chamber, positioned appropriately, lifted by a lifting carriage to the permanently installed docking plate and, at the same time, sealed.

The receipt of the liquid waste from the nuclear power station for the dryer takes place in the storage tank. To prevent sedimentation processes effectively, the liquid waste is constantly stirred and circulated. The liquid waste from the circulation system is fed into the container with a regulated level by means of the post-feeder. The heating operation will be started during the feeding of the liquid waste into the container. Therefore, the air contained in the drying chamber is circulated in the electrically heated flow channel by means of a hot air fan and, depending on the type of waste, heated to 130 to 190 C (heating circulation). The heated air is fed to the container in the drying chamber by an optimized flow.

After a short heating phase, the drying process begins under atmospheric conditions. The resultant water vapor is extracted via a separate steam pipe and condensed in the subsequent condenser. The condensate that arises is measured in the condensate measuring vessel and, after interim storage in the condensate collecting tank, it is returned to the waste water line of the nuclear power station.

During the drying process, liquid waste is again added to a controlled level, and the non-watery waste components, including the radioactive nuclides, are concentrated in the container. By this means, the drum volume is fully utilized.

Depending on the moisture content and the composition of the waste, different drying times occur. The process control is determined by the rate of condensation. With an increasing concentration of solids, the amount of condensate decreases over time. As soon as the rate of condensation is below a specific level, the post-feeding is interrupted. Simultaneously, the after-drying process begins. The after-drying process is deemed to be complete when no noticeable amount of condensate is produced.

During drying of evaporator concentrates, a solid salt block is finally formed.
After the drying process has been completed, the cooling operation is begun, and the filled container is removed from the chamber with the transport carriage, the lid is fitted and the container is transported to the unloading position.

EQUIPMENT DESCRIPTION

The liquid waste drying system shown in Fig. 2 is built as a tandem installation. This enables a higher throughput to be achieved.
Fig. 2
The system consists of two drying chambers (each with a post-feed system), one storage tank, one transport section (consisting of carriage for receipt and positioning of containers, lid fitting machine for lid removal and lid fitting, cartridge grapple to remove compressed cartridges from the container, cartridge lid fitting machine for lid removal and lid fitting and weighing system), Control panel, Process control system "BIKON", Video monitoring feature, Shielding system and Sampling system.
The technical data and consumption of operating materials are shown in Table I.
Table I

AUTOMATIZATION

In order to comply with the radiation protection regulations it is necessary to handle the full containers by remote control . The system is therefore shielded and prevents be entery by personnel during normal operation.
In the shielded zone, all functions are fully controlled automatically from a separate control panel. The automatization includes the transport of the container into and out of the appropriate lid fitting and lid removal stations and the packaging unit and the transport of the containers to the loading and unloading of the drying chamber, the filling of the storage tank, the feeding of liquid waste and the supervision of the drying process.
All functions can be addressed individually and can be carried out under video supervision.

DOCUMENTATION

In accordance with the regulations for the control of radioactive waste, the waste conditioning must be appropriately documented. This includes the precise recording of the operating data relevant to the process. This makes it possible to demonstrate that the waste product for permanent storage has been produced correctly.
The processing of each batch is recorded with a proprietary process control system "BIKON". The advantage of this system is, that the important data for the batch documentation can be taken directly from the process monitoring system. At the same time the entire process can be visualized on screen. The batch documentation which is saved on disk can be analyzed at any time with separate PC.
Furthermore, the operating parameters relevant to the process are documented in fail-safe form with a multi-channel printer.

OPERATING RESULTS

Since October 1994, extensive operating experience has been gathered with a drying system with post-feeding feature, configurated as a tandem installation. Several drying experiments have been carried out, which have produced major insights into the optimum control of the process. As

a result, the installation has been optimized with regard to the maximum evaporation rate and optimum product quality.
The typical behavior of the drying process is shown in Fig. 3:
Fig. 3
Because of the isothermic heating of the container with the liquid waste, a maximum evaporation rate is achieved at the beginning which is dependent on the feed temperature of the heating circuit, the material of the container (thermal conductivity) and the filling level of the container (heat transfer surface area).
As the drying process progresses, the evaporation decreases as a result of the corresponding increase in the salt concentration. This enables a gentle evaporation to be achieved. The end of the drying process is indicated by the sudden reduction in the evaporation rate.
The principle of the salt block formation as a result of isothermic heating of the container can be seen in Fig. 4:
Fig. 4
In the interior of the filled container, a flow profile is created by the convective transfer of heat to the liquid waste (natural circulation). This creates a temperature gradient from the container wall to the central area. As a result of the permanent removal of water, a compact salt block (monolith) forms under isothermic heating in the cooler area, and in the drying process this mass grows slowly outwards (evaporation crystallization). Therefore the structure of the monolith is layered in cross section.
Carrying out the drying process under atmospheric conditions leads to less water content during the salt block formation. At the end of the drying process, the container will be heated without post feeding of liquid waste. In this drying operation, the isothermic heating leads to firm products. The overall mass balance of the process shows, that it will produced 15 to 20 % solids, formed as a monolithic salt block in the container (calculated on the basis of 100 % evaporator concentrate). The salt block is characterized by a high specific weight (r = 1.7 kg/l), a high solidity, thermic stability up to 70C and a residual moisture content less than 10 % (crystallization water, no free water). The volume reduction factor of the liquid waste is between 8 and 11 : 1.

SUMMARY

The German regulations for the final disposal of radwaste prescribe, that these waste neither contain nor release free liquids, apart from reasonably achievable and unavoidable residual moisture. In addition, the volume reduction of radwaste to be preferred treatment prior to all other methods for conditioning.
Referring to these requirements, a new designed system is now available for drying water-containing, pumpable operational radwaste. The described drying system with postfeeding of liquid radwaste during the drying process offers the following advantages:
Drying under Atmospheric Conditions
High degree of drying
(lower residual water content of the dried product leads to maximum volume reduction in comparison with vacuum dried products)
Isothermic Drying with Hot Air
Monolith formation
(high end product solidity, no local overheating in the final drying phase)
Minimal Contamination Risk

Drying in containers
(no filling process after drying, separate vapor extraction)
Highly Automated
Fully automatic control from a shielded control panel with video supervision
High Documentation Standard
Immediate process visualization throughout the drying process
Simple quality control of product
High Level of Flexibility
Designed as a small compact unit on a modular principle
Supply of mobile or stationary installations,
on request, service operation possible

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Session 37 -- CHARACTERIZATION OF MIXED WASTE
Co-chairs: Mike J. Ryan, Bechtel Environmental Inc.;
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37-2

CHARACTERIZATION OF HANFORD TANK WASTE AND THE ROLE OF HISTORIC MODELING

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ABSTRACT

The tank waste characterization process is an integral part of the overall effort to identify, quantify and control the hazards associated with radioactive wastes stored in underground tanks at the Hanford Reservation.
Characterization of the current waste tank contents through the use of waste sampling is only partly effective. The historic records must be exploited as much as possible. A model generates an estimate of the current contents of each tank, built up from the estimated volumes of each of the defined waste components. The model combines the best estimate of the waste stream composition for each of the major waste generating processes. All available waste transfer records were compiled and integrated to track waste tank fill history. The behavior of the waste materials in the tanks was modeled, based on general scientific principles augmented with specific measurement data. Sample analysis

results were not used directly to generate any of the tank contents estimates, but were used to determine the values of variable parameters such as the solubility. By considering all available information first (including historical model estimates, surveillance data, and past sample analysis results), future sampling resources and other characterization efforts can best be spent on tanks that will provide the largest returns of information.

INTRODUCTION

The Hanford Reservation system of underground storage tanks contains approximately 230 million liters of waste material distributed among 177 tanks. The waste in these tanks was produced as a part of the nuclear weapons materials processing mission that occupied the Hanford Site for the first 40 years of its existence. Waste tanks (149 single-shell and 28 double-shell tanks) contain a wide variety of waste compositions generated by three distinct chemical separations processes, several waste management/waste volume reduction operations and two tank waste reprocessing flowsheets. Characterization of the tank wastes is required to maintain the safe storage of the wastes, and to guide retrieval, processing, and disposal technology development.

MOTIVATION FOR CHARACTERIZATION

The tank waste characterization process is an integral part of the overall effort to identify, quantify and control the hazards associated with radioactive wastes stored in underground tanks at the Hanford Reservation.

Knowledge of the physical, chemical and radiological properties of the wastes is prerequisite to operations to store, retrieve, process and dispose of the wastes safely.

The Tank Waste Characterization Project currently addresses the information needs identified for the Tank Waste Remediation System (TWRS) through the Data Quality Objectives (DQO) process (1).

Safe Storage of the Waste

Characterization for safe storage of the waste includes the screening of all tanks for key parameters that may indicate potential safety issues. The areas of concern are:

The presence of energetic compounds in the condensed phase of the waste in a configuration that could support a propagating exothermic reaction (i.e. dry, in the presence of an oxidizer, located where an initiating factor is feasible).

The presence of flammable gases in the dome space above the waste surface at sufficient concentrations that combustion is feasible.

The presence of fissile materials concentrated in the absence of significant neutron absorbers so that a criticality event is feasible.

In all areas of concern, several factors must interact to give rise to an actual safety issue. The presence of specific waste components, addressable by characterization, is only one of the contributing factors. Characterization can be used to identify and quantify hazards and develop the most appropriate responses for dealing with them. A combination of active and passive controls can provide multiple safeguards that prevent an event from occurring.

Safe Operation of the Waste Tank Facilities

Although many of the tanks are currently stabilized and undergo no operations, several operations are ongoing in a subset of tanks. A waste evaporator unit concentrates liquid waste to provide additional storage space in the existing tanks. Removal of pumpable liquid from single shell

tanks is performed to prevent leakage into the soil. Transfers of waste between operating double shell tanks is necessary to support these operations. Other specific operations may occur, such as the addition of caustic to a tank to maintain operating specifications.

All operations must be performed in accordance with all applicable regulations. Intrusive operations must be carried out in a manner that ensures that none of the safety issues identified above (see Safe Storage of the Waste above) are generated during operations. This requires the review of existing characterization information prior to initiation of the operation, and may require acquisition of new information.

In addition, other information on waste characteristics may be required to support physical operations. For example, prior to transfer of waste material, the physical parameters of the waste must be well enough understood to ensure that the material will not solidify during transfer, clogging the transfer lines.

Resolution of Safety Issues

The above areas address characterization needed to ensure that safety issues are recognized so that they may be corrected, are not created during waste management and disposal operations. Separately, a thorough understanding of the mechanism behind the safety issues is needed in order to select the appropriate response. For example, one needs to understand how flammable gas is generated in the waste material, and what causes it to be retained or released at a given rate. This understanding ensures that the symptoms of a potential problem are correctly identified, that the correct controls or treatment for the problem are applied, and that sufficient foresight is applied in the future to prevent occurrences during waste management and disposal operations. The information to resolve safety issues is obtained from many sources, including theoretical work, laboratory experimentation, and in some cases characterization data.

Preparation for Waste Disposal

Safe storage and operation of the waste tanks is an interim step until facilities are available for the retrieval, pretreatment, processing and final disposal of the material. The equipment and facilities for these steps are still in the design phase. Characterization of the waste requires that the physical and chemical properties be adequately quantified, both in terms of average values and of bounding values, to support design. In addition, physical samples of waste material are required for small and large scale testing of various treatment processes.

TANK WASTE CHARACTERIZATION PROCESS

Characterization of the current waste tank contents through the use of waste sampling is only partly effective. The waste tanks contain very few access ports, limiting the number of samples obtainable. The access port locations do not necessarily support obtaining representative samples, particularly in tanks where the waste material is heterogeneous. Sampling is expensive and complex because of the radioactive and chemically hazardous nature of the waste. The historic process records must be exploited as much as possible to improve overall waste characterization. By considering all available information first (including historical model estimates, surveillance data, and past sample analysis results), future sampling resources and other characterization efforts can best be spent on tanks that will provide the largest returns of information (2).

The Overall Process

The approach to characterization of each tank is to compile all available information about each tank to provide an estimate of the contents. Information sources include models of waste contents developed from historic records of processes and waste transfers, surveillance and monitoring data, sampling and analysis results, and models of chemical behavior. The estimates developed from these sources are improved by sampling or other measurement to provide additional data. Grouping of similar tanks is employed in developing the estimates (3,4). Based on the amount and quality of data extant, an estimate of tank contents and the identified needs, a plan is developed to obtain additional data (5,6). The development of a defensible plan requires an understanding of the quality of existing data and the capabilities and limitations of the tools which may obtain new data.

Role of Historic Modeling

The waste tanks contain a wide variety of waste compositions, principally generated by three distinct chemical separation processes, several waste volume reduction operations and two tank waste reprocessing flowsheets. All of the processes contributing to the waste generation underwent significant evolution over time. Extensive (albeit incomplete) records were kept describing the initial placement of waste in specific tanks and the later transfers of waste between tanks. The complete reconstruction of current tank contents through use of the records is complicated by several factors:

Although the flowsheets for the waste-generating processes are well known at the start of a process, the evolution of the process during plant operation is not well documented. In addition, process vessel corrosion and impurities in process chemicals can dramatically affect the nature of the waste stream.

Active concentration of tank waste supernatants in six different evaporator campaigns spanning the fifty years of processing have not been thoroughly documented. There are many uncertainties in the tank transaction histories for these campaigns. The waste heating during evaporator campaigns may also have accelerated chemical reactions, changing waste properties from those described in the flowsheets or in early sample analysis.

The waste transactions associated with the removal of tank waste for the two major tank waste reprocessing campaigns, the Uranium Recovery campaign in the 1950's and the Cesium/Strontium removal campaign in the 1960-70's, are incomplete. These processes were based on assumed waste characteristics, not actual waste characteristics. Significant changes were performed as the processes evolved during the course of the reprocessing campaigns. In many cases these changes were not well documented.

The construction of a model of tank contents combines the best estimate of the waste stream composition for each of the major waste generating processes and several minor processes. Altogether, 48 distinct waste types are modeled, referred to as the Hanford Defined Wastes (7). All available waste transfer records were compiled and integrated to track waste tank fill history (8-11). The behavior of the waste materials in the tanks is modeled, based on general scientific principles augmented with specific measurement data (12,13). Although sample analysis results are not used directly to generate any of the tank contents estimates, sample analysis is used to determine the values of variable parameters

such as the solubility and precipitation rates of specific analytes in specific waste materials.

The model generates an estimate of the current contents of each tank, built up from the estimated volumes of each of the defined waste components (14-17). The quality of these waste content estimates is now being reviewed and quantified through several parallel activities. A Monte Carlo simulation has been performed to quantify the uncertainty of the model estimates for several tanks, based on the variability of key model parameters such as limiting solubility. This intial variance or uncertainty estimate has been used in the systematic comparison of tank contents estimates with actual sample data (18). Initial results are promising in most cases (i.e., relatively good agreement between model estimates and sampling estimates). However, some results indicate that there are significant differences between the model estimates and sampling results. This observation could indicate:

- incomplete modeling of the major error sources in the Monte Carlo simulation,
- overly optimistic estimates of variance based on sampling results,
- systematic errors in the model that require correction,
- or some combination of the above factors.

Additional samples of key waste types are being acquired and analyzed to allow better definition of the composition of the defined wastes types used in the model (19). The evaluation and improvement of the historic model will continue until a quantitative variance estimate can be provided for each of the contents estimates. The availability of the resulting tank contents estimates will allow reduced sampling of many tanks, better grouping of similar tanks, and more effective planning of sampling events.

The Role of Sampling and Analysis

Analysis of sample material alone will not provide adequate characterization information for the Hanford underground storage tanks. It is not possible to design a sampling scheme that provides a statistically significant number of truly random samples.

Most tanks have very few available access ports or risers (many have no more than two accessible risers). The locations of the risers often leads to sampling of non-representative material. The risers have in some cases been used for dumping of additional waste material. Previous samples may have been taken from the risers, disrupting the solid materials. Instruments may have been introduced into the waste through a riser. Removal of instruments may have required decontamination, introducing water or other solutions into the waste near the area where samples are taken.

Given these constraints, the value of sample analysis may be called into question. However, when combined with the historic model and contemporary surveillance data, and allowing for potential biases and sampling error, a well thought out sampling scheme can provide significant information. The first step in the development of the sampling scheme is to consider the historic information, including the waste type predictions, records of any previous sample activity and photographs of the waste tank contents. After this initial assessment, contemporary surveillance data is considered. Review of the available risers is necessary so that any factors that may make a specific sample different from others in the tank are understood. Review of the photographs can reveal surface heterogeneity and give clues to the relationship between a sample at a

specific location and the overall tank content. It must be noted that some tanks may be so heterogeneous that very little information can be gained from any small number of samples. In these cases, an alternative approach (such as adding liquid to turn the waste into a homogeneous slurry and retrieval into an interim storage tank) must be considered. When a sampling approach is selected, it is also necessary to consider how the acts of acquiring, removing, transporting and analyzing the sample may affect the parameters being measured (20). Any systematic biases that are introduced during the sampling process need to be considered when using the results to reconstruct the total tank contents. Given all the constraints, the most realistic approach to tank waste characterization must consider all available data, start with a model of the waste generated from the historic records, and then use sampling to confirm or refine specific aspects of the model.

Optimization of the Process

Programs requiring data regarding waste composition document their issues and information needs and identify tanks which must be characterized to meet the needs. The programs identify criteria by which all tanks can be prioritized with respect to each issue. The Characterization Project integrates the information needs and tank priorities to define an overall plan for obtaining new information through sampling. The process for developing the priority list and generating a sampling schedule includes the application of technical and operational constraints. The most effective overall prioritization of tank sampling events ensures that early events provide data that supports characterization of multiple tanks. High priority sampling events are those that provide data to:

Allow resolution of safety issues affecting multiple tanks.

Define the appropriate approach to characterization of multiple tanks for safety issue identification and resolution.

Improve the estimates of waste content for multiple tanks, particularly with regard to important safety and disposal parameters.

It is not yet possible to determine the total number of samples needed to characterize the wastes adequately (either for an individual tank or for the entire tank farm system). An initial set of 28 tanks has been identified as high priority to address the above issues (2). It is anticipated that the information gained from those samples will support resolution of issues (reducing the characterization needs associated with other tanks) and provide quantitative information about the quality of the tank content estimates developed from historic data. Once the uncertainty associated with the historic models is quantified, it may be possible to use the model estimates alone to make future decisions regarding the operation and disposal of specific tanks. This application of the historic model has the potential to greatly reduce future sampling requirements.

CONCLUSIONS

Continued evaluation of historical information and modeling techniques is needed to enhance the value of that information and potentially reduce the need for future in-tank sampling operations. As additional knowledge is gained, that knowledge is fed back into the process to help prioritize, guide and define future efforts so that the most important information is obtained as soon as practicable. Continued evaluation and improvement of sampling and measurement methods is required to improve the ability to obtain new information and to understand its meaning and limitations.

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CHARACTERIZATION OF HANFORD N REACTOR SPENT FUEL AND K BASIN SLUDGES

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ABSTRACT

Over 2,000 tons of irradiated zirconium alloy clad uranium metal fuel are stored in the water-filled Hanford K Basins. Half of this N Reactor fuel is in open top aluminum and stainless steel canisters (in K East Basin) and half is in sealed vented canisters (in K West Basin). On the basin floor and associated with this fuel is an accumulation of sludge containing fuel, fission products, corroded structural material, and wind blown debris. Previous papers discussed plans and equipment for the sampling of gas and liquid from sealed canisters, sampling of sludge from the basin floor, and the movement of fuel to the Hanford hotcells. These sampling activities have now been accomplished. Various lessons have been learned from the execution of various sampling campaigns at the K Basins, and analytical characterization data resulting from the examinations continue to support decisions for the storage and disposal of fuel and sludge.

1. Data obtained from the sampling of gas and liquid in sealed canisters include radionuclide concentrations in the liquid and the degree of displacement of the nitrogen canister cover gas by significant quantities of hydrogen. The macroscopic condition of fuel and the volume of in-canister sludge, observed when selected canisters are opened has been shown. Inferences on the degree of fuel corrosion in canisters can be drawn.

2. The recent sludge sampling campaign for the K East floor has been much more comprehensive than any previous attempts with respect to both

quantity of sludge retrieved and the number of physical/chemical properties addressed. Sludge composition and properties are now known as a function of both basin location and of the position of various layers within the sludge.

3. Fuel elements from the K West Basin have been examined both visually and metallographically at the Hanford hotcells. Detailed observations on the condition (including corrosion) of damaged and undamaged elements have been made.

4. A controlled-atmosphere furnace has been installed in the Hanford hotcells. This new capability has allowed small sections of fuel elements to be subjected to the dewatering and conditioning processes envisioned for the fuel now stored in K Basins. Moisture reduction and hydrogen evolution have been monitored as a function of temperature and system pressure (vacuum and flowing gas scenarios). With this system, oxide layers have been applied to fuel samples to reduce the chemical reactivity of the exposed surfaces and the stability of these protective layers has been assessed.

INTRODUCTION

Characterization is in progress for the N Reactor fuel stored in the Hanford K Basins. These activities (1) support the strategy for removal of fuel from the basins and storage of fuel in a dry condition at an area remote from the Columbia River. This strategy currently consists of placing fuel in a Multi-Canister Overpack (MCO), drying the fuel while it resides in the MCO and conditioning some portion of the fuel to reduce its chemical reactivity. (2) Characterization includes the examination of fuel, canisters, and associated sludge. It consists firstly of in-basin activities such as visual examination, sludge depth measurements, and sampling of gas and liquid in canisters. Secondly characterization encompasses the examination of samples of fuel and sludge which have been removed from the basins and shipped to laboratories. This paper presents observations made in the basins during the most recent attempts to ship samples from the basins and data obtained in the laboratory hotcells.

EXAMINATION OF CANISTERS AND FUEL

Of the 2,000 metric tons of uranium metal spent fuel at the two Hanford K Basins, roughly half is stored in sealed aluminum and stainless steel canisters in the K West Basin. Each water-filled canister contains 14 fuel assemblies (14 inner elements and 14 outer elements distributed between two barrels). In March of 1995 three canisters were opened in K West Basin and three fuel elements were shipped to the Hanford hotcells. In-Basin observations were made during the fuel retrieval operations and the fuel was subsequently utilized in detailed hotcell examinations and conditioning tests summarized in the following sections. Specific canisters to open were chosen through a review of the data base of available fuel inventory and through sampling of canister gas and water to identify presence of uncontained fission products (which implies damaged fuel in a given canister). Only those canisters which were stainless steel, which showed evidence of gas generation and which contained fairly long fuel elements were considered candidates for opening due to emphasis, for this first shipment, on evaluating particular corrosion mechanisms.

Gas and Water Samples From K West Canisters

In order to identify canisters likely to contain failed fuel, liquid samples were obtained from 10 canisters (20 barrels). Also, in a few cases, gas samples were obtained from the sealed barrels. (Note canisters

nominally are water-filled and have a 2.5 inch nitrogen gas space when they are first loaded into K West Basin). Gas and water samples obtained in this manner were first evaluated by a mobile laboratory which provided immediate analysis for fission products i.e., cesium in the water and krypton in the gas. Subsequently gas and water samples were sent to laboratories at Hanford for additional identification of chemical species.

Analysis of cesium in K West canister water did lead experimenters to the desired fuel elements for hotcell examination. The most badly damaged element removed from the canisters was indeed from the canister with the highest cesium content (approximately 0.5 curies). Hydrogen gas was found to be the major constituent of all of the gas samples. Thus the original nitrogen cover gas had been largely replaced. Krypton gas concentrations correlated well with hydrogen which in turn correlated with cesium concentrations in the liquid. Thus krypton is a credible indicator of corrosion but krypton and hydrogen concentrations do saturate as the original nitrogen covergas is replaced.

Reaction of Basin Water with Fuel

After three canisters were opened in the K West Basin, no energetic reaction was observed between the newly introduced basin water and the fuel. After three fuel elements were placed in shipping containers the associated containers were capped with inverted graduated cylinders to trap bubbles and indicate any further reaction (i.e., hydrogen production). At the end of 2 to 5 days of monitoring, no accumulation of gas was found in any of the three cylinders even though at least two of the resident elements were failed with obvious exposed fuel.

Observations of Sludge During Fuel Handling

When canisters lid valves were opened during the canister flooding operation a continuous stream of bubbles was observed. Often accompanying this stream was a distinctly separate stream of reddish-brown liquid which appeared to have the character of a suspension and which was easily distinguishable from the surrounding basin water.

When canister lids were opened to remove fuel, large amounts of sludge were not observed. For two of the canisters no significant sludge was visible in the canisters even when fuel elements were removed. No sludge was seen to accompany the fuel elements retrieved from these two canisters during their transfer to shipping containers. For the fuel element retrieved from a third canister, sludge was seen trailing behind the element during its transfer to a shipping container. Removal of the target element from this latter canister stirred up sufficient flocculent sludge to fill the canister barrel to within a few inches of the top for several hours. Variations in observed sludge content may however be linked to variations in flooding and opening procedures.

Visible Condition of the K West Fuel

The retrieval of fuel went as follows:

1. When the first of three canisters was opened, it was found that a known fuel element breach (identified in the loading video tapes circa 1983) had not greatly deteriorated, if at all, since the time it was placed in the canister. This outer fuel element, with obvious missing fuel/cladding piece, was selected for hotcell examination and moved to a shipping container. Subsequent hotcell examinations did in fact identify some additional corrosion which can be attributed to in-canister storage.

2. The second canister was opened based on its high cesium concentration. One outer element with split cladding (Fig. 1, inset) was

easily visible. The fuel in this element certainly had reacted with the canister water (i.e., reaction was not apparent in video tapes made during original loading) and the element was selected for hotcell examination.

3. The third canister (Fig. 1) was opened based on written records of a breached element and on a moderate cesium content measured in its water. An inner element was selected for shipping to hotcells based on the desire for an intact element and on a visible dent in the end cap of the selected element.

Fig. 1

Metallographic Examination for Hydrides

Emphasis during the metallographic examination of the three K West elements was placed on the identification of uranium hydride due to its association with reported pyrophoric events. Specimens were cut using a tungsten-carbide slitting saw mounted on a remote-operated milling machine. Argon gas was used to cool the fuel element during the cutting, to provide an inert cover and thus, preserve the fuel microstructure as near to the as-received condition as possible. Cut samples were mounted, polished, attack-polished using chromic acid, and examined. The presence of uranium hydride was detected by a heat-tinting technique which delineates the oxidized hydride inclusions from the uranium matrix. Uranium hydride inclusions were found to be randomly distributed throughout the fuel matrix. It is not possible at this time to unambiguously determine the source of the hydrides. Possible sources include: 1) residual hydrogen which has been in the uranium alloy since fabrication, 2) hydrogen present as a result of corrosion processes and subsequent migration within the fuel, and 3) hydrogen present as a result of diffusion through the cladding. Thermodynamically, the zirconium alloy cladding is a favorable sink for hydrogen and therefore hydrogen diffusion through the cladding is the least likely source of hydrogen in the fuel.

Fuel Conditioning

Fuel samples (with cladding intact) from the K West elements were tested in a controlled atmosphere furnace to determine drying characteristics, dehydriding behavior, and oxide film formation for exposed uranium surfaces. These experiments are the first attempt to study a process whereby at least some of the fuel in K Basins will be conditioned in an oxygen containing atmosphere to reduce chemical reactivity through application of an oxide layer. Specimens were taken from a corroded fuel element but from areas which were not immediately adjacent to the principal corrosion sites. They were placed in a furnace system located in a hot cell and tested according to the following heating cycles:

A fuel drying step in which the fuel is dewatered for approximately 10 hours (free water removal) either at 373K in dry argon or at 323K in vacuum. The fuel is then dried at 573K under dry argon (or vacuum) for approximately 24 hours (to remove water of hydration and partially decompose any uranium hydride present).

A fuel passivation step, in which the fuel is exposed to a 98% argon-2% oxygen atmosphere at temperatures ranging from 423 K to 523 K for about 10 hours. The objective is to create if possible, an adherent passive oxide film on any exposed uranium-metal or hydride surfaces. In the testing runs to date, this step was conducted at near atmospheric pressure.

The evolution of moisture during the tests was monitored continuously by a moisture monitor while the total water release during the experiment was measured by trapping in a drying column. Hydrogen and oxygen in the off-gas stream were monitored by a gas chromatograph. Solid residues (spalled oxides) were analyzed by X-ray diffraction.

Specimen temperature, hydrogen in the off-gas stream, moisture concentration in the off-gas stream, and the oxygen concentration are presented in Fig. 2 for a typical two step drying cycle (Fig. 2a) followed by a conditioning cycle (Fig. 2b). The results indicate an increase in the moisture content of the gas stream for the low temperature drying cycle during the transient heat-up of the specimens, with the hydrogen concentration below detection limit. During the higher temperature drying cycle, both hydrogen and moisture concentration peaked during the transient heat-up of the specimens. The moisture eventually decreased to the detection limit of the moisture monitor, but the hydrogen maintained a steady concentration in the off-gas stream.

Fig. 2

Figure 2b shows the initial depletion of oxygen in the gas stream that occurs during the conditioning cycle. The total oxygen pick-up by the specimens was estimated by integration of the oxygen depletion curve and by specimen weight change.

The results of the furnace testing indicate evolution of both moisture and hydrogen from the fuel specimens. The probable sources of the observed moisture are absorbed water on the metal specimen, oxide layers, and accompanying sludge. The hydrogen, however, could be from uranium hydride decomposition and/or reaction. Recent tests with defueled cladding have ruled out cladding as a hydrogen source. The low limit of solubility of hydrogen in uranium cannot account for the level of hydrogen measured but can contribute a small fraction to the amount of hydrogen measured. The depletion of the moisture source even though hydrogen is being released, and the moisture response to the oxygen addition seems to diminish the likelihood of uranium reaction with gas as the principal hydrogen source.

SAMPLING OF BASIN FLOOR SLUDGE

Associated with Spent Nuclear Fuel is an accumulation of particulate layered material which is generally called sludge. Sludge is found on the basin floors, in canisters, and in the basin pits which are used for miscellaneous tasks such as cask handling. In fact, numerous different types of sludge have been identified depending on which basin, canister type, or pit location that the particular sludge is found. Each type of sludge is a unique nonhomogeneous mixture possibly containing corroded fuel, debris such as windblown sand or insects, rack and canister corrosion products, and/or fission products. All of the various sludges will need to be transported away from the K Basins and disposed with different types of sludge possibly having markedly different disposal paths. Characterization of sludge found on the K East Basin floor and that currently found in one K East Basin pit has been completed and is discussed below.

The central problem addressed in the current sludge characterization effort is "What is an acceptable way to retrieve sludge from the K East Basin (and Weasel Pit) and to process, transport, and store the material until a permanent repository becomes available?" The first part of this effort (i.e.,retrieval, transportation, and processing of the sludge) will possibly require the specification, design, and fabrication of

sludge handling/processing/dewatering equipment or the procurement of similar commercial services. Sludge will, for example, need to be pumped and dewatered with devices such as filters and hydrocyclones. Such an effort will require the knowledge of various physical parameters (i.e., fluid viscosity, particle size, etc.) which could be used directly to design apparatuses and/or could be utilized to specify simulants which can be used for evaluation of candidate equipment. Knowledge of the Special Nuclear Materials content of the sludge will be necessary to maintain accountability of material which leaves the Basins.

The second part of the problem is to designate a storage method whereby sludge can be stored away from K Basins in a more environmentally acceptable area. Two prime alternatives for storage have been identified. These are 1) transferring of sludge to Hanford double shell waste tanks and ultimate disposition along with other tank wastes, or 2) processing the sludge into a form appropriate for solid waste disposal. In these two cases the chemistry of the sludge must be determined, either to ensure compatibility of sludge with any non-fuel waste encountered in the tanks or to ensure that sludge does not contain chemicals which are incompatible with storage as solid waste.

Measurements of sludge depths in the K East Basin and Weasel Pit were reported previously (3) and have shown that the floor is covered with sludge to a depth of 5 to 19 cm (2 to 7.5 inches), with the Weasel Pit containing sludge approaching a meter in depth. Some of the fuel canisters in K East Basin have screened bottoms and slotted sides, and all of the K East canisters have open tops. This means that fuel corrosion products (uranium oxide and fission products mostly) found in canisters have, to some extent, mixed with the expected wind blown debris and corrosion products (from aluminum canisters and steel racks) in basin areas which are in close proximity to canisters.

A campaign to retrieve 20 representative samples of sludge from K East Basin and Weasel Pit has been completed. Locations for sampling were chosen to span a diversity of expected sludge constituents and to supply information needed for sludge removal in a statistically valid manner. Equipment was designed and utilized, Fig. 3, which assured that full representative core samples of material were collected from only specific localized areas. Sludge was found to contain significant iron, aluminum, and uranium and to be flocculent. Most particles are in the sub-micron range with most of the volume in 10 to 50 micron sizes.

Fig. 3

CONCLUSION

Canisters in the K West Basin do contain failed fuel. However the first shipping campaign demonstrated conclusively that serious deterioration of this fuel, even when failed, is not universal. Removal of this fuel, even when failed, is possible with the proper tools. For the specific type of fuel targeted by this examination, deterioration to rubble had not occurred and in fact large amounts of sludge were not observed. That sludge which was observed in canisters had a definite flocculent character and took some time (hours) to settle. Corrosion of fuel is certainly occurring in some of the damaged fuel elements but it is also possible for some exposed fuel to have undergone only minimal corrosion. The predominant constituent of the canister cover gas sampled was hydrogen produced either from radiolysis or more likely from corrosion. A good discriminator for finding canisters with failed fuel appears to be the concentration of cesium in the barrels. Chemical and physical

property data to facilitate transfer of sludge from K East Basin floor have been obtained. Campaigns to recover fuel samples from the K East Basin canisters and sludge from inside of canisters from both basins are planned for the near future.

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VENDOR ASSESSMENTS OF RADIOACTIVE/MIXED WASTE PROCESSING/DISPOSAL FACILITIES

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ABSTRACT

This paper was developed based on Yankee Atomic Electric Company experiences gained through the performance of vendor assessments of radioactive and mixed waste processing and disposal facilities. This paper will provide insights to companies or organizations who have radioactive and/or mixed waste which requires disposal. The paper will discuss the technical, legal and programmatic issues which should be considered when evaluating waste processing and/or disposal options. The discussion focuses on the methods utilized for the preparation, performance and reporting of assessments of waste disposal vendors. The paper includes a discussion of the scope and purpose of the assessment process, and the methodologies and approach taken to evaluate the technical and programmatic areas. This paper provides guidance and direction to those individuals involved in evaluating the capabilities of the waste processing and or disposal vendors. The paper is also a resource which identifies regulatory and industry guidance available for consideration in the planning for a waste disposal/processing vendor assessment.

The purpose of an assessment will primarily be based upon due diligence so as to support control of company production activities which result in the generation of radioactive and or mixed waste product which require disposal. The company needs to determine what is the most practical disposal method. This disposal method may be a combination of waste processing and direct disposal, which is consistent with the methodologies utilized by the Yankee Atomic Electric Company at the Yankee Nuclear Power Station and by other Yankee plants. Due to the anticipated/active need for utilizing a number of vendors for performing these activities. The determination has been made that assessments of the vendors supplying the services are required to ensure that activities are being effectively performed in order to minimize potential liabilities. The assessments need to consider not only the technical aspects of the operations, but also require an evaluation of the quality system(s) being utilized to ensure the consistent and effective implementation of applicable process controls.

ASSESSMENT PREPARATION

One of the first and most important activities in preparing for, and performing a vendor assessment, is developing a scope which is consistent

with the purpose of the evaluation. The key parameters which should be included in the evaluation are determined based on the vendor's activities which will be utilized in support of the waste disposal processes. Depending on the intended utilization of the vendor's services (disposal or processing activities), certain programmatic and technical activities need to be considered in developing the assessment plan. Sample assessment plans are provided in Attachment A (Disposal Facility) and Attachment B (Processing Facility). These assessment plans were developed based on services being utilized in support of the disposal of YNPS radioactive and mixed waste generated during operational and decommissioning activities.

In developing assessment plans, specific consideration must be given to the licensing organization and the associated governing regulatory bodies (State and/or Federal). The licensing agency determines which of the governing requirements the facility must be licensed under to support the vendor's operations. Many facilities have multiple licenses depending on the activities they are performing. Licenses may be issued in support of radwaste processing activities in accordance with 10CFR20, 30, 40 and or 70 requirements. If the vendor is performing mixed waste processing or disposal activities, the facility will be permitted under 40CFR261 requirements to operate a Temporary Storage and Disposal Facility (TSDF), which requires consideration within the assessment plan. Additional licensing requirements could be governed by state agencies, depending on whether the state is an Agreement State. Under these situations, the licenses will be issued to the facility in accordance with state regulations. Waste disposal facilities can be similarly licensed, although the licensing requirements will include 10CFR61 or the Agreement State's comparable regulations. These regulations should be factored in the assessment plan developments.

In addition to the technical requirements being considered, the quality systems which are in place need to be evaluated to determine if sufficient process controls are implemented within the vendor's operations, to ensure an effective and consistent approach to waste-related activities. The facility should have a formal Quality Assurance Program that is consistent with the criteria of 10CRF50, Appendix B. Although required, the program may be the bases for the quality assurance program to meet the quality control requirements of 10CFR61. Additional guidance is provided in NUREG-1293, "Quality Assurance Activities for Near Surface Disposal Facilities." The specific guidance that should be considered in the evaluation is included in Attachment C to this report and can be utilized for developing an assessment plan. The sections that should be included in the assessment plan are the purpose, scope, applicable requirements, assessment dates and makeup of the assessment team.

Once an assessment plan has been developed and the regulatory bases have been identified, the allocation of personnel resources needed to support the assessment should be determined. Since the waste generator has potential liabilities, e.g., potential Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) responsibilities, the assessment team should comprise both technical and quality assurance personnel. The assigned technical specialists should have expertise in the activities being evaluated. Ideally, the assessment team should include an Assessment Team Leader, an expert in licensing activities (radiological and hazardous waste), an individual with expertise with

radwaste and mixed waste, an individual with expertise in geology and hydrology (waste disposal facility only), and an individual with expertise in radiological effluent and environmental monitoring. This assessment team is consistent with the teams utilized by YAEC to evaluate waste disposal facilities. Smaller assessment teams have evaluated waste processing facilities. Consideration should also be given for the development of teaming arrangements, in which several facilities can provide resources to perform the vendor assessment. Teaming arrangements improve the depth of the assessment, while reducing costs to the clients. Yankee's experience in utilizing expertise from other waste generators in teaming arrangements has proven to be an effective method for performing these assessments.

There are many resources that may be considered in developing the assessment plan. See Attachment C for a list of documents that have been considered during YAEC vendor evaluations.

It is important to obtain the key documents prior to performing the vendor assessment. The vendor should be contacted approximately 30 days prior to performing the assessment to establish a point of contact. This individual will usually be a quality assurance representative, but may also be a customer service representative. The assessment plan should be sent to the vendor to facilitate obtaining information, in preparation for the assessment. The following documents are important in preparing for the assessment and should be obtained prior to the start of the assessment:

- Radioactive Material License
- Quality Assurance Program
- Index Of Vendor Procedures
- Organizational Chart
- State Regulations for Agreement States
- Environmental Assessment Report (Disposal Facility Only)
- Design Documents
- Facility Capability Document
- Associated Purchase Order

After obtaining these documents, a checklist may be developed to incorporate specific program requirements to be considered during the performance of the assessment. Once the assessment plan and checklist have been developed, a meeting with the assessment team members should be conducted to ensure that the expectations for the assessment are clearly defined and understood by all participating individuals.

ASSESSMENT PERFORMANCE

The initial phase of the assessment should consist of conducting an entrance meeting to ..... This meeting is an integral part of the assessment because there is limited time and there are a number of activities to be evaluated. At this meeting the purpose and scope of the assessment should be presented to management. The assessment team members should be introduced and the specific activity that each one is assigned should be discussed. The vendor contacts for the assessment should be established at this time. The schedule for the assessment should be conveyed to the vendor's contacts. The team should have a well-defined approach for the assessment. At this time, the vendor's representatives should be provided with a list of additional documents that may be considered for review and evaluation during the assessment. While the vendor is preparing the documents requested, the assessment team should be provided with a tour of the facility.

During the tour of the facility, the team should evaluate the vendor services being considered for use. The tour may need to be limited in scope based on time constraints for the assessment. The tour provides an opportunity for all assessment team members to become familiar with the facility and the overall setup of the organization. The tour of the facility will also provide an opportunity for the assessment team to ask questions about the capabilities of process equipment and operators. It also enables assessment team members to identify areas that may require further evaluation. Once the tour has been completed, the document review should be initiated. These reviews should include a variety of technical and administrative procedures and records. The following is a sample of documents that should be considered in the review:

- Internal Quality Assurance Audits and Inspections
- NRC and/or State Inspection Reports
- Vendor Audits from Other Facilities
- Radiological Effluent and Environmental Reports
- Nonradiological Environmental Monitoring Reports
- Geologic and Hydrologic Studies of the Facility Site
- Financial Reports
- Sureties and Closure Funding
- Operating and Administrative Procedures
- Organizational Charts and Responsibilities
- Quality Assurance Program

At this time, the assessment team members should begin evaluating the activities assigned as determined during assessment planning. The team members should attempt to gather sufficient information to determine whether the vendor's programs and processes are sufficient to ensure compliance with regulatory and state requirements. The review of documents and records should also evaluate if the programs are sufficiently established to assure the vendor's compliance with the applicable purchase order requirements. This evaluation needs to consider where potential liabilities may exist and where the vendor may have potential weaknesses. While assessing the vendor's programs and processes, it is important to ensure that sufficient information is gathered to support the development of the assessment report. This information should support the overall conclusions of the audit, including whether the vendor is capable of effectively providing the services being considered. Refer to the Audit Plans in Attachments A and B for specific activities that should be considered in the evaluation.

Quality Assurance Program Evaluation

Evaluation of the vendor's Quality Assurance Program is a very important facet of the assessments. The initial part of this review should be performed by evaluating internal and external audits and assessments, if possible. The review of these documents will enable the assessment team leader to quickly determine the quantity and types of issues being identified by the vendor. (Whether the issues are technical or administrative in nature.) The evaluation will also help determine the effectiveness of the vendor's corrective action program. These enable the assessment team leader to assess the Vendor's credibility and may enable the focus of the assessment to be better defined.

The Quality Assurance Program should be evaluated to determine whether the vendor has quality systems in place that will ensure the effective implementation of program and regulatory requirements. These systems should cover the activities being performed at the vendor's facility. The

assessment should focus on the systems being utilized for these vendor services being considered for use. Guidance for Quality Assurance Programs for low level radioactive waste disposal facilities is provided in NUREG-1293. This document provides a good basis for evaluating the adequacy of the attributes of the vendor's Quality Assurance Program. Most facilities will have Quality Assurance Programs that are consistent with the 18 criteria of 10CFR50, Appendix B, which are defined as follows:

- Organization
- Quality Assurance Program
- Design Control
- Identification and Control of Material, Parts and Components
- Inspection
- Test Control
- Control of Processes
- Inspection
- Test Control
- Procurement Document Control
- Instructions, Procedures and Drawings
- Document Control
- Control of Purchased Material, Equipment and Services
- Control of Measuring and Test Equipment
- Handling, Storage and Shipping
- Inspection, Test and Operating Status
- Nonconforming Materials, Parts or Components
- Corrective Actions
- Quality Assurance Records
- Audits, Surveillance and Managerial Controls

The review should consider each of the elements of the Quality Assurance Program, as applicable to the vendor's services being considered. The evaluation should determine if the vendor has developed sufficient controls to ensure the effective implementation of the vendor's processes. The evaluation should be performed utilizing personnel interviews, a review of records and program assessments. There should be sufficient awareness of the Quality Assurance Program requirements and management's expectations to enable the effective implementation of the program. The program should enable the vendor to self-identify and correct program and/or process weaknesses.

Technical Evaluation

The technical evaluation should be performed to determine whether sufficient controls have been developed for the processes being evaluated. The assessment should also determine the effectiveness of the controls which are being utilized for controlling the day to day operations of the facility. The evaluation should be performed utilizing personal interviews, a review of procedures and records, observations of activities in the field, and tours of the facility. The technical evaluation should focus on issues that are important to the safe and effective operation of the vendor's processes. For example, if the vendor is supplying waste disposal services to your facility, the evaluation should determine if the vendor has procedures in place for receipt, tracking and preparation of the waste for processing and/or disposal. The vendor should have individuals who are trained and qualified to perform the activities defined in the procedures. The vendor should be performing the processing and or disposal activities as described in the vendor's

procedures, regulatory requirements and license commitments. The vendor should have records that support and document the performance of these activities. An effective methodology in performing an evaluation similar to this is to select a small sample of similar type material as your facility's waste, review the records from arrival through processing to disposal, including verifying the current location. This provides assurance that the processes are in place for tracking waste through disposal. Individuals involved in the receipt and handling of waste prior to processing and disposal should be interviewed. Observation of any activities that are ongoing at the time of the assessment should be conducted. This may be performed at any phase of the assessment and will provide some assurance that the vendor is utilizing the programmatic controls that have been developed. Observations can also be utilized to determined if the individuals performing the work are knowledgeable of the programs controlling the activities they are performing. This approach can be utilized for any vendor service being evaluated, whether it is mixed/radioactive waste processing, packaging or disposal activities.

During the performance of the assessment it is very important for the team leader to keep in constant communication with each of the assessment team members. At the conclusion of each day, the assessment team leader should conduct an informal meeting to discuss the progress of the assessment. Additionally, any concerns should be discussed amongst the entire team, along with the next days' scheduled activities. The assessment team leader needs to ensure that the assessment is progressing on schedule and that the entire scope of the assessment will be completed, based on the time constraints of the assessment. During these meetings, consideration needs to be given to the effectiveness of the programs and processes being evaluated. The discussions should also include a determination as to whether any findings have been identified and or any concerns require further evaluation.

The team leader should maintain good communications with all assessment team members also to ensure the effective preparation for the close-out meeting which is conducted at the conclusion of the assessment. The close-out meeting is conducted with the vendor's management and quality assurance representatives. This meeting is an important part of the assessment performance because it provides a forum for the assessment team and the vendor management to discuss any issues that were identified during the assessment. The meeting also provides an opportunity to listen to any proposed corrective actions being considered by the vendor. The adequacy of the responses and the attitude of the vendor's management should be considered in determining whether the vendor's services can be utilized. The close-out meeting also provides an opportunity for the assessment team to discuss the impressions of the facility and its operation. At this meeting, a documented list of concerns should be provided to the vendor's management. It is important to utilize at least two (2) categories of concerns, e.g., Findings and Observations or Recommendations. This provides the assessment team with an opportunity to present not only any violations of program or regulatory requirements, but also allows the team to indicate areas that may require enhancements. At this time the vendor should be informed of the process for assessment reporting and follow up corrective action response and verification. The vendor should be informed of the time table expected for the report issuance and when any vendor responses would be expected.

ASSESSMENT REPORTING

The method of reporting the results of the assessment is vitally important, in order to properly reflect the evaluations performed by the assessment team. Reporting can be difficult, if the approach to documenting the evaluation was not determined during preplanning. The assessment may be documented in several different ways depending on who will be reviewing the report and how the information will be used. The documentation may be in the form of an elaborate checklist, which is utilized during the performance of the assessment. The assessment team leader may decide to utilize a report format that is primarily narrative in form, or the report may consist of some combination of the two methods. If the assessment is being used primarily for internal use, the detailed checklist may be the least time consuming approach for documenting the results of the assessment. However, if the report is intended for external distribution, then a detailed presentation may be the most appropriate reporting mechanism. In either case, a formal assessment report should be developed which delineates the effectiveness of the processes which were evaluated and whether the programs are consistent with regulatory and license requirements. The report should also discuss the adequacy of the Quality Assurance and Technical Programs which were evaluated and the adequacy of the processes utilized to implement the vendor's activities.

The assessment report should be completed in a timely manner to facilitate rapid notification and quick resolution of any concerns identified during the assessment. The distribution of the assessment report should be similar to that of the assessment plan, including cognizant groups in the assessment results, e.g. the user department, purchasing, management, etc. The report may be issued to the vendor, but is at the discretion of the evaluating organization. However, any identified findings and/or observations should be issued to the vendor to ensure the concerns can be evaluated and corrective actions can be taken. The report should include the following:

- Assessment of the Effectiveness of the Activities Evaluated
- Assessment Team Members
- Purpose
- Scope
- Details of the Assessment
- Personnel Contacted
- Findings, Observations and/or Recommendations

Each of the activities described in the assessment plan should be identified and discussed in the "Details of the Assessment" section of the report. The depth and details of the assessment report should have been determined during the planning phase of the assessment and should be the basis for the development of the report format.

Upon issuing the report to the vendor, responses to any concerns identified should be requested within a specified time period, e.g., 30 working days. The responses should be submitted to the assessing organization for review and approval. For significant concerns, consideration may be given to performing a follow-up evaluation to verify the effectiveness of the corrective actions. The final phase of the reporting process includes documentation of the assessment responses, evaluation of the proposed corrective actions for adequacy, verification that corrective actions and final closure of any concerns have been identified during the assessment. The documentation and reporting of the

assessment allows some flexibility, as long as the reporting process and methodology is defined up front and ensures closure of any identified concerns.

ATTACHMENT A

WASTE DISPOSAL FACILITY ASSESSMENT PLAN

PURPOSE:

The purpose of the assessment is to evaluate the effectiveness of the vendor's operations for adequacy and effectiveness of the quality controls, management practices and supporting processes governing the land disposal of radioactive/mixed waste(s).

SCOPE:

The assessment will evaluate the adequacy of the handling, processing, and disposal of radioactive and mixed waste(s), as well as evaluating both the current and historical execution of the Quality Assurance Program. The assessment will include the following elements:

Adequacy and effectiveness of the vendor's PROCESS CONTROLS for receiving, handling and burial of radioactive waste.

Adequacy and effectiveness of the vendor's GEOLOGIC AND HYDROLOGIC MONITORING Program.

Adequacy and effectiveness of the vendor's RADIOLOGICAL ENVIRONMENTAL AND EFFLUENT MONITORING Programs.

Adequacy and effectiveness of the vendor's PROCESS CONTROLS for receiving, handling, processing and burial of mixed waste.

Adequacy and effectiveness of the vendor's QUALITY ASSURANCE Program and its implementation.

Adequacy of the vendor's site CLOSURE PLAN.

APPLICABLE REQUIREMENTS:

State and Regulatory Requirements (10CFR, 40CFR, 49CFR, Agreement State Regulations,...)

(Reference Appendix C for Additional Guidance.)

ATTACHMENT B

WASTE PROCESSING FACILITY ASSESSMENT PLAN

PURPOSE:

The purpose of the assessment is to evaluate the vendor's operations for adequacy and effectiveness of quality controls, management practices and supporting processes governing the processing of radioactive and mixed wastes.

SCOPE:

This assessment will evaluate the handling, processing, storing and shipping of radioactive and mixed waste(s), as well as evaluating effectiveness of the implementation of the vendor's Quality Assurance Program. The assessment will include the following activities:

Adequacy and effectiveness of the vendor's ADMINISTRATIVE CONTROLS for activities being performed.

Adequacy and effectiveness of the vendor's WASTE TRACKING SYSTEM for the entire processing evolution.

Adequacy and effectiveness of the vendor's RADIOLOGICAL SURVEYS AND MEASUREMENT capabilities.

Adequacy and effectiveness of the vendor's RADIOLOGICAL ENVIRONMENTAL AND EFFLUENT MONITORING Programs.

Adequacy and effectiveness of the vendor's PROCESSING METHODS being considered for use.

Adequacy and effectiveness of the vendor's MIXED WASTE OPERATIONS, including facility design, safety, security, management and facility history.

Adequacy and effectiveness of the vendor's QUALITY ASSURANCE Program.

APPLICABLE REQUIREMENTS:

State and regulatory guidance (10CFR, 40CFR, 49CFR, Agreement State Regulations ...)

(Reference Appendix C for additional guidance.)

ATTACHMENT C

ASSESSMENT REFERENCE DOCUMENTS

COMMON REFERENCE MATERIAL FOR WASTE DISPOSAL AND PROCESSING ASSESSMENTS

State or federal radioactive material licenses (Could vary in number)

Facility Quality Assurance Program

State Regulations for Agreement States

Federal Regulations (10CFR20, 10CFR30, 10CFR40, 10CFR61, 10CFR70, 40CFR)

COMMON REFERENCE MATERIAL FOR WASTE DISPOSAL FACILITIES

"Nonradiological Ground Water Quality at LLW Disposal Sites," NUREG-1183, 1986

"Geochemical Studies of LLW Disposal Sites: Topical Report," NUREG/CR-4644, 1986

"Data Input Guide for Swift II, the Sandia Waste-Isolation Flow and Transport Model for Fractured Media," NUREG/CR-3162

"Theory and Implementation for Swift II, the Sandia Waste," NUREG/CR-3328

"Hydrologic Factors in the Selection of Shallow Land Burial Sites for Disposal of Low Level Radioactive Waste," USGS Circular 973

"Standard Format and Content of Environmental Reports for Near-Surface Disposal of Radioactive Waste," Regulatory Guide 4.18, 1983

"Environmental Standard Review Plan for the Review of a License Application for a Low-Level Radioactive Waste Disposal Facility," NUREG-1300, 1987

"Quality Assurance Guidance for Low-Level Radioactive Waste Disposal Facility," NUREG-1293, 1987

"Special Nuclear Material Inspections at Near Surface Low-Level Waste Disposal Facilities in Agreement States," NRC Inspection Manual, Inspection Procedure 84100, 1992

"Low-Level Radioactive Waste Storage," NRC Inspection Manual, Inspection Procedure 84900, 1994

"Near-Surface Low-Level Radioactive Waste Disposal Facility Inspection Program," NRC Inspection Manual, Inspection Procedure 2401, 1991

"Standard Review Plan for the review of a License Application for a Low-Level Radioactive Waste Disposal Facility," NUREG-1200, 1987

"Site Suitability, Selection and Characterization," NUREG-0902, 1982

"Environmental Monitoring Programs for Low-Level Waste Disposal Facilities," NUREG-13, 19

Session 38 -- POSTER - LLW/ILW

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38-1

EFFECT OF VARIABLE RECHARGE RATES ON
GROUND WATER TRANSPORT OF RADIONUCLIDES
AT AN LLRW DISPOSAL FACILITY

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ABSTRACT

For the proposed Central Interstate Compact (CIC) low-level radioactive waste (LLRW) disposal facility, simulations of transport of radionuclides in ground water with variable annual recharge were made to compare the results with those obtained when a constant, "bounding" recharge was used. This was done to ensure that the constant "bounding" recharge resulted in bounding concentrations of the radionuclides in ground water.

To generate a long-term annual series of recharge rates, the calibrated recharge rates for the years for which ground water level data exist were first correlated with annual precipitation using regression analysis. It was found that the logarithm of the recharge rate for any year correlated closely with the precipitation of that year and the preceding two years. Next, a long-term series of annual precipitation was generated by correlating the existing record at the site with the longer record at a climatically similar station. This, together with the regression relation between recharge rate and precipitation was then used to generate a long-term series of annual recharge rates.

Analytical solutions of ground water transport equations for a system representing the actual system were used to generate concentration histories for various nuclides for both the variable and constant recharge cases. It was demonstrated that the concentration history of a radionuclide for constant, "bounding" recharge rate bounded the concentration history for the variable recharge rate.

INTRODUCTION

A low-level radioactive waste disposal facility is being proposed for the Central Interstate Compact (CIC) member states consisting of Nebraska, Kansas, Oklahoma, Arkansas, and Louisiana. The proposed waste disposal facility will be located on a 100-acre (0.4 km2) site in Boyd County, Nebraska, near the town of Butte.

To support the site characterization and the performance assessment analysis, a regional and a local three-dimensional flow model and a two-dimensional transport model were developed. A key calibration parameter of the flow models was the deep percolation (recharge) rate. The region in which the Butte site is situated is semiarid, with the hydrologic regime characterized by low annual rainfall and relatively high potential evaporation. Recharge rates are typically low. Calibration recharge rates were in the range of 0.04 to 0.1 in/yr (0.1 to 0.25 cm/yr).

To provide a bounding case analysis, hydraulic conductivity values in the flow and transport models were set equal to the 95 % upper confidence

limit of the geometric mean of all measured values (i.e., given the number of hydraulic tests performed within each layer, the probability that the mean does not exceed the conductivity value used in the model is 95 percent). Furthermore, the bounding case was calibrated by requiring that the rate of recharge be high enough to produce a high water table very near the ground water surface and remains there at all times. This condition is considered to be extremely conservative assumption because the water table at the site is known to fluctuate, and in average years it has been observed to be approximately 8 feet (2.4 meters) below the ground surface. The recharge rate needed to sustain these conditions was estimated to be 0.18 in/yr (0.46 cm/yr) which is more than 4 times the base case recharge rate.

Although the bounding case analysis provides a long-term high level of conservatism, it does not specifically account for individual years of extreme rainfall when the actual deep percolation rate might exceed the bounding case recharge rate of 0.18 in/yr (0.46 cm/yr). The purpose of this paper is to assess the impact of variable annual recharge rates on radionuclide transport by comparing radionuclide concentrations obtained with the constant bounding case analysis to those obtained with a variable rate of recharge. This analysis consists of the following steps:

Estimation of annual recharge rates for the individual years of the period of water level observations (1990 through 1994) using the local flow model

Regression analysis of estimated recharge rates and precipitation data

Generation of long-term annual time-series of recharge rates.

Transport simulations with the variable recharge rates.

ESTIMATION OF THE ANNUAL RECHARGE RATES FOR THE YEARS 90-94

The model was calibrated for recharge year by year for years 1990 through 1994, by matching the time-averaged yearly observed well water levels. For these yearly simulations, the boundary conditions in the local model were altered only to account for the yearly prevailing climatic stresses. These simulations were made using an exponential evapotranspiration (ET) function for the dependence of ET on the depth to the water table. The calibrated recharge rates are shown in Table I.

Table I

GENERATION OF LONG-TERM RECHARGE RATES

Extension of the Butte Rainfall Record

The 46-yr rainfall record of Butte (1949-1994) is considered short relative to the time scale of the transport simulations for the performance assessment which is in the thousands of years. Since there are no readily available climatic records of this scale within the region of the site, the extension of the Butte rainfall record was sought. The advantages of extending the record is that longer record may contain yearly extremes, and dry or wet sequences not seen in the 46-yr record. Therefore, Butte record was extended to year 1899 based on the rainfall record of Atkinson, Nebraska using a linear regression equation developed between the records of these two gages. The Atkinson gage is located about 25 miles (40 km) south of Butte at latitude 42 degree and 33 minutes and longitude 98 degrees and 58 minutes, and at elevation 2130 feet (650 m) MSL. The coefficient of determination for the linear regression was 0.8, indicating that 80 percent of the variability of the annual rainfall amounts at Butte can be explained by the regression equation. The annual statistics of the extended record shown in Fig. 1 are quite similar to the 46-yr record at Butte. The minimum and the

maximum of the 96-yr record both occurred within the period of site observations: the minimum in 1989, and the maximum in 1993. These extreme events have approximately a 100-yr return period. Also the 1992-1993 two-year period had the highest rainfall for any consecutive two-year period of the 96-year record.

Fig. 1

Generation of Long-Term Recharge Rates

The recharge rates generated by the local model for 1990 through 1994 were 0.055, 0.048, 0.115, 0.75 and 0.51 in/yr (0.14, 0.12, 0.29, 1.91 and 1.3 cm/yr) respectively. The rate of recharge depends on precipitation, runoff, evapotranspiration and antecedent soil moisture conditions. The only one of these parameters which is directly measurable is precipitation. Antecedent moisture conditions depend on precipitation in the preceding year or years. Therefore it is reasonable to assume that the rate of recharge in a particular year depends on precipitation during that year and the preceding years. This dependence was evaluated by performing a linear regression analysis between recharge and precipitation at every year and the two preceding years, i.e. by assuming that:

Eq. 1

The coefficient of determination of the multiple linear regression was 0.82. To explore if it is possible to obtain a relationship with a higher coefficient of determination, a multiple linear regression analysis was performed on the precipitation and logarithm of the recharge:

Eq. 2

The regression coefficient of determination was 0.96, indicating that 96 percent of the variability of the annual recharge rates is explained by the regression relationship. The high regression coefficient of determination suggests that the dependence of recharge on precipitation is exponential. The regression coefficients were B=-7.01967, b0 = 0.0656, b1 = 0.06066, and b2 = 0.07493.

Therefore, the nonlinear multiple regression relationship thus established between the recharge rates and the annual rainfall amounts was used to model the variability of recharge rates. The generated relationship accounts for both the nonlinear dependence of recharge on water levels as well as the for the dependence of the water levels on antecedent rainfall (storage or persistence effects).

Based on the annual rainfall record for the 1899-1994 period at Atkinson and using the above equation, annual recharge values were generated for Butte, and are shown in Fig. 2. The maximum recharge rate is 0.75 in/yr (1.91 cm/yr) and the minimum is 0.03 in/yr (0.08 cm/yr). The mean recharge rate is 0.12 in/yr (0.3 cm/yr) with the lower and upper 95% confidence limits of 0.097 and 0.137 in/yr (0.25 and 35 cm/yr). The median of the data is 0.092 in/yr (0.23 cm/yr), with the lower and upper 95% confidence limits of 0.078 and 0.102 in/yr(0.2 and 0.26 cm/yr). The distribution of the recharge rates can be best approximated by the lognormal distribution, therefore the long-term average recharge rate is best estimated with the median of the data which is 0.09 in/yr (0.23 cm/yr). The maximum recharge rate which occurred in 1993 has a 100-yr return period approximately.

Fig. 2

TRANSPORT SIMULATIONS

Mathematical Derivation

If the rate of recharge varies from year to year, the mass flux of nuclides leaching from the waste area also varies with time. A simplified and conservative description of nuclide transport under these conditions can be obtained by neglecting transport through the vadose zone and assuming that nuclides from the waste storage area reach the water table instantaneously. If we assume that the flow through the saturated zone remains constant in time, and that the nuclides reaching the water table mix vertically within the uppermost saturated layer providing a pathway to the site boundary receptor, then the variable-in-time recharge and associated nuclide flux produce a variable concentration source at the water table, with the concentration varying in proportion to the rate of recharge. Under these assumptions the effect of a variable source on concentrations at the site boundary receptor can be evaluated with the aid of an analytic solution of the one dimensional transport equation. This equation provides a simplified description of transport in the saturated zone under the assumption that flow is uniform, the aquifer is homogeneous, and there is no lateral dispersion. The transport equation used for this purpose accounts for advection, dispersion, decay and retardation. The solution of the general one-dimensional transport equation and its solution are listed below (1).

The partial differential equation describing the one-dimensional advective-dispersive solute transport is

Eq. 3

Equation (3) can be solved for the following initial and boundary conditions:

Eq. 4

Eq. 5

Eq. 6

where

l decay constant [T-1],

Ca time-independent component of the boundary concentration, and

Cb the time-dependent component of the boundary concentration [M/L3].

The analytical solution of Equation (3) subject to the initial and boundary conditions described by (4)-(6) is given by van Genuchten and Alves (1)

Eq. 7

Because of the linear nature of the advection-dispersion equation, the solution for a time variable source, e.g. for a year-to-year variable source concentration, can be obtained with the aid of the principle of superposition. The solution of the transport equation for a variable source obtained by superposition is given in the following section.

For the special case where:

a. initial concentration Ci=0,

b. production rate g=0,

c. boundary concentration c(0,t)=C0e-lt (i.e., Ca=0), and

d. decay rates m=l,

the solution to the one-dimensional advective dispersive equation is given by:

Eq. 8

Furthermore, when there is no retardation (i.e., kd=0 and R=1), w=v and the concentration is given by:

Eq. 9

For the case with no decay (l=0) Eq. (8) is reduced to the solution given in page 391 of Freeze and Cherry (8).

Eq. 10a

When the boundary concentration is variable with time and, at the same time, undergoes decay, the following model can be used to represent the boundary concentration:

Eq. 10b

where j = 1,...,N is a time index and N is the total number of time steps. Because Equation (3) is linear the principle of superposition can be used to estimate the concentration function c(x,t). This can be achieved by representing the system by a series of individual boundary concentrations (Cj - Cj-1) for the time intervals [tj-1,tj] and summing up the contribution of each:

Eq. 11

where Dtj=t-tj is the time since the concentration c(o,t) = Cje-lt is imposed at the boundary.

Simulation Results

The analytic solution was used to compare concentrations at the site boundary receptor for a time dependent source scaled in proportion to the year-to-year variable recharge and for a constant source scaled in proportion to the recharge rate for the bounding case analysis (0.18 in/yr, 0.46 cm/yr). It is noted again that this calculation accounts for transport through the saturated zone only. The additional travel time and attenuation through the vadose zone will lead to lower concentrations than those obtained from the calculations.

The variables used in this simulation are representative of the conditions observed at the site. Specifically the flow velocity was assumed to be equal to 3 m/year (~10 ft/year) and the dispersivity equal to the longitudinal dispersivity value of 5 m. The source concentration was assumed to vary in proportion to the rate of the recharge for the 94 years of the synthesized recharge record. It is also assumed that the 94-year cycle repeats itself. Figure 3 shows the variability of the assumed source over a period of 500 years. Figure 3 also shows the calculated concentrations at a distance of 250 m (~ 820 ft) from the source, a distance representative of the distance between the B/C cells and the north site boundary. The results presented in Fig. 3 are for three combinations of parameters corresponding to:

Fig. 3

- non-retarded nuclides with very long half-lives, i.e. practically non-decaying nuclides such as iodine and technetium (Fig. 3b),
- carbon-14 with kd=0.1 ml/g and a half life of 5723 years (Fig. 3c), and
- americium with kd=0.1 ml/g and a half life of 459 years (Fig. 3d).

Figure 3 shows also the solution for a constant-in-time source corresponding to a recharge rate of 0.18 in/yr (0.46 cm/yr). The concentrations presented in Fig. 3 are normalized by the initial concentration of a source associated with a nuclide mass flux form the waste area corresponding to a recharge rate of 0.18 in/yr (0.46 cm/yr). As can be seen from Fig. 3, in all cases the solution obtained for the constant recharge of 0.18 in/yr (0.46 cm/yr) is bounding the solution for the year-to-year variable recharge rate, despite the fact that few individual recharge values, representing extremely wet years, are higher than 0.18 in/yr (0.46 cm/yr).

CONCLUSIONS

The effect of the annual variability of recharge rates on concentrations at the site boundary receptor was evaluated by combining the results of the local ground water flow model with 94 years of hydrologic data to synthesize a record of recharge rate, and then employing an analytic solution of the one-dimensional transport equation to calculate concentrations under two different conditions. Concentrations were estimated first for a year-to-year variable mass flux of nuclides, proportional to the synthesized annual recharge rates, and then for a constant mass flux corresponding to the recharge rate of 0.18 in/yr (0.46 cm/yr), used in the bounding case analysis. These calculations account for transport through the saturated zone only. The additional travel time and attenuation through the vadose zone will lead to lower concentrations than those obtained from the calculations. The calculations were repeated for several combinations of parameters corresponding to those that describe the major contributors to the dose, i.e. technetium, iodine, carbon-14 and americium. In all cases the calculated concentrations for the recharge rate of 0.18 in/yr (0.46 cm/yr) were higher than the peak concentrations for the variable recharge rate, even though few individual recharge values, representing extremely wet years, were higher than 0.18 in/yr (0.46 cm/yr).

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HANFORD LOW LEVEL WASTE MELTER TESTS

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ABSTRACT

Approximately 230,000m3 of defense nuclear wastes is stored in underground tanks at the US Department of Energy site in Hanford, Washington. Retrieval and pretreatment will lead to a low level waste stream that contains sodium nitrate and nitrite salts in a highly alkaline liquid slurry. Westinghouse Hanford has been evaluating alternative vitrification technologies for treating this low level waste stream. GTS Duratek and the Vitreous State Laboratory of the Catholic University of America demonstrated low temperature vitrification (11500C) on the DuraMelterTM 100 and 1000 joule-heated vitrification systems. The Hanford LLW simulant was successfully vitrified at sustained feed rates that were twice the nominal capacity of the melters. Approximately 610 kg and 10,700 kg of glass was produced in the DuraMelterTM 100 and 1000 tests, respectfully. All glasses produced far exceeded stated leach resistance requirements. The off-gas system performed effectively with

reduced nitrogen oxide emissions, and final particulate and metal emissions from the process were all below measurable and regulatory limits.

INTRODUCTION

There are 177 underground storage tanks storing approximately 230,000 m3 of defense nuclear wastes containing 242 MCi of radioactivity at the U.S. Department of Energy (DOE), Hanford site in Washington State. The Tri-Party Agreement (TPA) signed between the U.S. Environmental Protection Agency (EPA), the DOE, and the Department of Ecology, Washington State, calls for separating this waste into a high-level and low-level radioactive fractions and then stabilizing these fractions. The low-level radioactive waste stream will consist mainly of sodium nitrate and nitrite salt in a highly alkaline liquid/slurry. Vitrification has been selected as the treatment technology for this waste stream which will produce approximately 400,000 metric tons of vitrified waste. Seven vendors, selected by Westinghouse Hanford Company (WHC) conducted tests of several alternative vitrification methods on a non-radioactive surrogate for this waste.

This paper presents the findings obtained from the melter tests using the WHC supplied simulant performed by the Vitreous State Laboratory of America of the Catholic University of America (VSL) and GTS Duratek, Inc. (GTSD). These tests were carried out in GTSD DuraMelterTM 100 and DuraMelterTM 1000 vitrification systems at the VSL. The first test was carried out on in late September 1994 with the DuraMelterTM 100 and satisfied a TPA September 30, 1994 milestone.

TEST OBJECTIVES

The primary objectives of the test were:

Conduct proof of principle tests to demonstrate that the DuraMelterTM vitrification systems will process a highly alkaline, high nitrate/nitrite, LLW feed and produce a glass of consistent quality.

Demonstrate a practical and reliable feed system.

Demonstrate the ability to produce a durable, homogenous glass with a target composition.

Determine specific requirements for feed preparation, secondary waste, and off-gas treatment systems.

Describe general operating behavior during the tests including upsets and operating problems.

Collect mass balance data across the system for potentially volatile components such as Cs, Na, B, K, Mo, to determine partitioning of these components between the glass, condensed deposits, off-gas entrained particulates, and scrub solutions.

Measure NOx and SOx concentrations, flow rates, and quantities and composition of entrained particles.

Collect data to assess melt characteristics including phase separation, foaming events, cold cap behavior, etc.

Collect glass samples throughout the tests for assessment of glass composition consistency and uniformity as well as leach resistance testing. Collect feed, product glass, off-gas scrub solutions, and off-gas sampling every few hours after steady state operation achieved to supply WHC for analysis and archiving.

Perform pre-test and post-test inspection of equipment to identify solids buildup, deposits, plugging, corrosion, erosion, refractory wear, electrode wear, and equipment damage.

SCOPE OF TESTING

A series of crucible melts was completed first and each characterized for key processing information such as melt viscosity, electrical conductivity and liquidus temperature. PCT tests were conducted to assure that the selected composition met the WHC requirement for a normalized sodium release of below 1 g/m2/d. The results, obtained by the VSL and an independent analysis by the Pacific Northwest Laboratory, showed the glass to be well within this criterion

Two DuraMelterTM systems were used for these tests which allowed a factor of ten scaling (nominal glass production rates of 100 and 1000 kg/day, respectively) in the data sets. The operation of the smaller DuraMelterTM 100 first provided information which was useful in the operation of the DuraMelterTM 1000. The smaller system is large enough that the general behavior is similar to the larger system and is scalable. The power supply for the smaller melter has greater connected power relative to melter volume than the power supply for the larger system. This would have allowed higher relative production rates to be investigated int he smaller system if connected power was the controlling factor. Production rates of over twice the nominal values were comfortably demonstrated and higher rates were certainly achievable.

The DuraMelterTM 100

The DuraMelterTM 100 is a Joule-heated ceramic-lined melter with a nominal glass melting rate of 100 kg/day. The actual melting rate depends upon a variety of factors such as composition of feed, melter temperature, glass properties (viscosity etc), rate of bubbling (mixing), and other operational parameters. This melter is approximately 3 feet by 3 feet with a discharge chamber on one side that is 1.5 feet by two feet. The glass refractory is a Monofrax K3 refractory which forms a 14" by 14" melt pool. The normal glass depth is maintained at 15" and the K3 refractory extends several inches above that level. There are two 1" thick flat plate Inconel 690 electrodes on opposing walls of the melt chamber. The surface area of each electrode is 162 square inches. The resultant glass volume is 2500 cubic in. (41 liters). This is 100 kg for a glass with a density of 2.5 g/cm3. There is a 15" air space between the melt surface and the ceiling. The melter contains an inner shell to prevent leakage of any molten glass into the layers of insulating refractory. The melter has three drains, one is on the bottom for direct discharge of the entire melter, one is a side discharge to drain any floating, secondary phases and the third drain exits the bottom through a side port and riser to a pour trough. This, along with an air lance, forms an airlift, which is activated by bubbling air through the air lance. Glass drains through the pour trough into the waste container that is sealed to the discharge flange.

The melter uses a patented bubbling system to promote mixing and increase the melt rate. This bubbler is designed to produce a curtain of bubbles rising form the melter floor between the electrodes. This not only mixes the glass pool, it also keeps the melt well oxidized.

The off-gas system consists of a melter exhaust film cooler, evaporative quencher, packed bed scrubber, air reheater, heated air dilution port, air-jet bag filter and HEPA filter units. The off-gas systems for both DuraMeltersTM are functionally identical. Both are designed to treat particulate, aerosol, and acidic gaseous emissions other than NOx.

The DuraMelterTM 100 was successfully operated with a slurry feed system as well as the larger DuraMelterTM 1000 system. Several feed additives were evaluated with respect to their effects on emissions and based on

these tests, urea was selected as the primary additive to reduce NOx emissions.

The DuraMelterTM 1000

The DuraMelterTM 1000 is very similar in design to the DuraMelterTM 100 and is more than an order of magnitude greater in size. The nominal production capacity is 1000 kilograms per day but actual production capacity may but much larger depending on feed composition and operating conditions. This is also a Joule-heated melter with a pair of Inconel 690 electrodes on opposing walls. The outside dimensions are approximately 6 3/4 ft. by 6 3/4 ft. By 9 ft tall with a 2 ft. By 4 ft. Discharge chamber on one end. The surface of the glass pool is 42" by 42" and is nominally 38" deep. The melt volume is 67,000 cubic in. (1100 liters) of glass at a density of approximately 2.3. The refractories are contained in an inner shell with penetrations for drains and electrodes. The melter has a bottom drain and the normal drain via an airlift. The off-gas system is functionally similar to the DuraMelterTM 100.

Off-Gas Monitoring and Sampling

Off-gas monitoring and sampling was performed by CUA-VSL staff and a certified air monitoring laboratory. Two different monitoring modes were performed: continuous emission monitoring (CEM) of the gaseous compounds NO, NO2, SO2, O2, CO, and total hydrocarbon concentration (THC), and standard isokenitic sampling for metals and particulates. Periodically, VSL staff monitored HCl and NH3 during the DuraMelterTM 1000 steady state run.

Glass Formulation

A series of formulations consisting of glass-forming additives and chemicals representing the waste stream were prepared. The target glass was to have the following characteristics: waste loading of 25 wt% (corresponding to sodium oxide at 20 wt%); viscosity below 100 poise at melt temperatures between 1100o C and 1150oC; electrical conductivity between 0.3 and 0.5 Siemens/cm at melt temperature; liquidus temperature below 950oC; and normalized Product Consistency Test (PCT) release rates below 1 g/m2/d. Five separate glasses were prepared. All met the above conditions and the one that exhibited superior processing rates and good PCT performance was chosen.

DuraMelterTM 100 Runs

Prior to the start of the first run, the melter was flushed (three turnovers) with the Hanford simulated waste feed to remove the standby glass before the start of the steady-state tests. The turnover runs occurred between September 21-24, 1994 and the steady-state runs between September 26-29, 1995. Approximately 300 kg of glass were produced in the turnover and steady-state periods from approximately 370 liters of feed for each of these runs. The average glass production rate for the steady-state run was 7.7 kg/hr which is above the nominal 100 kg/day production rate. The primary consumption of energy is through the electrodes, the lid heaters and the discharge chamber heaters. The average power usage was 33 kW (4.3 kWhr per kg glass produced).

Two of the glasses produced during the steady-state run were subjected to PCT leach test procedure by the United States Geological Survey (USGS). Leach rates were well below the SRL-EA standard glass. The normalized average leach rates for sodium were approximately one order of magnitude below the 1 g/m2/day goal. Corrosion tests were performed on coupons of the following melter materials: Inconel 690, Inconel 601, K-3 brick, and

Zirmul brick. On visual inspection at the conclusion of the tests, there did not appear to be any sign of corrosion of any of these coupons.

DuraMelterTM 1000 Runs

The melter was flushed with Hanford simulated waste feed for three turnovers to establish the starting conditions for the test. The steady-state run was conducted on January 19-21, 1995. Approximately 7000 kg of glass was produced in the turnover run and 3700 kg of glass was produced in the steady-state run (from 4500 and 3500 liters of feed respectively). The average production rate during the steady-state run was 1800 kg/day, well above the nominal 1000 kg/day production rate. Unlike the DuraMelter 100 run, there was significant drum-to-drum variation in the nature of the supplied simulant. The total average power consumption for the electrodes, lid heaters and discharge chamber heaters was 145 kW or a power consumption rate of 2 kWhr per kg of glass produced during the steady-state runs. This rate is about half of the rate for the smaller melter. The fraction of the total power supplied to the lid and discharge heaters is considerably smaller for the larger melter, as is also the effect from thermal losses. The leach rates from the PCT were well below that of the SRL-EA standard glass and the leach rates for sodium were more than one order of magnitude below the goal of 1 g/m2/day.

NOx Emission Control

The LLW stream at Hanford contains significant amounts of nitrogen in the form of nitrates and nitrites. Various NOx treatment options were investigated in several sets of experiments. NOx removal of over 50% was demonstrated with liquid scrubbing alone although this process is highly dependent on the residence time of the gasses in the scrubbing tower ant he temperature of the scrubbing solution (the lower the temperature the greater the amount of NOx removed). Spraying hydrogen peroxide into the gas stream at vaporizing temperatures was observed to lower the NOx emissions by an order of magnitude. Urea addition also proved to be very effective to lowering the NOx emissions to 5% of the original amount. The effectiveness of the urea addition depended upon the size of the coldcap on the glass. As the cold cap increased to 100%, the NOx emissions increased by a factor of four.

SUMMARY AND CONCLUSIONS

The melters were successfully operated to demonstrate the vitrification of Hanford LLW simulant. The objectives detailed in the WHC test plan were met during the turnover and steady-state runs. The feed rate that was achieved and maintained was nearly twice that originally planned. Excellent material mass balance were achieved for the steady-state runs which shows minimal carry over from the melter. Urea was effective in reducing the NOx emissions-forty to fifty per cent of the nitrogen entering the DuraMelterTM 1000 was decomposed to N2 in the plenum area with no additional engineering controls.

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CYCLONE VITRIFICATION OF HIGH NA CONTENT LOW-LEVEL RADIOACTIVE WASTE

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ABSTRACT

The suitability of a cyclone furnace was investigated for the vitrification of a low level radioactive, high sodium content waste

simulant. Vitrification was accomplished by blending the waste simulant with glass formers prior to thermal treatment. The simulant and glass former recipe was designed to achieve a "target" composition in the glass that was anticipated to yield a glass with suitable properties for long term disposal, a durable glass not susceptible to leaching of any of its constituents. The primary objective, the demonstration of continuous glass production for over 24 hours was successfully completed. The process streams, feed solids, off-gas solids, and glass, were subject to material balances, chemical analyses, and microanalytical analyses, and provided information which will be useful toward selecting process modifications leading to process improvements in future testing.

The off-gas solids (cyclone carryover) are comprised of mechanical carryover and volatile carryover. Mechanical carryover are injected feed solid particles which escape the cyclone furnace through physical entrainment with the combustion gases. Volatile carryover are glass constituents which vaporize off the glass stream at high temperature, enter the off-gas stream and condense at cooler temperatures downstream to contribute to the cyclone carryover. During the Demonstration Test, approximately 32% of the total carryover was due to vaporization, and 68% was mechanical carryover. Therefore, total cyclone carryover was enriched in the volatile constituents B, K, and Na. With 26.7% of the total solid feed partitioning to the cyclone carryover during the Demonstration Test, 72.5% of the B, 54.5% of the K, and 43.0% of the Na, partitioned to the cyclone carryover. Mechanical carryover is expected to have a composition very close to the target composition. However, volatile carryover caused significant deviations of the actual glass composition from the target glass composition. In addition, the high Al content furnace refractory was physically and chemically eroded, and blended into the glass stream. Microanalytical analyses of the glass indicate macroscopic inhomogeneities caused by refractory contamination, but with relatively homogenous regions comprised of glass formed from feed solids. As a result of the deviations from the targeted glass composition, leach rates of several glass constituents which were determined by the Product Consistency Test were actually lower than the ideal laboratory glass with the targeted composition.

INTRODUCTION

The Department of Energy (DOE) Hanford Site, covering a 560 square mile area of Washington state's southeastern desert, was constructed during World War II to produce plutonium for atomic weapons. The storage of large quantities of hazardous and radioactive wastes occurred at this site during the cold war period. This ultimately lead to the Hanford Federal Facility Agreement and Consent Order, also known as the Tri-Party Agreement in 1989 among the State of Washington Department of Ecology, the U.S. Environmental Protection Agency (EPA), and the DOE. This agreement established a multi-phase program to clean up the site over a 30 year period. As part of this program, the DOE contracted the Westinghouse Hanford Company (WHC) to coordinate and evaluate seven technologies to convert 60,000,000 gallons of low level waste, separated off from underground storage tanks, into a silicate glass. The technologies currently under evaluation include joule, slagging furnace, plasma, and carbon arc melters. This paper covers research performed by the Babcock and Wilcox Company on the applicability of the cyclone furnace for the vitrification of the low-level radioactive liquid waste.

The cyclone furnace is a B&W designed high intensity slagging combustor originally designed for the combustion of low grade coals. The cyclone furnace has been used for commercial steam generation since the 1940s. However, more recently, the cyclone combustor has been investigated for its applicability in the thermal treatment and vitrification of hazardous waste. To date, B&W research in cyclone vitrification includes the treatment of dry soil contaminated with organic compounds, dry soil contaminated with organics and radionuclide surrogates, an industrial waste sludge, and a simulated Hanford low-level waste, the subject of this paper. The most comprehensive of the previous studies was the U.S. Environmental Protection Agency (EPA) Superfund Innovative Technology Evaluation (SITE) Demonstration (1).

The Small Boiler Simulator (SBS), a B&W pilot scale facility was constructed to simulate the gas side environment of large utility boilers for the purpose of research and development on equipment utilized on these systems. With the SBS in its cyclone fired combustion mode, slag forms from the inorganic constituents of the coal. Since the cyclone was designed to produce molten slag from the coal ash, it was thought that the cyclone had potential application in the vitrification of hazardous inorganic wastes. Based on the positive results from B&W's early cyclone vitrification studies, the cyclone furnace was assessed for its potential for the vitrification of Hanford low level tank waste using a simulated liquid radioactive waste formulated and prepared by WHC.

Figure 1 illustrates the SBS cyclone vitrification configuration utilized in Phase I of the current program. Vitrification was achieved by mixing the simulated waste with glass formers before introducing the mixture into the cyclone. Glass formers included boric acid, coarse sand, fine sand, hydrated alumina, and limestone. The various glass melter subsystems include the feed system, the cyclone furnace (glass generator), the SBS radiant furnace and convective pass, and the back-end cleanup system including a pulse jet baghouse and dry scrubber. The feed system maintains a well mixed waste slurry (glass formers and waste simulant), and pumps the slurry into a cyclone combustor through a feed injector that propels the slurry toward the hot cyclone walls. The SBS radiant furnace and convection pass extract heat from the flue gas to lower flue gas temperature while boiling water to generate steam. The baghouse removes essentially all particulate, and the dry scrubber removes acid gases from the flue gas. The locations for sampling molten glass, and solids that escape through the convective pass with the flue gas are also depicted in Fig. 1. Chemical analyses of these samples were necessary to perform the material balances presented in this paper.

Fig. 1

OBJECTIVES

The primary goal of Phase I was the execution of a 24-Hour Demonstration Test to generate a vitrified product of a surrogate waste specified by WHC. Additional objectives of Phase I were aimed at providing key information for the WHC assessment of this technology, and include:

1) Formulate a feed with the desired physical properties and composition
2) Characterize the feed and the individual feed ingredients
3) Set up and check out a pilot feed handling system
4) Evaluate glass/refractory interactions and glass build-up in the pilot system
5) Identify the flue gas composition
6) Evaluate volatilization and partitioning of the constituents

7) Determine the glass quality and identify possibilities for improvement
8) Provide technical inputs (engineering and life expectancy/reliability information).

All the objectives were completed successfully, and are covered by Holmes (2). This paper focuses on data analysis pertaining to objectives 4, 6, and 7.

RESULTS AND DISCUSSION - PARTITIONING

In the development of this technology, the partitioning of major and minor glass components between the glass and the off-gas streams is important to characterize in order to establish the basis for off gas treatment and material recycle. The portion of solids material that partition to the off-gas stream is called cyclone carryover, which necessitates off-gas treatment to remove potentially hazardous particulate prior to emission to the atmosphere. The quantity and the nature of the off-gas particulate determine the design of particulate removal equipment necessary for proper performance. Cyclone carryover can be collected for treatment as a secondary waste stream and recycled back into the cyclone feed.

The extent to which glass constituents partitioned to the flue gas (off-gas) rather than the process glass stream was estimated with an overall material balance. The following data was utilized to perform the material balance: 1) calibrated feed rates of simulated waste, fuel, and combustion air, 2) off-gas suspended solid concentrations at the convective pass outlet by EPA method 5, and 3) elemental analyses by Inductively Coupled Plasma (ICP) of feed, glass, and carryover solids. The material balance is described by the following equation.

Eq. 1

Wfsolids: The total estimated solids input from the waste feed as oxides (751.7 kgs)
R: The total weight of the refractory loss
G: The total glass weight collected in the quench tank (576.3 kgs)
CP: The total solids measured exiting the convection pass (131.6 kgs)
g* The fractional weight change of the volatile glass elements by existing as gas phase compounds other than the oxides.
D: The total weight of the deposition on internal surfaces as oxides

Solids that deposited on internal boiler surfaces (D) and refractory loss (R) from the cyclone and lower furnaces walls were not measured quantities. These quantities were estimated by the simultaneous solution of Eq. 1 and Eq. 2 shown below.

Eq. 2

xAl : The average measured mass fraction Al for the glass (0.0761)
yAl The average mass fraction of Al for the EPA-5 particulate samples (0.0292)
WF: The total weight of the injected feed (DAS recorded)
fAl: The average "as fed" elemental Al concentration of the feed (0.0294).

Equation 2 is a material balance of the element Aluminum (Al). Al was chosen since it comprised the major portion of the furnace refractory (Plibrico 88-Special), and consequently provided the best tracer for

refractory contamination. The refractory manufacturer specified an Al content of 0.4455 mass percent. The refractory loss (R) and solids deposition of internal surfaces (D) were estimated as 25.0 kgs and 80.4 kgs respectively. Since measured solids at the convection pass outlet yielded 120 kgs as oxides, the total mass % carryover was 26.7%, with 40.1% depositing on internal boiler surface prior to the convection pass outlet.

Average cyclone carryover for specific elements for the Demonstration Test is shown in Fig. 2, representing the overall material balance. Exact mass recovery (100%) is expected for total mass and Al since Eqs. 1 and 2 must be satisfied when D and R equal 25.0 kgs and 80.4 kgs, respectively. The calculated values of D and R lead to excellent mass recoveries for the other elements. Recovered mass for Na, B, K, Si, and Ca amounted to 98%, 105%, 103%, 98%, and 103% of the feed, respectively. Feed concentrations were very accurate by the fact that the feed recipe composition was confirmed by the close agreement with the elemental feed analyses of feed samples. Since feed rates were also accurately calibrated, the calculated mass of each glass element into the process is very accurate. Consequently, Fig. 2 shows the percent partitioned to the glass and carryover streams normalized to yield 100% recovery of the feed for all elements. Since high temperature vaporization significantly contributes to the cyclone carryover, the percent elemental carryover correlates with the vapor pressure of common compound forms of the respective elements.

Fig. 2

Trace species were selected for inclusion in the waste simulant to act as surrogates for radioactive species in the real waste. These included Cs, Mo, Cl and F. Mo was the surrogate for radioactive Technetium (Tc). There was a detection problem in the fluorine analysis, and thus fluorine carryover is not included in these results. Average cyclone carryover for Cs, Mo, and Cl were estimated for the Demonstration Test as shown in Table I.

Table I

RESULTS AND DISCUSSION - VOLATILE VS. MECHANICAL CARRYOVER

Cyclone carryover is comprised of mechanical carryover and volatile carryover. Mechanical carryover is defined as the solid particles that remain physically entrained in the highly turbulent cyclone fluid flow and exit the cyclone with the off-gas. Volatile carryover pertains to the solid glass constituents that vaporize at high temperature. These off-gases ultimately condense and contribute to the cyclone carryover solids. Insight into what caused cyclone carryover is critical in determining how to minimize it through process modifications. For example, if carryover is predominantly volatile, it is critical to control the process temperature. On the other hand, if mechanical carryover dominates, more attention may be given to an optimal design and placement of the feed atomizers.

A method was developed to approximate the relative fraction of total carryover that was generated by mechanical entrainment versus vaporization. It is assumed that the feed solids initially have the designed target composition before any volatilization occurs. The feed recipe was formulated to produce a glass with the optimal (target) composition, and assumes that no glass constituents will preferentially partition to the off-gas. This target glass composition was determined in separate laboratory studies performed by Pacific Northwest Laboratories.

The measured deviation of glass and carryover solids composition from the target composition was a measure of vaporization off the glass to the carryover, and was represented by "T" in Eq. 3.

Eq. 3

yi : Measured carryover solids mass fraction of element "I".

xt,i : The target glass composition of element "I" assuming formation of the pure oxide glass with no preferential partitioning to carryover solids.

Ti : The elemental mass transfer rate (kg/hr) from the glass at the target composition to the carryover required to achieve the measured particulate elemental concentration, yi.

Ytotal: The total mass flow rate (kg/hr) of carryover.

bj: For Na, B, and K only (major volatile constituents).

j : Mass conversion factor for element j to its assumed compound form in the oxide glass (i.e. Na2O, B2O, K2O).

(Ytotal /g - SbjTj): The mass flow rate (kg/hr) of mechanical carryover as oxides.

The equation above represents six separate equations for each of the major glass elements. Since the volatile loss of Si, Al, and Ca are negligible relative to B, Na, and K, only these volatile elements are included in the summation term. Each solution set "Ti" applies to an hourly data set acquired during the demonstration test or a preliminary scoping tests.

An estimate of mechanical carryover is computed as the total carryover minus the total volatile loss from the glass stream. Of the total carryover, the relative proportions of the mechanical carryover and volatile carryover were estimated as 68%/32% and 63%/37% for the Demonstration Test and the earlier exploratory tests, respectively. Mechanical carryover correlates with volatile carryover as depicted in Fig. 3. The rate of volatile carryover is normalized by the maximum possible elemental mass transfer rate from the glass stream (xt,i*(X+ SbjTj)), where X is the feed solids flow rate to the glass stream.

Fig. 3

The physical basis for this correlation can be explained as follows. Through much of the preliminary exploratory testing, the accumulation of simulated waste feed occurred on internal cyclone surface, often significantly reducing cyclone volume, and on two occasions causing shutdown due to blockage of fuel and air inlets. This was confirmed by post test inspection after these two cases. A modification to the fuel injection configuration performed prior to the Demonstration Test eliminated the problem by more uniformly distributing the heat of combustion inside the cyclone melter. Low feed accumulation cases could be related to relatively high heat transfer measured to the glass melter water jacket during these periods where feed solids were not insulating the internal cyclone walls as much. Tests for which little accumulation of feed solids had occurred on the feed atomizer and internal cyclone surfaces were associated with tests where mechanical and volatile carryover were estimated to be high. Low feed accumulation (an uncontaminated open cyclone) generated relatively high cyclone temperatures and high volatile carryover because this condition allowed the maximum gas residence time for the combustion of more fuel within the

cyclone, and also allowed a fully developed cyclonic flow pattern which intensified the rate of fuel and air mixing. Low feed accumulation (an unimpeded waste feed atomizer) was also favorable to the generation of a relatively fine spray aerosol which is known to increase mechanical carryover. This also increased the internal cyclone surface area upon which the spray impinges and maximized heat transfer to feed solids, increasing volatile carryover. Thus, high volatile carryover correlated well with high mechanical carryover.

It is important to recognize that operating conditions during the Demonstration Test were not optimized to minimize carryover. Feed rates during the Demonstration Test were low relative to the previous exploratory tests. Low feed rates were maintained because there was still concern over the excessive accumulation of feed on internal cyclone surfaces before the modified cyclone furnace natural gas firing configuration proved to be successful. Therefore, as it turned out, the feed rates were set too low, and led to the relatively high gas side temperatures which exacerbated volatile carryover. By merely increasing feed rates in future testing, it is expected that volatile carryover can be significantly reduced.

It is also expected that mechanical carryover can be reduced by optimal design and placement of the feed atomizer(s). The atomizer(s) can be located in closer proximity to the cyclone wall in the tangentially oriented combustion air flow entrance. Atomizer design affects spray momentum, shape and particle size distribution which all should be optimized to achieve uniformity of feed coverage on the cyclone's internal walls, and the minimization of mechanical carryover. Cyclone design changes such as a larger slag tap and/or a smaller exit area can also reduce mechanical carryover.

RESULTS AND DISCUSSION - GLASS QUALITY

Uniformity and homogeneity are desired physical qualities of the process glass. Inhomogeneities might be indicative of a glass that has not completely formed, or has been contaminated by furnace refractories. A measure of glass durability utilized in the evaluation of the technology was the Product Consistency Test (PCT). This is a standard procedure for determining the elemental leach rate from a ground glass sample into a mildly acidic solution. Leach rates were normalized to the sodium content by dividing by the sodium mass fraction.

To assess the homogeneity of the glass, micro-analytical analyses were performed, and included Light Microscopy, Energy Dispersive X-Ray Spectrometry (EDX), and Scanning Electron Microscopy (SEM). Light Microscopy clearly showed dark streaks embedded in a homogenous phase which were pink regions on these images, an example of which is depicted in Fig. 4. High magnification SEM revealed more physical details of the inhomogeneities, characterized as cords, streaks, and inclusions. EDX analyses of these regions together with manufacturer supplied average furnace refractory material compositions suggest these refractories are the source of the inhomogeneities. Physical evidence of refractory loss was also obtained by collecting refractory specimens off the furnace walls, and also by photographing the lower furnace walls and cyclone surfaces after testing.

Fig. 4

SEM/EDX analyses also showed uniform homogeneous regions where macroscopic inhomogeneities did not exist. The elemental spectra measured in these areas showed consistency in the signal strength intensity of the

major glass constituents, and indicate that these areas are comprised of the glass produced in the cyclone. Back scattered electron imaging showed little atomic number contrast, indicating composition uniformity in these areas. However, due to preferential elemental cyclone carryover, there was significant deviation of the glass composition from the target composition as Table II below illustrates.

Table II

The furnace refractory, predominantly Plibrico 88-Special, contaminated the glass to a significant extent due to chemical attack by the molten glass. The material balance indicated that on the order of 10% of the lower furnace refractory blended into the glass stream. Another refractory material, Shamrock, was tested for resiliency by inserting a 6 inch plug on the lower furnace water wall. It did show improved resistance to chemical attack under the conditions of the test.

An intense effort to find the most chemically resistant refractory may not be the approach in the future development of this process. It is believed that significant refractory loss can occur without any detrimental effect to the process because of experience gained with coal slagging furnaces. The glass that blends with and partially replaces the refractory should form a hard protective scale over uncontaminated refractory closest to the water wall. The sharp temperature gradient near the furnace water wall create sufficiently low temperatures for this to occur. Refractory closest to the gas side will degrade and blend into the glass stream until only molten glass comprises this layer adjacent to the combustion gases. Therefore, a steady state condition will be reached where the rate of refractory loss will fall to zero. This point had not been reached during the Demonstration Test since the glass samples acquired toward the end of testing still displayed refractory contamination. However, in future work, the long term effect of the above process on melter performance should be evaluated. Although this glass may not be as aesthetically appealing during the initial period of contamination by refractory, the data shows that this does not appear to adversely effect PCT results.

A low PCT value is an indication of a stable, durable glass. The ideal laboratory glass formulated by PNL (Pacific Northwest Laboratory) underwent PCT analyses on its major constituents, and acted as a benchmark against which the process glass was compared. The negative PCT deviation from the target glass (Table II) is attributed to the carryover of Na and the dissociation of Al from the refractory. Although PCT values can not be predicted for a glass with a full complement of Na when material recycle is incorporated into an integrated process, the Na that was captured in the glass was reasonably well reacted on a micro scale.

CONCLUSIONS

The 24-hour demonstration test was successfully performed, and therefore, feasibility has been established. A glass product was continuously produced for over 28 hours.

The glass was macroscopically non-homogeneous, but was amorphous on a microscopic level. The macroscopic non-homogeneity of the glass is attributed to the blending of the alumina-based refractory into the glass stream.

The Plibrico refractory used in this project was prone to chemical attack by the glass. A Shamrock refractory plug inserted in the system for comparison was more resistant to the molten glass.

Based on sample analyses performed by Corning and USGS labs, the glass samples all had PCT leaching rates better than the laboratory glass.

The mechanical and volatile carryover are estimated to comprise approximately 68% and 32% of the total carryover during the Demonstration Test, respectively. Total carryover was roughly 26.7% by mass of the injected feed solids. Preferential carryover of relatively volatile compounds caused some deviation from the targeted glass composition, but with no adverse effect on PCT leach rates.

Of the major glass constituents, boron, potassium, and sodium showed the highest cyclone elemental carryovers of approximately 72.5%, 54.5%, and 43.0% by mass of injected feed, respectively.

The trace glass constituents Cl, Cs, Mo showed cyclone elemental carryovers of 80.8%, 87.9%, and 59.9% by mass of injected feed, respectively.

The relative cyclone carryover of glass elements followed the relative volatility of many of their gaseous compound forms. Volatile partitioning was most significant for the volatile elements (B,K,Na,Cs,Mo,Cl).

RECOMMENDATIONS

As with commercial systems, refractory loss can be expected until steady-state operation is reached and the refractory is covered by a protective layer of glass or slag. This is typically the manner in which coal-fired utility cyclones operate. Steady state operation in this regard had not been reached in the Demonstration Test. It is recommended this type of operation be tested, with the glass ultimately providing part of the protective scale on the water wall.

Means for mitigating volatile carryover include such possibilities as lowering the operating temperature of the cyclone, recycling of volatile constituents back to the melter, using less volatile glass formers, or sending volatilized constituents to the high level waste process. Phase II testing will determine the necessary adjustments to the feed recipe to accommodate cyclone carryover and solids recycle so that the targeted glass composition can be closely attained.

Physical entrainment losses (mechanical carryover) from the glass were significant. Cyclone design changes and optimization of the feed injection will be used to resolve this issue.

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LONG TERM NATURAL TESTS OF NPP VITRIFIED RADIOACTIVE WASTE

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ABSTRACT
Vitrified intermediate level radioactive waste was obtained during test operation of experimental vitrification plant at SIA "Radon". The behavior of borosilicate glass matrix was investigated in natural conditions simulating shallow ground disposal. Long term tests of vitrified radioactive waste from nuclear power plants were performed in a shallow ground repository as well as on an open testing area. Radionuclide leaching differs significantly in natural conditions in contrast to laboratory behavior. There are many fluctuations -as when the leach rate grows by an order of magnitude- after that it remains almost smooth and slowly decreases as usually. Changes in leaching rates are greater and more frequent for open area tests. Average leaching factors were obtained for open area tests as well as for repository tests. These factors can be used for long term predictions since they take into account the real character of leaching.

INTRODUCTION

Glass now is one of the most useful and utilized material. Historical experience of application of glass gives evidence of high stability of this material and good environmental compatibility. Certain glass specimens on the earth have a few millions age. Glass specimens from the moon are a few hundred million years old.

One of the conventional functions of glass is utilization as packaging for the storage of various materials including dangerous products. Glass was found to be an appropriate host material for the isolation of radioactive waste. Most of radioactive waste components can be included into the structure of glass. Glass incompatible components can be also included into the glass matrix in the form of disperse phase. High physical and chemical durability of glass provides long term retention of radionuclides.

Although glass was initially proposed for high level radioactive waste treatment, now vitrification is considered as possible process for intermediate level radioactive waste immobilization. Equipment to provide waste vitrification in this case is much simpler. Requirements for the glass product are not as stringent as in the case of high level waste. For instance, inhomogeneities in glass are permitted, therefore glass composite materials can be used to immobilize glass-incompatible waste (1). On the other hand, durability of glass provides simplification of disposal facilities, shallow ground disposal being the most suitable. Long term laboratory tests as well as long term in-situ tests of waste glass have great importance for the assessment of vitrified waste behavior. Now there are many well established data on the behavior of waste glass in the case of high level vitrified waste (2-6). Most of these data can be used to understand the nature of glass behavior of vitrified intermediate level waste. Nevertheless some peculiarities of glass products and storage conditions have to be considered in this case. Laboratory testing of glass products is an essential part of waste management program. It provides parameters of glass durability. However natural tests give the most reliable data on the real behavior of materials under conditions close to those of real disposal.

The aim of this paper is to review the results of long term observation on the behavior of vitrified intermediate level waste under conditions close to those of shallow ground disposal. Two types of tests were performed: the first one being in conditions similar to shallow ground

repository, and the second one on a open testing area. These tests were conducted more than 7 years.

SPECIMEN PREPARATION

Vitrification of intermediate and low level radioactive waste is a new perspective trend, providing the maximum waste volume reduction and obtaining a highly durable product. SIA "Radon" began experimental vitrification of radioactive waste in the early 70's. Borosilicate glass was selected as a matrix material for the immobilization of waste components. A ceramic melter with direct Joule heating and capacity up to 50 kg/h for glass mass was used for vitrification Electrical power supply of the system was 150 kW, temperature in melter being 1250C. Many types of wastes with specific activity up to 37 MBq/l and among them wastes from atomic power stations with reactors WWER and RBMK were vitrified. Volume reduction factors for vitrification are 4,2 - 4,5. Losses of radioactive Cs from the melter in the process of melting were not more than 3,5%. Leaching rate for Cs137 from final glass product was within 1,410-5 - 4,410-6 g/cm2day. Total amount of glass produced by ceramic melter constitutes more than 10 tons.

Recently SIA "Radon" has developed a new vitrification method (7). It is based on coreless induction melting of glass in a cold crucible. Glass composite materials, which contain separately radioactive components in the form of disperse phase in the glass matrix, are considered for immobilization of glass-incompatible chemical components (such as sulphates, hard metal oxides, etc.) (8). The new vitrification plant with a capacity of 75 kg of glass per hour is under the test stage.

Specimens for long term tests were selected during the experimental vitrification of real intermediate level radioactive waste from nuclear power plants (NPP). The initial liquid radioactive waste consists of aqueous sludges. About 30 - 40% of waste salts were incorporated into glass The composition of glass product is given in the Table I.

Table I

The main radionuclides in waste were Cs137(63.2 - 82%), Cs134 (17 - 35.1%), Co60 (1 1.6%), Pu239 and Sr90 (less than 0.1%). The parameters of tested specimens (glass blocks) are presented in the Table II.

Table II

Glass blocks were disposed for long term tests during 1987 - 1989. They had a prismatic form.

TESTING CONDITIONS

Two types of testing programs were initiated at experimental testing site of SIA "Radon". The first program comprises long term tests of specimens on an open area. Obviously here the glass is subjected to the maximum influences of erosion and weathering that contribute to radionuclide release (9). These tests give assessment of capability of glass to retain radionuclides in extreme conditions related to the possible accidental opening of waste repository.

Glass blocks (see Table II) were placed on stainless steel trays (52 X 52 cm) at 60 cm height from the surface of the ground. They were able to collect all water which contacted the glass. Atmospheric sediments that contacted waste glass were sampled for chemical and radiometric analysis. Usually water sampling was performed twice per month.

The climatic parameters of the open testing area are presented in Table III.

Table III

The second testing program comprises long term tests in a shallow ground repository. Glass blocks were placed here (see Table II) on stainless steel trays in a loamy soil experimental burial site at the depth 2 - 2.4 m. These blocks were covered first by pure bank sand to exclude direct contact with surrounding soil. After this glass blocks were covered by host soil up to the surface of ground (see Fig.1).

Fig. 1

As distinguished from in-situ tests (3, 6), radionuclide sorption onto soil was eliminated. The trays were supplied by traps for the collection of ground water which contacted the glass. Ground water that contacted waste glass was sampled for chemical and radiometric analysis. Usually water sampling was performed twice per month.

The parameters of experimental shallow ground repository are provided in Table IV.

Table IV

The ground water on the testing place has a hydro-carbonate-sodium-calcium content with a summary mineralization of about 0.5 g/l.

RESULTS AND DISCUSSION

Due to the absence of additional sorption of radionuclides in the host soil (as well as in the case of open area tests) it was possible to calculate leaching rates for every glass block on the basis of the analysis results. In Table V one can see results of long term tests. Here one can compare these parameters for 1 year testing time and 7 years. A special computational method was developed to obtain both the leaching factor, L, and the effective diffusion coefficient, D, for radionuclides in glass (10). The main results of the computations for two specimens are in Table V also.

The leaching factor, L, takes into account the real character of leaching including discontinuities in the leaching rate. Therefore the leaching factor, L, should be used for long term predictions of radionuclide losses.

Table V

The leaching process under natural conditions has some peculiarities when compared to laboratory testing results. The leach rate is higher under open area conditions. Moreover the leaching process is not monotonic. There are many fluctuations. For example, the leach rate grows by an order of magnitude- after that it remains almost constant and slowly decreases in accordance with theoretical prediction. Changes in the leaching rate are greater and more frequent for open area tests.

The changes in the leaching processes can be caused by the generation of new surface regions that contact water. Actually, many small cracks can be viewed on the glass surface after prolonged tests in an open area. In time, they form an entire network over the surface of the glass. Therefore the formation of cracks is responsible for the fluctuating character of radionuclide leaching. Nevertheless one should mention that basically the glass status after prolonged tests remains satisfactory and radionuclide retention is reliable (see Table V).

An important value is the level of contamination of water which contacted vitrified radioactive waste. Obviously this level will be much higher if the water contacts fresh regions of the glass surface. Therefore in the initial stage, when small portions of water contact fresh glass blocks, the contamination level can be high. A similar situation occurs in the case of crack formation when new contacting areas appear on the surface of waste glass.

The specific radioactivity of water which contacted vitrified radioactive waste is shown in Table VI for 1 year testing time and for the 7-th year.
Table VI
Only Cs-137 was detected in the exposed water. As for chemical contamination one can see the results of examination of ion content in water which contacted the glass blocks in Table VII.
Table VII
Changes in the ion composition of water are not significant especially for the main ionic species.

CONCLUSIONS

Long term test of vitrified radioactive waste has given appropriate data on the leaching process and glass behavior under conditions of real near surface disposal. The leaching is discontinuous; however, radionuclide retention by glass matrix remains reliable. Accurate prediction of radionuclide losses can be done by using results of long term tests.

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VALIDATION OF THE MCNP MONTE CARLO CODE FOR GERMANIUM DETECTOR GAMMA EFFICIENCY CALIBRATIONS

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ABSTRACT
Canberra designs a variety of instruments that are required to accurately measure the radioactivity content of large and/or complex sources. The Monte Carlo code MCNP was evaluated to determine if it would be suitable to predict the performance of these instruments. A series of experiments was designed, starting with simple source-detector geometries and becoming increasingly more complex. For each geometry the full energy peak efficiency was computed with MCNP, and also measured experimentally. Multiple energies (nominally 100-1500 keV) were used for each experiment. The lessons learned from each experiment are described, and were suitably incorporated into later experiments.
Computer run time can be quite long, and therefore a geometrical biasing scheme was developed to make these computations more practical. The speed advantage can be a factor of 10 or greater.
It is concluded that this technique is very powerful and accurate when properly applied. For the geometries tested, efficiency calibrations are accurately predicted to within 5% for simple geometries, and up to 15% for complex geometries at low energies.
INTRODUCTION
The primary purpose of this project was to develop a design tool for the evaluation of the various detector-geometry factors as we create new radioactive waste assay systems. These gamma spectroscopy systems are required to measure large and often complex samples. The sample sizes range from 200 liters to 36000 liters. It is necessary to create the optimum detector placement strategy in order to achieve the lowest minimum detection limit and to minimize sample non-homogeneity errors. The traditional methods we have employed in the past (1) have used real detectors of varying sizes, and multiple geometries made with radioactive sources. While suitable for simple [cylindrical] and small [200 liter] samples, this becomes less practical as the sample size and complexity increases. And, the use of radioactive samples for testing and calibration generates radioactive waste.
A potentially useful tool would be Monte Carlo modeling. MCNP (2) is a well known general-purpose Monte Carlo code commonly used for neutron transport applications. It is also capable of modeling photon problems. There have been a few very useful publications describing the use of MCNP to model gamma ray spectra and efficiencies for Ge detectors (3-6). However, these have generally evaluated fairly simple detector-source geometries, or they have not performed direct efficiency calibrations (6). Important issues such as what are the critical parameters, how much detail is necessary in the model, etc. have not been studied, or reported.
At a minimum, the goal of this project was to develop a tool useful for relative comparisons. This tool could then be used for selecting the optimum size, number, type, and placement of detectors. We could then also investigate the relative error associated with non-homogeneity of the sample in the matrix. The ultimate goal, however, is to demonstrate that we can accurately model these complex geometries, and them be able to determine system performance prior to building the first system. If we can accomplish this goal, then we can use MCNP to accurately and economically perform the primary calibration for a wide variety of sample conditions.

In this report, we present the results of a detailed study showing the validity of the use of MCNP to perform efficiency calculations of complex geometries with Ge detectors.

METHODOLOGY

A sequence of different source-detector geometries was created. Six major geometries with a total of 16 different source-detector geometries were modeled for efficiency determinations. These started with simple geometries, and proceeded to increasingly more complex geometries. The geometry was first modeled and the efficiency computed using MCNP. Then a multi-energy calibration source was used to develop a traditional efficiency calibration for the same geometry.

The test geometries modeled and measured for this study were the following:

1. Full energy peak efficiency for a small multi-energy gamma source positioned at a distance of about 40 cm and at 0, 45, and 90 degrees with respect to the axis of a 32% relative efficiency coaxial Ge detector.
2. Full energy peak efficiency for a Eu-152 multi-energy line source of 80 cm in length, positioned 14 cm from the endcap of a 30% relative efficiency coaxial Ge detector.
3. Full energy peak efficiency for a multi-energy planar gamma source 50 x 50 cm positioned 8 cm from the endcap of a 25% relative efficiency coaxial Ge detector.
4. Full energy peak efficiency for a series of single nuclide sources in a 1 liter water-equivalent matrix in a Marinelli beaker with a 25% relative efficiency coaxial Ge detector.
5. Full energy peak efficiency for a Eu-152 point source successively shielded by 0, 1.0, 3.0, 6.1 and 9.1 cm of steel placed between the source and a 25% relative efficiency coaxial detector.
6. Full energy peak efficiency for a multi-energy simulated volume source in a 200 liter (55 gal) drum at 4 different drum matrix densities in a Canberra Q2 Low Level Waste Assay system with 3 nominally 28% relative efficiency Ge detectors.

The multi-energy source had nuclides from 88 to 1836 keV. Eu-152 was analyzed for each of the lines to give data points from 122 to 1408 keV. The Marinelli beaker sources had energies from 60 to 1115 keV.

Each of the 6 experiments was performed separately and sequentially. The two efficiencies [modeled and measured] were compared. Where there were differences that were statistically significant and greater than approximately 5%, both the calculation and the source measurement were examined carefully. Generally, this resulted in increasing the complexity of the source and/or detector model, and then the recalculations were in better agreement. But some changes were also made in the setup of MCNP, or to the code to improve performance and/or speed. When each experiment was at it's best agreement, then the next experiment was started. What was learned in earlier experiments was successively applied to later experiments. However, we generally did not go back and try to refine the earlier experiments.

MCNP CONSIDERATIONS

The MCNP input files for all of the above geometries were created by following the standard procedure described in detail in reference (2). The input files consist of four parts: a geometry part, a source definition part, a material part and a tally part.

Only the photon mode was used. Electron mode was not used because it takes at least 40 times longer than that with photon mode alone. Although

electron transport was not simulated, the electron-induced photons were not neglected. A thick-target bremsstrahlung model (TTB) was used instead. The TTB model also generates electrons, but assumes that they travel in the direction of the incident photon and that they are immediately converted. The bremsstrahlung photons produced by these electrons are then banked for later transport.

The detailed physics treatment (default MCNP choice) for photon interactions was chosen for all geometries in this report.

There are many tally types in MCNP. For our application, only tally 8 (pulse height tally) was necessary. For this tally, the only possible variance reduction scheme is biasing of the source itself.

The direction of a gamma photon at birth is determined by an azimuthal angle (0-2pi) and a polar angle (0-pi). In reality, photons are emitted isotropically over 4pi solid angle. When a source is far away from the detector, most of the photons will never reach the detector due to the low geometrical efficiency. Since there is no variance reduction scheme to shorten or terminate the life of those photons for this tally type, there is much wasted calculation time. However, if one is only interested in the total efficiency of a detector, and if in the energy range of interest the elastic (Thomson, or coherent) scattering is not important, then directional biasing schemes can be used. In this method, only a small range of the azimuthal and polar angle is used. Instead of the full 4pi solid angle, particles are forced to be emitted from the source in a direction toward the detector, and with all photons in a cone which entirely encloses the detector volume. Since only a small fraction of the particles are modeled (in most cases less than 10 %), this method can greatly reduce the computation time. The results are then corrected by the ratio of the solid angle subtended with the biasing cone to the 4 total solid angle. This will then yield the same efficiency response as if no biasing was used.

When elastic scattering becomes important (which is more likely at lower energies and in heavily attenuated sources), using highly collimated directional biasing will result in somewhat lower computed "efficiency". If this loss is considered significant, then one must use the full 4-pi emission angle in order to achieve the desired accuracy.

Unless otherwise stated, directional biasing was the default method used for most of the MCNP efficiency calculations in this report. The few cases where this method did not give accurate results are discussed.

The MCNP output for pulse height tally is in counts (normalized by number of histories at that energy) vs energy. For all the efficiency calculations, the channel width was set to 1 keV. The efficiencies are obtained from output files by subtracting the background continuum counts (the average of the counts in the left and right neighboring energy bins) from the peak counts. This was done with a custom software routine.

All of the experimental results were taken and analyzed by standard Canberra instruments and software. Because these tests covered over a year in time, various detectors, MCAs, and data analysis software were used. All MCAs were properly calibrated for energy, and had adequate gain to sufficiently resolve the peaks of interest. Although the spectra were analyzed using various Canberra software packages, all of the analysis algorithms used compute the full energy background-corrected net peak count rate for each of the relevant peaks of the spectra in a similar manner. Therefore, the results are comparable.

In almost all of the MCNP calculations for full energy peak efficiency, the statistical precision of MCNP calculations and experimental measurements was kept less than 3% to ensure the statistical validity of the results.

Three kinds of computers were used for the MCNP calculations: DEC Alpha AXP (DEC chip 21064), VAX 4000 and IBM PS/Value Point (Intel 486DX2-66Hz). The ratio of the speed for these machines is roughly 6:1:1 (AXP:VAX:IBM).

RESULTS AND DISCUSSION

Each of the experiments described in this section has an introductory text description, a simple graphical portrayal of the detector geometry, and a graphical presentation of the energy/efficiency comparison, and a discussion of the results.

Unless otherwise mentioned, the total number of particles are equal for each energy (equal weights are given for each energy). However, more CPU time is spent on the higher energy photons because the lifetime of higher energy photons is usually longer than that of lower energy photons in the simulation.

Test 1: Multi-energy Small Source at Various Angles

The detector used was a Canberra P-type coaxial Ge detector (S/N 3155). The detector had 32% relative efficiency at 1332 keV. The detector physical size is 53.4 mm diameter and 62.0 mm length. Other physical features of the detector (e.g. top and side dead layer thickness', detector holder, detector well and groove, detector holder and all endcap features) are included into the model. The source used in the experiment was a mixed gamma source in a 20 cc cylindrical liquid scintillation counting vial from Analytics, Inc. (S/N 46481-121). It contained nuclides with 10 energies from 88 keV to 1836 keV. Three cases were studied:

- source on axis at a distance 36.5 cm from the detector endcap;
- source 45 degrees at 48.3 cm;
- source 90 degrees at 38.5 cm.

The geometry and graphical presentation of the results are shown in Fig. 1. The agreement between MCNP and experimental measurements is excellent. For the 0 degree angle source position, the largest relative difference is 5% for all energies, and the overall average difference is 3%. This geometry is straightforward to model since the shape of the materials, such as the Al endcap and detector dead layer between the source and the detector, is cylindrical. For 45 degrees, the largest relative difference is 6% for all the energies and the overall average difference is 2%. For 90 degrees, the largest relative difference is just slightly higher at 7%, and the overall average difference becomes 4%. For the off-axis geometries, we found it necessary to add many more features to the detector model (holder, endcap, side dead layer, etc.) in order to achieve the results shown.

Fig. 1

Test 2: Eu-152 Line Source

Although in the previous geometry both the source volume and density were included in the model, the distance effect makes it more or less like a point source. The objective of this geometry is to see how accurate MCNP is when the source is extended linearly.

The detector used was a Canberra P-type coaxial Ge detector (S/N 3202). The detector had 30% relative efficiency at 1332 keV. The detector physical size is 53.0 mm diameter and 59.5 mm length. The source used in the experiment is a Eu-152 line source from North American Scientific,

Inc. (S/N A0980). The activity of the source was uniformly distributed in an epoxy matrix with a density of 1.07 g/cc and cast in 6.53 mm O.D. Tygon tubing. This tubing is then inserted into rigid plastic tubing with an overall length of 80 mm and an outer diameter of 10.3 mm. The source was placed horizontally, 14 cm above the detector Al endcap and supported by a plastic disk. The detector axis passes through the center of the line source. All of these factors were included in the model.

The initial results indicated an average 17% bias between the experimental data and the calculated results. Additional features to the model, and repeated measurements of the line source did not change the bias. Other detectors were used for both MCNP and experimental measurements and gave the same bias. But a summation of point source measurements done to simulate the line source had quite good agreement with the MCNP data. This led us to suspect the calibration source accuracy. This was later confirmed by the manufacturer with an amended calibration certificate.

The test geometry are shown in Fig. 2(a) and the final results are shown in the graph of Fig. 2, middle data set. The agreement is excellent at all energies, with largest relative difference of 3% and overall relative difference of 2%.

Fig. 2

Test 3: Mixed Gamma Plane Source

This experiment is intended to extend the calibration validation from a one dimensional source to two-dimensional planar large area source.

The detector used was a Canberra P-type coaxial Ge detector (S/N 3313). The detector had 25% relative efficiency at 1332 keV. The detector physical size is 52.0 mm diameter and 53.0 mm length.

The source used was manufactured by North American Scientific, Inc. It is a thin (1mm) source with an area of 50 x 50 cm, backed by a support plate for a total thickness of 1 cm. It was placed 8 cm above the Al endcap of the detector. The source is supported by a 10.2 cm I.D. 0.64 cm thick plastic pipe. All of these geometrical factors were included in the model.

The test geometry is shown in Fig. 2(b), with the graphical results in Fig. 2, bottom data set. The largest relative difference is 10%, and the overall difference is 6%. There is a positive bias of approximately 7% in the MCNP efficiency over the source efficiency. It is unclear what is causing this bias. The two likely possibilities are that the source has a non-uniform distribution in the source plane, or that the source activity is different than on the source document, as in the line source case. However, by the time this was identified, the source was no longer available for re-calibration to verify these hypotheses.

Test 4: Marinelli Beaker Source

This common geometry is an additional extrapolation of the extended plane source in experiment 3, but with a moderately thick sample.

The detector used was a Canberra P-type coaxial Ge detector (S/N 3313). The detector had 25% relative efficiency at 1332 keV. The detector physical size is 52.0 mm diameter and 53.0 mm length. Three Marinelli beaker sources made by Analytics, Inc. were used (Cs-137 for 662 keV, Am-241 for 60 keV, and Zn-65 for 511 and 1115 keV). These sources are specifically chosen to eliminate coincident summing effects in the experimental data for this high efficiency geometry. The source volumes are all 1 liter with matrix density of 1.09 g/cc. The source sits right

on top of the detector. No directional biasing was used because this is a relatively high efficiency geometry.

The test geometry is shown in Fig. 2(c) and the results are in the graph of Fig. 2, top data set. The agreement between MCNP and experiment is excellent. All the relative differences are within the statistical uncertainty given by the source and MCNP calculation. The largest relative difference is 4 % and the overall difference is 2 % with an uncertainty of 2.5 %.

Test 5: Heavily Shielded Mixed Gamma Source

The objective of this test is to evaluate the ability of MCNP properly calculate the effect of heavily attenuated sources. This is also important for successful calibrations of very thick sources.

The detector used was a Canberra P-type coaxial Ge detector (S/N 3313). The detector had 25% relative efficiency at 1332 keV. The detector physical size is 52.0 mm diameter and 53.0 mm length. The source was made by North American Scientific, Inc. The activity of the source is contained on a 1 mm diameter resin bead in the center of a 2.5 cm diameter by 0.64 cm thick plastic disc. Attenuators made of steel disks with thickness' of 0.0, 1.0, 3.0, 6.1 and 9.1 cm were placed between the source and the detector. The steel shield disks have a radius of 10 cm. The shield is supported by a 10.2 cm I.D. 0.64 cm thick plastic pipe and a Eu-152 button source was placed right on top of the shield. All of these are included in the model.

The test geometry and the graphical results are shown in Fig. 3. The low energy data points are missing in the 6.1 and 9.1 cm cases because the corresponding experimental and MCNP lines are almost completely shielded. For 0 cm thickness, the agreement between experimental and MCNP results is excellent with the largest difference of 3 %. Even at the maximum thickness where there is 99.9% attenuation, the agreement is still very good (within 6%) and all data are individually statistically acceptable.

Fig. 3

Test 6: Multiple Line Sources in a 55 Gallon Drum in a Canberra Q2 System with Three Detectors

This experiment was designed to demonstrate the capabilities of MCNP to accurately model very complex sources.

The Q2 counter is a standard nuclear waste assay system designed and manufactured by Canberra. It is shown schematically in Fig. 4(a). The typical size of the steel shield is nominally 100 x 100 x 100 cm with a wall thickness of 16 cm. A 200 liter drum which contains the sample to be assayed sits on a rotating table. The standard system is equipped with 3 vertically aligned Ge detectors viewing the sample radially through holes in one of the shield walls. The drum is rotated about the axis multiple times during the measurement.

Fig. 4

The Ge detectors used are Canberra standard coaxial (S/N 3292, 3277 and 3290) with similar sizes (50 mm O.D. and 45 mm length) and similar performance (28 % relative efficiency at 1332 keV).

The calibration geometry used by Canberra is shown in Fig. 4(b). It is intended to simulate a uniformly distributed source, but without the expense of constructing 4 different radioactive calibration drums. Six line sources are placed at 6 different radial positions in a non-radioactive drum. The radial distances are each at the center of 6 concentric hollow cylindrical volume elements, with each volume element containing 1/6 of the drum volume. Four different non-radioactive drums

have been constructed, each with a different density. They have standard 200 liter drum shells, and are filled with foam (0.02 g/cc), cellulose board (0.43 g/cc), particle board (0.75 g/cc) and sand (1.70 g/cc). This is a relatively difficult geometry to model because of the number of components involved. The cross sectional view of the 200 liter drum is shown in the top drawing of Fig. 4(b). The line sources are Eu-152, similar to those used in experiment 3. The source is enclosed in a plastic tubing of 7/8" O.D., 11/16" I.D. and 33" length. The drum rotates with a constant velocity through many rotations while being counted. But, since MCNP is not capable of simulating a rotating source, the 6 rotating line sources in the drum were approximated by 6 uniform cylindrical sources with zero thickness. The plastic tubes and source matrices were modeled by adding the appropriate cylindrical layers outside the drum as shown in the bottom drawing of Fig. 4. Since the six Eu-152 line sources have similar activity (about 3 % difference). The gammas have equal probability to be emitted from the cylindrical source surfaces.

Because of the low geometric efficiency, and the high probability for photon absorption, the calculation time would be extensive, even when the standard directional biasing is used. The MCNP general purpose source definition input card only allows the user to bias all of the photons in a single direction and within a single conical angle. But, since the geometry is quite variable depending upon the starting location of the photon in the source, it was necessary to prepare a special subroutine to more efficiently do this task. In this subroutine, each photon is focused toward the detector in it's own biasing cone. The size of the biasing cone is variable, and depends upon the location of the origin of the photon. The cone completely encompasses the detector. The biasing fraction is for each photon is cumulated during the run. After the computations are complete, the biasing is removed with this cumulative factor.

The results are graphically displayed in Fig. 4. For densities 0.02, 0.43 and 0.76 g/cc, except the energy 122 keV, the agreement between MCNP and experiment is quite good. Except for the 122 keV data points, all others are within 1 standard deviation and less than 6% bias. For the 1.7 g/cc density, MCNP shows a consistent bias of approximately 10% (except 122 keV). It is suspected that this bias may be caused by one or a combination of errors in the source measurement. Likely candidates are errors in the true drum size (there was a slight difference in size among the drums which we did not include in the model), the geometrical placement of the drum during the calibration, and/or imprecise knowledge of the true physical location of the outer line source in the drum.

CONCLUSIONS

A series of benchmark tests have been conducted to validate use of MCNP for efficiency calibration of Ge detectors for simple and complex geometries. The analyses result in the following conclusions.

For accurate efficiency calculations, especially at low energies, it is critical that the source and detector model be complete. All source and detector dimensions must be known and entered into the model. The density and elemental composition of the source, the detector, and any intervening absorbers must be accurately known. For example, we have found it necessary to use approximately 20 parameters in the model just to define the detector, mounting structure, and endcap.

For Ge detector peak efficiency, and for energies above 200 keV, MCNP is capable of achieving better than 10 % accuracy (all tests with the

exception of Q2 at 1.7 g/cc density) in a reasonable amount of computer time (less than 24 hours for an AXP), with the use of directional biasing. Abandoning directional biasing can keep the accuracy within 10% down to 100 keV, but at the expense of 10-100 times longer computer time. It is easy to make mistakes in using MCNP. The program is not user friendly. Many physical parameters of the source and the detector must be entered to adequately describe the source-detector geometry. While there are some complementary software programs to provide some degree of error checking of the model, nothing can replace careful multiple checking of all input parameters, and reality checks of the results. First to verify the detector model, we compute a point source efficiency for a source on the detector axis, and at 90 degrees. This is then verified with a multi-energy source measurement. We then independently do something to check the source geometry, (e.g. reduce it to a point source or comparison the results to a previous good calibration).

MCNP, when used properly, is likely to be more accurate for efficiency calibrations of large and complex sources than calibration standards created for such geometries. For small well defined geometries at unit density, it is easy to obtain 3% calibration sources. However for small sources of other than unit densities, for sources with large volumes (tens to thousands of liters), or for complex shapes (long rods, large flat plates, cylindrical shells, etc.), the additional errors in source distribution and fabrication and calibrations are likely to be larger than the 10% MCNP accuracy demonstrated here. And, when a calibration using MCNP run is completed, there is no radioactive disposal cost. It is our plan to make MCNP the calibration method of choice for the large geometries in our gamma spectroscopy Waste Assay product line.

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PHASE I RESULTS OF THE WASTE INSPECTION TOMOGRAPHY SYSTEM*

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ABSTRACT

Waste Inspection Tomography (WIT) consists of a self-sufficient mobile semi-trailer for Non-Destructive Evaluation and Non-Destructive Assay (NDE/NDA) characterization of nuclear waste drums using X-ray and gamma-ray tomographic techniques. WIT is a Program Research and Development Announcement (PRDA) contract funded by the Environmental Management's (EM) Office of Technology (OTD/EM-50) for the United States Department of

Energy (DOE). Bio-Imaging Research, Inc. (BIR), of Lincolnshire, Illinois has completed Phase I involving design, fabrication, factory testing, evaluation and demonstration of WIT. The recently completed 23-month WIT Phase I included the design, fabrication, and initial testing of all WIT subsystems installed on-board the trailer. Initial test results include 2 MeV Digital Radiography (DR), Computed Tomography (CT), Anger camera imaging, Single Photon Emission Computed Tomography (SPECT), Gamma-Ray Spectroscopy, and Active and Passive Computed Tomography (A&PCT) using a 1.4 mCi source of 166Ho. These techniques were initially demonstrated on a 55-gallon phantom drum with three simulated waste matrices of combustibles, heterogeneous metals, and cement using check sources of gamma active isotopes such as 137Cs and 133Ba with activities between 9 Ci and 250 Ci. Waste matrix identification, isotopic identification, and attenuation-corrected gamma activity determination were all demonstrated nondestructively and noninvasively in Phase I.

BACKGROUND INFORMATION

The United States Department of Energy has in excess of 1,000,000 nuclear waste drums currently stored at nearly 50 sites within the United States that need to be characterized over the next few years. The contents of these drums must be characterized as either high-level waste (HLW), low-level waste (LLW) or transuranic waste (TRU), before the drums are assigned to one of three permanent storage locations. Strict permitting regulations also require information to be gathered about the condition and contents of the waste containers.

X-ray imaging is an established method for waste container inspection. The technique generally used is real-time radiography (RTR) using a 420 kV radiation source, in which a TV camera is coupled to a two-dimensional, light-producing X-ray detector, such as an image intensifier or a scintillation screen. The camera output provides a TV image that is viewed on a monitor during X-ray exposure which, as an example, can see the motion of a moving liquid surface.

RTR systems have several disadvantages however. Area X-ray detectors typically suffer from blooming artifacts. Blooming is caused when a saturated signal spills over into neighboring sensor elements resulting in excessive brightness and limited spatial resolution. RTR systems have limited contrast discrimination with a true dynamic range of usually less than 14-bits (16,384 gray levels in the image), meaning that contrast in a single exposure is limited. An image intensifier is also limited to a small area of the drum.

RTR limits geometric depth perception because of super-positioning, and it lacks quantitative information including two- and three-dimensional spatial and density measurements because the data is not in digital form. The combination of 420 kV source and a restricted detector dynamic range limits RTR penetrating and discriminating capability for inspecting the denser waste containers including cement-solidified drums, glass, and sludges, which make up nearly half of DOE's inventory of nuclear waste drums.

Emerging technologies, like WIT, designed for nondestructive evaluation (NDE) of low level, transuranic, and mixed nuclear waste, include high-energy 2 MeV X-ray computed tomography (CT) and digital radiography (DR), with 18-bit dynamic range. Figure 1 shows a 2 MeV transmission with a full-drum, DR projection image of a 55-gallon drum phantom. Clothing in the top layer simulates low-density combustible waste. Cans of liquid, metal rods and plastic in the mid section simulate higher density waste.

Cement in the bottom layer simulates solidified high-density waste. A 133Ba isotope bottle is visible in the upper left plexiglass tube.
Fig. 1
In its conventional approach, WIT CT/DR imaging uses a curved linear array of solid-state X-ray detectors. The array is composed of individual, closely aligned detection channels. The channels are separated by thin septa that minimize crosstalk and blooming, while offering superior spatial and contrast resolution with high image quality, compared to real-time approaches. These detectors have enough dynamic range to provide contrast sensitivity of 18-bits (up to 262,144 gray levels). The greater the dynamic range and penetrating radiation, the greater the advantage in examining denser waste forms mentioned above. Thus, the WIT approach can image nearly all dense DOE waste streams with even faster throughput for the lighter waste forms. This cannot be said for the more commonly used RTR X-ray imaging systems.
For WIT CT, X-ray projection data is collected from a thin plane of the object, using a linear detector array on the arc of a curve while the object rotates within a thin X-ray fan beam with spiral-like motion capability. This technique is called third-generation CT. The data are mathematically combined to form a cross-sectional image of the thin, irradiated plane or slice. The image in Fig. 2 is a 1024 x 1024 reconstruction of a 10 mm thick slice through the upper section of the phantom drum.
Fig. 2
Slices can be stacked to form a volume rendering of drum content such as the one shown in Fig. 3. In WIT DR, the drum is moved vertically in front of the linear detector array while projection data is collected one line at a time. These techniques measure the X-ray attenuation of the waste matrix and drum. The lines are then displayed as a two-dimensional, freeze-frame projection image (like a baggage inspection X-ray) for DR.
Fig. 3
Active CT on WIT is similar to the conventional X-ray CT techniques. The difference is that a radioisotopic source and single-channel high-purity germanium detector are used with a first-generation CT approach. Active CT data result in the absolute determination of the attenuation of the drum and its contents.
Two emission imaging techniques are employed on WIT for characterizing materials in waste containers. The first of these is gamma emission tomography, commonly called single-photon emission computed tomography (SPECT). Rather than measuring gamma-rays from an external radiation source, SPECT measures the gamma-ray emission inherent in the radioactive waste emitting from within the drum. In this case, emission from actual nuclear waste within a container can provide three-dimensional volume or slice data of the radioactive sources within the container. SPECT uses large area sodium iodide crystals with a two-dimensional array of photomultiplier tube (PMT) detectors for rapid localization of gamma-ray emissions in two-dimensional space and in 3-D with SPECT. These area cameras are called Anger cameras.
The second WIT emission technique uses an energy sensitive single-channel high-purity germanium detector for gamma-ray nuclear spectroscopy. This technique, for nondestructive assay (NDA), can directly identify the emitting isotopic species and the external radioactivity.
The mobile feature of WIT allows inspection technologies to be brought to the nuclear waste drum storage site without the need to relocate drums

for safe, rapid, and cost-effective characterization of regulated nuclear waste. See Fig. 4. The combination of these WIT characterization modalities provides the inspector with an unprecedented ability to non-invasively characterize the regulated contents of waste drums as large as 416 liters (110 gallons), weighing up to 726 kg (1,600 lbs). Any objects that fit within these size and weight restrictions can also be inspected on WIT, such as smaller waste bags and drums that are 19 and 132 liters (5 and 35 gallons).

Fig. 4

Bio-Imaging Research, Inc. (BIR), from Lincolnshire, Illinois has completed Phase I over 23 months involving the design, fabrication, factory testing, and evaluation of WIT. BIR has designed the trailer and multiple inspection techniques including DR, CT, SPECT, and area gamma-ray imaging. BIR has also developed the WIT operational software, the computer hardware, and the gantry mechanical systems. Lawrence Livermore National Laboratory (LLNL), as a subcontractor to BIR under a Work-for-Others agreement with BIR, has developed the A&PCT scanning technique and is participating in WIT evaluation. Early BIR efforts prior to WIT involved investigating the feasibility of using CT to characterize nuclear waste between 1990 and 1993 under Small Business Innovative Research (SBIR) grants from DOE.

DETECTOR DESCRIPTION

The three WIT detection technologies include a linear array of solid-state high-energy X-ray detectors, two area gamma-cameras, and a single high-purity germanium detector. The throughput of WIT inspection is dependent on the physics of the drum being inspected. As an example, the lighter the drum weight, the faster the WIT NDE inspection, and the higher the drum radioactivity, the faster the NDA inspection. The reverse is also true where heavier drums with little radioactivity will require longer inspection times. Ideally, WIT is designed to inspect four drums per hour, per technique. Throughput extremes could yield inspection times of between one and seven drums per hour for each technology employed.

The WIT linear detector array is curved and consists of 896 individual channels of cadmium tungstate (CdW4) crystals mounted on individual photodiodes with thin septa between channels to eliminate crosstalk, blooming, and in-plane scatter. These detectors have an 18-bit (262,144 gray levels) dynamic range for analog-to-digital conversion. The wide dynamic range is necessary to image the variety of material densities and geometries found in DOE waste streams, including combustibles, glass, cement, sludges, and metals that may be present in the same drum. The linear array utilizes a 2 MeV high-energy accelerator as an externally transmitted radiation source using a thin fan beam output with a measured flux of 70 rads per minute at one meter. This high-energy source is needed to penetrate the denser and thicker DOE waste forms like glass logs, sludge, and cemented drums, while allowing for an optimum inspection throughput. The use of energies above 2 MeV is not practical for WIT because of WIT's mobile requirements resulting in weight restrictions for radiation shielding limiting close operator interaction.

The WIT linear detector array and 2 MeV source provide for single-pass digital radiography (DR), which yields a freeze frame projection X-ray image of an entire drum (e.g., like a chest X-ray) using an imaging technique similar to X-ray baggage inspection systems with the drum elevating through a stationary horizontal fan beam of X-rays. A single DR drum view at 2 MeV with wide dynamic range can be acquired in less than

60 seconds with only one X-ray technique. Unlike RTR, both high and low density objects can be examined with WIT DR using a single DR image due to the wide dynamic range.

The linear array and 2 MeV source also provide for transmission computed tomography (CT or TCT) with cross-sectional (two-dimensional) slice and volume (three-dimensional) imaging of drum content based on density distribution. WIT TCT slice thickness ranges from 2 mm to 10 mm through the drum. CT images are acquired by simple drum rotation for data collection and drum elevation for slice location. CT is used for waste drum content identification with a spatial resolution of nearly 2 mm and a density sensitivity of nearly 1%. Typical CT reconstructions and DR images have formats of 256 by 256, 512 by 512, or 1024 x 1024 pixels. Individual CT slice scan times for data collection can be as short as eight seconds for a slice or eight seconds for two slices with an optional dual array for imaging lightweight (S.G.>1) combustible waste. WIT has slower CT scan times (as long as 20 minutes per CT slice) for dense solid glass logs (S.G.2.7). As a bench mark, 100 slices for a low-density combustible waste-filled 55-gallon drum could be acquired and volume rendered in less than ten minutes (with a dual array), whereas a glass log with a density of equal dimensions could take nearly eight hours for 100 slices. A cemented drum (S.G. 2.1) could require a scan time of less than 0.5 hours. Drum wall thickness resolution using WIT TCT is between 0.25 and 0.5 mm. Volume measurements (i.e., for free liquids) of pixels with similar density indications has nearly cubic centimeter resolution with WIT TCT.

WIT's two large area (14" x 17") detectors are each single crystals of sodium iodide in what are typically called Anger cameras. These detectors provide for rapid gamma emission drum area imaging and single photon emission computed tomography (SPECT or ECT) for slice and three-dimensional volume localization of gamma-ray emissions from a drum. Both crystals have a combined total of 110 photomultiplier tubes (PMT) for two dimensional gamma ray emission localization within the waste drum with a spatial resolution of nearly 25 mm over a cross-sectional area of the drum. An emission slice through the drum may also have a thickness of nearly 25 mm. Typical gamma emission projection images and SPECT slice reconstructions have image formats of 32 x 32, 64 by 64, and 128 by 128 pixels. Typical inspection times for a single drum can range from seven minutes to one hour, depending on the emitted gamma ray activity. The higher the activity, the faster the scan time and the reverse is also true.

The single-channel, high-efficiency, (>100% of that of sodium iodide), high purity, germanium detector (HPGe) uses a 50 mm on-a-side square collimator and an active source of 1.4 mCi of Holmium (166Ho). The HPGe detector and 166Ho provide for active and passive computed tomography (A&PCT) with excellent energy sensitivity (of less than 2 kV) for nuclear spectroscopy.

A&PCT on WIT are each first-generation CT techniques, each using a single channel energy sensitive HPGe detector. WIT can detect energies between 10 kV and 1.33 MeV using an 8,000 discrete channel multi-channel analyzer (MCA). This detector and source are used to determine an absolute linear attenuation coefficient of the waste drum and matrix. Passive CT identifies and localizes the radioactivity. The active CT data are used to correct the passive CT data for attenuation caused by the waste matrix and drum itself. The combination of both techniques results in a more

accurate nondestructive assay of the waste drum. Lawrence Livermore National Laboratories (LLNL) has developed the A&PCT techniques. This single detector has a spatial resolution of nearly 50 millimeters over the area with slice thicknesses of nearly 50 mm. Typical A&PCT reconstructions have formats of 14 by 14, 28 by 28, and 42 by 42 pixels. Determination of internal radioactivity approximations for waste drums have thus far shown errors between 5 and 10%. Scan time for a single slice can range from one minute to one hour, depending on the level of radioactivity.

RESULTS

The images included in this paper were taken on-board the trailer and demonstrate CT, SPECT, and A&PCT capabilities. Additional analysis such as drum wall thickness, can be determined to help quantify the relative condition of nuclear waste containers and density measurements of drum contents can be analyzed to further identify and quantify drum contents. WIT can quantify the internal radioactivity which has been corrected for attenuation caused by the waste matrix solely with external measurements without opening the drum. The WIT system's combination of inspection technologies lets inspectors choose an appropriate level of characterization for each site or waste container.

FUTURE WORK

WIT is a Program Research and Development Award (PRDA) contract number DE-AC21-93MC30173. The WIT PRDA is funded by Environmental Management's (EM) Office of Technology Development (OTD/EM-50) for the United States Department of Energy (DOE). The program is managed by the U. S. government from the DOE Morgantown Energy Technology Center (METC) in Morgantown, West Virginia.

Phase II, began in December, 1995, and consists of a twelve-month program for the integration and DOE site demonstration of WIT. Phase II, site demonstrations of WIT have commenced at Lawrence Livermore National Laboratories in Livermore, California, and are planned at Westinghouse Savannah River Company (WSRC) in Aiken, South Carolina, and at the Idaho National Engineering Laboratories (INEL) in Idaho Falls, Idaho. These demonstrations will involve characterizing real waste to validate the system's ability to identify regulated contents and to verify system throughput.

BIR plans to commercialize WIT and plans offer drum scanning/characterization services to DOE and other sites requiring mobile capabilities.

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CHARACTERIZATION OF LOW-LEVEL WASTE STREAMS AND SUSPECT WASTE FROM THE LARGEST LOS ALAMOS NATIONAL LABORATORY GENERATORS

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ABSTRACT

A detailed waste stream characterization of four primary generators of low-level waste at Los Alamos National Laboratory (LANL) was performed to aid in waste minimization efforts. Data was compiled for these four generators from 1988 to the present. Previous waste minimization efforts have focused on identifying waste stream processes and performing source materials substitutions or reductions, where applicable. In this

historical survey, the surveyed generators included an accelerator facility, the plutonium facility, a chemistry and metallurgy research facility, and a radiochemistry research facility. Of particular interest in waste minimization efforts was the composition of suspect low-level waste in which no radioactivity is detected through initial surveys. Ultimately, this waste is disposed of in the LANL low-level permitted waste disposal pits, which fills a scarce and expensive resource with sanitary waste.

Detailed analyses of the waste streams from these four facilities have revealed that suspect low-level waste comprises approximately 50 percent of the low-level waste by volume (47 percent by weight). However, there are significant differences in suspect waste density when the type of radioactive contamination is considered. For the two facilities that deal primarily with beta-emitting activation and spallation products (i.e., the radiochemistry and accelerator facilities), the suspect waste has a much lower density than the total low-level waste coming from those facilities. For the two facilities that perform research on transuranics (i.e., the chemistry and metallurgy research and plutonium facilities), suspect waste is higher in density than the total low-level waste from those facilities. It is theorized that the low density suspect waste is composed primarily of compactable laboratory trash, most of which is not contaminated but can be easily surveyed. The high density waste is theorized to be contaminated with alpha-emitting radionuclides, and, in this case, the suspect waste demonstrates fundamental limits in detection.

INTRODUCTION

Los Alamos National Laboratory performs a wide spectrum of research on radioactive and non-radioactive materials in support of the United States Department of Energy (DOE). In the course of performing such work, solid low-level radioactive waste is generated and ultimately disposed of in the low-level waste disposal pits at LANL Technical Area (TA)-54, Area G. Due to the limited disposal volume (approximately 59,000 m3 of pit volume remained as of July, 1995) and the large cost associated with low-level radioactive waste disposal, the Laboratory has a concerted waste minimization effort aimed at reducing laboratory-wide waste volume. Current building upgrades at the Chemistry and Metallurgy Research (CMR) facility will result in a dramatic increase of low-level waste generation. The Laboratory is responsible for examining all waste streams and identifying opportunities for waste reduction where technically and economically feasible. The following study was performed to identify waste from four facilities (i.e., CMR, the Plutonium Facility, TA-48 and TA-53) and quantify the amount of suspect radioactive waste from those facilities. Suspect radioactive waste is waste that originates in a radioactive materials management area (RMMA) and, therefore has the potential for radioactive contamination. However, radioassay and survey measurements on the material detected no radioactive contamination. Cursory analyses have indicated that approximately 50 percent of the waste disposed at TA-54 consists of suspect radioactive waste. By segregating actual radioactive waste from waste that is known through acceptable knowledge to be non-radioactive, Laboratory waste management can develop an approach that will result in non-contaminated waste being sent to the sanitary landfill for disposal and minimize the amount of suspect radioactive waste being disposed of at TA-54. Gross alpha/beta surveys and gamma detection systems would be used to verify the non-

contaminated status of the waste to levels acceptable to the Department of Energy.

FACILITY DESCRIPTIONS

Chemistry and Metallurgy Research Facility

The Chemistry and Metallurgy Research (CMR) facility has served as the primary special nuclear materials analytical laboratory for LANL since 1952. The CMR facility is located in TA-3. The three-story structure contains approximately 550,000 ft2 of floor space and is presently designated as a security "category 1" facility with a safety classification of "moderate hazard." The building consists of eight wings: the Administration Wing and Wings 1 through 5, 7, and 9. The Administration Wing contains office and conference room space occupied by the Chemical Science and Technology Division Office (CST-DO).

Operations

Wings 1 through 5, 7, and 9 house experimental facilities. The experimental facilities are occupied by personnel from several line organizations at LANL. Approximately 48,0000 ft2 of the first floor space is devoted to laboratories, and approximately the same amount of space is devoted to offices. The basement and attic spaces provide utility services to the first floor laboratories and offices. Some basement areas have been modified for use as laboratories. Each wing is designed to operate independently with its own electrical power distribution and ventilation systems. Wing 9, which contains the hot cell facilities for the CMR, was added in 1960 (1).

The Plutonium Facility

The LANL Plutonium Processing Facility is located at TA-55 and occupies a 30-acre location with 70 structures. Only the Plutonium Building (PF-4) and the Health Physics Assay Laboratory (HPAL) contain nuclear materials. Each area is designated as an RMMA. PF-4 is a two-story, 151,000 ft2 building, with support equipment in the basement and laboratory rooms divided among four wings on the main floor. The major laboratory rooms contain over 300 gloveboxes for the handling of plutonium, uranium and other nuclear materials.

Operations

The Residue Processing Group (NMT-2) develops and proves processing technology for plutonium and other actinides through aqueous and molten-salt based technologies. This group supports the LANL Plutonium Facility by recovering and purifying plutonium scrap residues and converting them to an oxide or metal that can be used or placed in long-term storage.

The Advanced Technologies Group (NMT-6) conducts fundamental and applied research in actinide chemistry. This group focuses on new and emerging separation technologies and improving existing technologies.

The Waste Management and Environmental Compliance Group (NMT-7) handles the hazardous and radioactive waste materials generated at TA-55. NMT-7 also assures that the facility complies with all environmental requirements, including those regulating waste, water, and air discharges (2).

Technical Area 48

TA-48 is used for the study of nuclear properties of radioactive materials by using analytical and physical chemistry. Measurements of radioactive substances are made and hot cells are used for remote handling of radioactive materials.

Operations

The hot cell facility includes a number of laboratories and rooms within building RC-1 at TA-48. This area is used to process targets that are irradiated in the proton beam at the Los Alamos Meson Physics Facility (LAMPF) for production of medical radioisotopes. There are large amounts of radioactive materials present in the hot cells within the facility. Virtually all materials present in the facility are beta/gamma emitters; by policy, large amounts of alpha emitters are not handled by the facility (3).

The alpha cell area is the primary area at TA-48 used for plutonium and uranium research. This area is segregated from the hot cell. Waste from this area accounts for a very small percentage of the total waste emanating from TA-48.

Other laboratories at TA-48 consist of radiochemistry laboratories used for the characterization of very low levels of radioactive contamination. Waste generated from the processing of research environmental samples has not included suspect radioactive waste produced over the past 7 years.

Technical Area 53

Located within TA-53 is the Los Alamos Meson Physics Facility (LAMPF). LAMPF is a high energy particle accelerator facility that produces protons, neutrons, and subatomic particles for use in basic research, isotope production, radiochemistry, solid-state physics research, and accelerator technology.

Operations

The accelerator and beam tube provide a primary beam of 800 MeV protons at an average current of one milliampere for basic nuclear and elementary particle physics research. Positive and negative hydrogen ions are generated and injected into a high voltage dome. A low density plasma is produced and protons are extracted and accelerated through a potential field. The beams are steered and focused into the linear accelerator drift tube, into the side-coupled cavity linear accelerator stage, and switched by bending and quadrapole magnets to various experimental areas.

Experimental Area A conducts experiments for particle physics and nuclear structure studies. Pions, muons, and protons are produced and used in a variety of experiments. Using the highest intensity main proton beam from the accelerator, the spectrometers in Experimental Area A measure particle scattering from interactions of the proton beam with targets located along the beam line to determine number, type, direction, momentum, velocity, and electrical charge.

Experimental Area B conducts experiments in the Neutron Physics Laboratory using a medium resolution spectrometer. A negative ion beam is divided into two portions. One portion generates a polarized neutron beam by charge interchange in a deuterium target which is delivered to experiments in the neutron area. The other portion of the negative ion beam is delivered to the External Proton Beam line where it is accessible for experimental setups.

Experimental Area C houses the High Resolution Proton Spectrometer (HRS). It measures the scattering of protons from nuclei and the production of other particles resulting from the incident protons. The HRS can also measure the spin of scattered particles (4).METHODS

Historical low-level waste data from the four facilities was used to determine the composition and nature of all low-level waste over the past seven years. Waste contents, activity, weight and volume were used to identify facility specific trends for actual and suspect low-level waste.

To perform this analysis, the Low-Level Waste Database was used extensively.

Low-level Waste Database

LANL has an effective method of tracking all generated waste (e.g., transuranic, mixed transuranic, low-level, mixed low-level, chemical). Generators must complete a Chemical Waste Disposal Request (CWDR) form for any waste to be disposed, treated, or stored. This form is forwarded to the Waste Management Program, and the form is verified for accuracy and completeness, and subsequently entered into the Waste Management waste tracking databases. These databases track the final disposition of each item of waste received. Volumes, weights, radionuclides, activities, locations of waste generation and generator information submitted on the CWDR is entered into the databases. This provides for a centralized location with comprehensive data accessible to auditors, waste management, and waste minimization programs.

Volume Data

The total volume of solid low-level waste disposed of at TA-54, Area G in 1994 was 2963 m3 (Fig. 1). The four facilities of interest in this study accounted for slightly more than 28 percent of the total volume. The environmental restoration projects at the Laboratory are the single largest generator of low-level waste; however, the waste and waste type are highly dependent on the site and contamination of the site. For the purposes of this study, the four chosen facilities represent a cross-section of laboratory-wide activities.

Fig. 1

Database Analysis

Information from all low-level waste disposal requests from the low-level waste database for the CMR facility over the past 7 years were analyzed for each of the facilities. Since 1994, "suspect radioactive" waste is no longer allowed for disposals. As a consequence, waste originating from an RMMA that has no detectable activity is typically described as having isotopic activities of 1 nCi. Therefore, any waste package with radionuclide activity averaging less than 2 nCi is considered as suspect in the 1994-1995 timeframe for this study.

Since 1988, the four facilities have disposed of 6,639 low-level waste packages with 1,829 of the packages containing suspect radioactive waste. The total volume disposed over the same time period is 14,762 m3 with suspect waste accounting for 7,726 m3.

FINDINGS

Findings for each of the four facilities are summarized below.

CMR Waste Profile

Low-level waste arising from standard operations at the CMR facility consist primarily of contaminated and potentially contaminated laboratory equipment, personal protective equipment (PPE), and laboratory waste products. Currently, the CMR building is undergoing extensive upgrades. Due to these upgrades, CMR low-level waste now includes construction and building debris (e.g., electrical conduit, water pipes, ventilation ductwork) originating in radioactive materials management areas (RMMAs). Typically, this waste is packaged in Scientific Ecology Group (SEG) B-25 boxes (2.66 m3).

Radionuclides present in CMR waste can include virtually any radionuclide found at LANL. However, most waste items contain either fission products (primarily 137Cs and 125Sb) or plutonium-contaminated materials and their progeny (238Pu, 239Pu, 240Pu, 241Pu, 241Am, 237Np, 233Pa and 237U). Most

waste items contain either fission products or plutonium, often both. The form of contamination is dependent on the origination point of the waste.

Plutonium Facility Waste Profile

Virtually all items disposed of as low-level radioactive waste from TA-55 over the past 7 years have been described as contaminated with only plutonium and plutonium progeny (primarily 241Am). The only exceptions have been for disposals of spent radioactive calibration sources. Most waste items are described as either compactable-boxed room trash (e.g., small lab items, PPE, paper) or building debris (e.g., wood, plastic, metal, paper, rubber, glass, waste rags, absorbed liquids, equipment, concrete, general building debris).

TA-48 Waste Profile

Two primary waste forms dominate the waste generated by TA-48. Waste originating from the hot cell areas accounts for more than 80 percent of the waste. This waste is typically compactable laboratory trash contaminated with radionuclides identical to those seen in waste from TA-53. The radioactivity is easily detected, and, contaminated waste usually contains large amounts of activity.

The second waste form comes from plutonium-based work in the alpha cell area. Recently, 67 2-ft3 boxes from this area were assayed. Of the 67 boxes, 27 boxes were found to have measurable amounts of transuranic radionuclides (i.e., 241Am, 238Pu and 239Pu) and nearly all had measurable amounts of 137Cs, 85Sr and 95mTc. In only one box was there contamination from only transuranic radionuclides.

TA-53 Waste Profile

The majority of radioactive waste generated by TA-53 consists of activation and spallation products. These radionuclides are produced when high energy particles collide with matter. Nuclear capture and scattering mechanisms subsequently result in the production of radioactive material. The radionuclides produced are dependent on the scattering and capture cross-sections of the target material. In general, activation of iron accounts for the majority of the activity contaminating waste at TA-53. Past radioassay activities conducted on waste packages from LAMPF and the Los Alamos Neutron Scattering Center (LANSCE) have shown that the common radionuclides contained in the waste packages are strong gamma emitters that are readily detectable by commercially available detection systems. The primary radionuclides include 7Be, 56Co, 60Co, 54Mn, hafnium, lutetium, and europium isotopes.

DISCUSSION

CMR Facility Data

The data for CMR shows that approximately 49 percent of all waste packages from the CMR facility have been suspect low-level. A total volume of 5,978 m3 of low-level waste has been disposed since 1988, with 3,704 m3 being suspect low-level waste. As a point of reference, Pit 38 at TA-54, Area G (the largest low-level radioactive waste disposal pit at LANL) has a total capacity of approximately 37,000m3 below the spill line (5). The amount of suspect waste from the CMR facility over the past 7 years amounts to approximately 10 percent of the volume of the largest disposal pit at LANL.

Every CMR item considered to be suspect over the past 7 years, has, in theory, been contaminated with transuranic radionuclides. Virtually every suspect item has been described in the database as building debris. These items are among the hardest to survey and swipe for contamination;

therefore, it is not surprising that little or no actual values are assigned to these items.

The Plutonium Facility Data

Since 1988, the volume of Plutonium Facility low-level waste disposed at TA-54, Area G is approximately 3,570m3 with 1,910 m3 containing suspect radioactive materials.

Figure 2 shows data for CMR and the Plutonium Facility waste. From the data, it appears that suspect package density is significantly higher than the average density of all waste from these facilities. Because the majority of suspect waste from these facilities tends to be building debris, the packages tend to be larger and more dense than laboratory trash packages. Building debris surfaces are typically very irregular with much of the contaminated surface inaccessible for swipes, smears and direct survey (i.e., ductwork, plumbing and electrical conduit). These waste items tend to be disposed of in SEG B-25 waste boxes rather than 55 gallon drums. The resulting package cannot be counted on the facilities' drum counters. Furthermore, the detection capabilities for B-25 waste contaminated with plutonium is poor. Therefore, if the waste is not grossly contaminated with large amounts of plutonium, the waste will likely be listed as suspect waste. Figure 3 shows the detection limits for the waste management High Purity Germanium Detector and a 2,000 lb B-25 box.

Fig. 2

Fig. 3

Figure 2 shows the density of all waste from these facilities has increased steadily over the past 4 years. This is because of waste minimization efforts to reduce the volume of waste from these facilities. Efforts have been made to pack waste more efficiently with less void space.

TA-48 Data

A total of 1,017 packages accounting for 2,001 m3 of TA-48 low-level waste, have been disposed of at TA-54, Area G since 1988 with 423 packages (1,174 m3) being classified as suspect radioactive waste. The majority of suspect waste packages are described as laboratory trash and laboratory equipment with activation and fission product suspect contamination.

TA-53 Data

Data for TA-53 is strikingly similar to that of TA-48. A total of 654 packages, totaling 3,213 m3 of low-level waste, have been disposed of at TA-54, Area G. Suspect radioactive low-level waste accounts for 198 packages with a total volume of 939 m3 since 1988.

Unlike the data for CMR and the Plutonium Facility, data for TA-48 and TA-53 show that suspect waste tends to be lower in density than the total low-level waste. The waste is typically laboratory trash and, in theory, is contaminated with gamma-emitting activation products.

Because of the nature of the activation and spallation processes, radioactive contamination at TA-48 and TA-53 is expected to consist of fixed contamination of target and beam-line materials. It is very likely that cellulosic materials used in the handling of target and beam-line activities are not contaminated.

CONCLUSIONS AND RECOMMENDATIONS

CMR and the Plutonium Facility

Both the CMR and Plutonium facilities are among the Laboratory leaders in waste characterization capabilities. Because of the nature of the suspect

contaminated waste, it is not likely that current state-of-the-are detection equipment will meet any reasonable criteria for free release. Furthermore, because of the widespread contamination in these aging facilities, most building materials tend to exhibit low levels of fixed radioactive contamination. It is concluded that the majority of suspect waste from these facilities are actually contaminated with radioactive materials, but due to detection limits for transuranic radionuclides, relatively small amounts of contamination pass through characterization systems undetected. Waste reduction efforts for these facilities should concentrate on source reduction/substitution, decontamination and material compaction rather than detection and free release.

TA-48 and TA-55

Because the majority of suspect low-level radioactive materials are low density cellulosics (Fig. 4), a materials segregation and characterization program may be used to verify that cellulosic materials are cleared for free release. Currently, waste box counters are capable of minimum detectable concentrations of less than 1 pCi/g (based on 73As and a 200 s count time). If even one-half of the suspect contaminated cellulosics can be free released as non-radioactive, the saved volume would be approximately 150 m3 annually from these two facilities alone. This method of characterization and free release would require guidance and approval from DOE for volume-contaminated materials (i.e., similar to methods used in the nuclear power industry).

Fig. 4

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RADIOLOGICAL CONTAMINATION PENETRATION DEPTH IN FERNALD TRANSITE PANELS

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ABSTRACT

To characterize the penetration depth of radiological contamination through the thickness of transite (an asbestos-cement building material) from the Department of Energy Fernald site, both destructive and non-

destructive analysis techniques were used. The destructive techniques were based on progressively removing layers of material and subsequent direct analysis of the exposed surfaces. These layer removal analyses included quantitative measurements using a Geiger-Mueller (G-M) detector and qualitative measurements based on autoradiography and ultraviolet photography. G-M detector measurements during layer removal provided quantitative distributions consistent with diffusion theory, serving to validate a novel non-destructive technique. The ultraviolet analysis provided qualitative information with the advantage of instantaneous results that may be useful for screening samples. The autoradiographic analysis also provided qualitative results for comparison and image analysis. The novel non-destructive technique involves acquiring gamma-ray spectra with high-purity germanium detectors placed on both sides of a panel and relies on relating the difference in gamma-ray attenuation for different energies to the spatial distribution of contamination through the thickness of the panel. Both quantitative and qualitative results from this study indicate that the contamination did penetrate into the volume of the transite. However, this penetration depth was observed to be strongly dependent on the manner in which the transite was exposed to the contamination. Consequently, it is likely that significantly different penetration depths will be observed for different processes, buildings, and sites.

INTRODUCTION

A typical decontamination and decommissioning concern for both Department of Energy (DOE) facilities and nuclear power industry facilities is accurately determining the depth to which radiological contamination has penetrated building materials (1,2). The optimal means of decontaminating or disposing of such materials will often be dependent on this penetration depth. The building material of primary concern in this study is transite, an asbestos-cement material used in construction of both internal and external walls in process buildings at the DOE Fernald site. Transite was used extensively as building panels at DOE sites (Fernald and others) prior to the implementation of asbestos-controlling regulations. Uranium processing at the Fernald site has resulted in the radiological contamination of these panels. In the current era of remediation, prior to the disposal of these transite panels, the multiple concerns of toxic (asbestos-containing) and radioactive (process contaminated) waste must be addressed. Such a mixed waste is much more costly to dispose of than waste which is solely toxic or radioactive. It is therefore desirable to minimize the amount of mixed wastes. In the case of Fernald transite, surface decontamination may potentially restore the bulk of the material to the simpler toxic waste form. The viability of decontamination hinges on the distribution of the contamination through the thickness of the transite panels. Consequently, characterizing this contamination distribution is the first step in the disposal of the transite. Note: the term "contamination distribution" will be used to indicate the distribution of contamination through the thickness of transite samples, and does not refer to the distribution of contamination on the surface of the samples unless specifically mentioned.

The potential exists for a variety of destructive and non-destructive techniques to be used to characterize the contamination distribution through the thickness of transite. In this study, the choice of techniques was based on selective criteria, including minimizing cost,

ability to perform in-house, time constraints, quantitative or qualitative nature of the results, ability to measure contamination distribution, accuracy, and precision. Possible techniques were grouped into three categories based on the ability of the method to provide information about the contamination distribution in the sample: 1) destructive techniques which require destroying the sample to provide any information; 2) non-destructive techniques which only provide information about the surface of the sample and therefore require destructive analysis to determine the contamination distribution; and 3) non-destructive techniques which directly provide information about the contamination distribution. Analytical techniques considered included neutron activation analysis (NAA), atomic absorption spectrometry (AAS), and inductively coupled plasma spectroscopy (ICP), proton-induced X-ray emission (PIXE), X-ray fluorescence (XRF), backscatter spectroscopy, alpha, beta, and gamma-ray detection, ultraviolet (UV) fluorescence, autoradiography and gamma-ray spectroscopy. Requirements for the third category were not met by any known established technique. However, a novel non-destructive gamma-ray spectrometry technique has been proposed by Chung, et al. (3) and was implemented in this study. While the development of other non-destructive techniques was considered, all such techniques required some sort of activation or excitation source. This passive gamma-ray spectrometry technique represents a simple, inexpensive means of quantifying the contamination distribution. Considering the selective criteria, the techniques of beta and gamma-ray detection using a Geiger-Mueller (G-M) detector, UV fluorescence, and autoradiography, as well as gamma-ray spectrometry were chosen. All but the non-destructive gamma-ray spectrometry method required sequential layer removal and subsequent analysis of the exposed surfaces. Use of such destructive techniques was limited due to the difficulties in handling radiologically contaminated asbestos made friable by the destructive process. The establishment of a suitable non-destructive technique would bypass such handling difficulties, allow in-situ analysis, and minimize reliance on inferential statistics by allowing a much larger scale sampling plan. However, prior to validation of the novel non-destructive technique, the destructive methods were required.

SAMPLES

Transite is the brand name of an asbestos-cement construction material consisting of chrysotile asbestos (magnesium silicate) and portland cement. Manufacturing consisted of bonding multiple layers in a hydraulic press to form composite laminar panels. Transite was chosen as a building material for its strength and ability to resist moisture, heat, and corrosion. At the Fernald site, these panels typically have the dimensions of 1.22 m 3.05 m and are either flat or corrugated in form. The flat panels have a nominal thickness of 0.65 cm whereas the corrugated panels have a nominal thickness of 0.95 cm. In general, the flat panels were used for internal walls and the corrugated panels were used for external walls.

The samples used in this study to develop a methodology for determining the contamination distribution were taken from two different buildings at the Fernald site with very different process histories, representing extreme cases. One was Building 7, a dry process area where very little process contamination occurred, and the other was Building 2, a wet process area where more process contamination occurred. Two groups of flat panels (A and C) were taken from these buildings for analysis. Both

of these groups consisted of six panels with each panel being of approximately 25 cm 25 cm. Group A consisted of flat panels from the interior of Building 7. Group C consisted of flat panels from the interior of Building 2. Each panel within both groups was further labeled 1 through 6, resulting in panels A1 through A6 and C1 through C6. Furthermore, these panels were subdivided into 9 samples each, a through i. For example, panel C5 was divided into approximately 7.6 cm 7.6 cm squares: C5a through C5i. Each of these samples represents the basic unit of analysis and has an area of approximately 60 cm2. This sample size allows sufficient area for practical application of analytical techniques while attempting to maximize the degree of contamination homogeneity across the surfaces.

EXPERIMENTAL PROCEDURES

The techniques used in the destructive analysis for this study provided information only about the amount of contamination on the sample surface. To obtain information about the contamination distribution, layers were removed to allow measurements at various depths. In maximizing the number of successive surfaces analyzed by minimizing the thickness of each removed layer, it was possible to approximate the continuous contamination distribution.

Destructive Layer Removal

A 1.0 m 0.5 m 1.2 m high efficiency particulate air (HEPA) filtered glove box, built specifically for this study, was used to contain radiologically contaminated friable asbestos generated during layer removal. Samples of transite were placed in the containment along with a belt sander and micrometer. Each sample was then placed in a vise, such that about half of the sample thickness extended above the top of the vise. The belt sander with rough grit sandpaper was then used to remove layers of material. Applying the sander in a variety of directions ensured as uniform a removal as possible. With extreme care, it was possible to remove a sufficiently uniform layer with a minimum thickness of approximately 0.025 cm. Variation in the thickness measured by the micrometer at eight locations on the sample provided a measure of this uniformity.

Quantitative Destructive Analysis

Prior to the initial layer removal and after each of these layer removals, the sample was analyzed with a G-M detector, UV photography, and autoradiography. The G-M detector was used by placing the probe, which was about the same size as the sample, on the sample face and recording the count rate. Layer removal and analysis were continued until the level of contamination was below the threshold of detection (background level). The G-M detector was used because of its sensitivity to the beta particles emitted by the uranium daughters which represent the contamination.

Qualitative Destructive Analysis

When uranium is exposed to an oxygen-containing environment, such as air, uranyl ions (UO_2^{+2}) are formed. These ions fluoresce visible green light when exposed to UV light. Thus, when a sample is contaminated with uranium, exposing that sample to UV light provides a technique for localizing the presence and extent of contamination as the intensity at which the sample fluoresces is proportional to the uranium concentration. Consequently, the presence of uranium contamination can be determined qualitatively by direct observation. UV photography provides a direct measure of the distribution of contamination on the surface of the

sample, illustrating surface contamination homogeneity. Furthermore, through the use of standards to determine relative intensities, image analysis could likely produce quantitative results. Because UV light only interacts with the surface of the sample, this method provides information about contamination on the surface, and is very quick, simple, and inexpensive (4,5). For the UV photography in this study, the transite samples were placed in a light box equipped with ultraviolet fluorescent bulbs which emitted light at a 254 nm wavelength. A 35 mm camera was attached to the viewport with 200 speed film and an exposure duration of 4 seconds. For comparative purposes, consistent settings and development were crucial.

Autoradiography differs from typical radiography only in that the source of radiation is provided by the sample itself and a separate radioactive source is not required. Classical radiography involves capturing the image of the sample on a photographic plate based on the attenuation of radiation from an external source by the intervening sample, while autoradiography captures the actual pattern and intensity of radiation from a sample. Consequently, the only requirements for autoradiography are radiosensitive film and a means of handling the light sensitive media for exposure purposes. Since autoradiography also provided a spatial representation of contamination across the surface of the sample, illustrating surface contamination homogeneity, it was used for comparison to the UV photographic results for verifying which regions contained uranium contamination. Autoradiography provides surface or near surface information because the energetic charged particles that are most likely to interact with and register on the thin film are significantly attenuated with depth. The autoradiographs were produced using standard X-ray imaging film with a central polymeric base coated on both sides with a thin emulsion covered with an anti-scratch layer. The film was placed in direct contact with a sample and stored in a light-tight box for an appropriate exposure time. Consistent developing was provided by a standard automated developer.

Non-Destructive Analysis

The contamination distribution results from the destructive analysis were used to both characterize the transite and to provide verification of the novel non-destructive technique using gamma-ray spectrometry proposed by Chung, et al.(3). This non-destructive method was based on measuring the gamma-ray spectra from both sides of the sample. Figure 1 illustrates the implementation of this technique. Note the different photopeak areas (C1 and C2) recorded by the two detectors in the two gamma-ray spectra. In conjunction with knowledge of the gamma-ray linear attenuation coefficient for the material, the ratio of photopeak areas at several energies from these spectra can be used to infer the most-probable contamination distribution. Use of a high-purity germanium (HPGe) detector provided sufficient energy resolution to discern all of the photopeaks of interest. Comparing the measured ratio of the respective photopeak areas from both sides of the panel (Fig. 1-b) over a range of energies to computer-generated ratios for possible contamination distributions provided a prediction of the most-probable distribution (Fig. 1-c). The computer generated ratios were based on distributions predicted by diffusion theory for given types of exposures, or initial conditions. In particular, for a single instantaneous exposure, the distribution was based on a gaussian function, and for a constant

exposure, the distribution was based on a complementary error function (6).

Fig. 1

RESULTS

Destructive analysis was conducted on both sides of one A sample and eleven C samples using the G-M detector to obtain quantitative contamination distributions. Approximately six layer removals were required per side to reach background level. As representative G-M detector data, sample C5a produced the results shown in Table I and Fig. 2. Figure 2 also illustrates the correlation of the measured contamination distribution with the distribution predicted by the non-destructive technique. The shape of the measured contamination distribution is consistent with the diffusion theory predictions of the non-destructive technique. For this study, the condition of constant exposure was approximated and the complementary error function proved to be applicable. Similar contamination distributions were predicted and measured for the other ten C samples.

Table I

Fig. 2

To determine how the contamination distributions vary from sample to sample, the depth at which count rates from the G-M detector approached background were recorded. Table II lists the sample sides analyzed and their corresponding depth to background level. For the twenty-two sample sides analyzed from Building 2, the average background depth was 0.1460.0815 cm. Although the results in Table II illustrate significant variation and uncertainty, it is apparent that there are differences between samples with different process histories.

To further verify the penetration of uranium contamination into the thickness of the transite panels, both UV photographic and autoradiographic analyses were performed on several samples. Since the detection thresholds for both UV photography and autoradiography are significantly larger than that of the G-M detector, these analyses became insensitive prior to reaching background levels. Consequently, only about four layer removals of 0.03 cm each provided useful image information. The autoradiography required an exposure time of either 24 or 48 hours, depending on the contamination level. It was empirically determined that a G-M detector reading of about 1000 counts per minute (cpm) or greater could be exposed for 24 hours to produce a sufficient image. Lesser activity required a 48 hour exposure.

As representative of UV photographic and autoradiographic results, sample C5a produced the images shown in Figs. 3 and 4, where Fig. 3 shows the sample prior to layer removal and after a single layer removal and Fig. 4 shows the sample after two and three layer removals. The progression of decreasing contamination and image correlation for the same layer are illustrated by the dark green areas in the UV photographs which correspond to the light areas in the autoradiographs.

One concern with results based on layer removal using a belt sander was the possibility of cross-contamination between subsequent layers due to embedding by abrasion. The well-resolved contamination pattern images from the UV photography and autoradiography and the ability to achieve background level after several layers have been removed indicate that this effect is negligible.

Fig. 3

Fig. 4

CONCLUSIONS

The UV photographic technique proved to be a good qualitative tool for quickly analyzing the samples. The autoradiographs were used as a reference to provide a gross, qualitative indication of what aspects of the UV image corresponded to contamination. The autoradiographs were not limited by extraneous background imaging (visual noise), as the UV images were, since exposure of the film resulted only from radiation produced by the contamination. The threshold of detection is theoretically better for the autoradiographs than for the UV photographs, given sufficient exposure time. However, this autoradiographic exposure time is quite lengthy, on the order of days. Both UV photography and autoradiography provide the degree of surface contamination homogeneity.

Given the primary concern of determining whether the transite can be decontaminated such that the bulk of the material meets free release criteria, a key characteristic of interest is the depth to background or the thickness of material that potentially requires removal. Should this characteristic be too large, decontamination may not be viable for a given method of removal. Samples with a more significant contamination exposure history (group C samples - Building 2) had a background depth that resulted in approximately 40% of the total thickness of the transite being contaminated to some level. For samples with a process history that resulted in significantly less contamination (group A samples - Building 7), the background depth represented about 8% of the total thickness of the transite. The viability of decontamination is illustrated to be heavily dependent on the process history. For future studies, it is recommended that a greater variety of sample contamination exposure histories be used for a stronger correlation to background depth.

The contamination distribution results indicate the degree to which a given sample can be labeled as surface or volumetrically contaminated. The quantitative data provided by the G-M detector produced distributions consistent with the non-destructive technique. Therefore these G-M detector distributions adhere closely to what was predicted by diffusion theory, a complementary error function distribution, given the appropriate condition of constant exposure. The value of this technique is its concept of using gamma-ray spectrometry without destroying the samples and that the computer simulations produce a reasonable representation of the actual contamination distribution.

While this study has focused on the characterization of Fernald transite, other radiologically contaminated materials could also be characterized using these techniques. As long as gamma-ray spectra can be acquired from two opposing sides, a range of photopeak energies are available, and the assumption of uniform contamination across the surface is not restrictive, the non-destructive gamma-ray spectrometry technique can be used. To accurately predict the contamination distribution for a particular sample, a reasonably accurate knowledge of the manner in which the sample was contaminated is also required. In lieu of this knowledge, the destructive techniques can be used to establish appropriate mathematical representations of the contamination distribution. In any case, the destructive techniques should likely be used on a few samples from any new set of building material samples to confirm the results of the non-destructive technique. This recommendation will be necessary until the non-destructive technique has been further validated for a broader set of contaminants and exposure scenarios. However, it is envisioned that at some point the non-destructive technique will

accurately predict the contamination distributions without the need for any destructive measurements for confirmation. These techniques are well-suited for analyzing transite samples from other DOE sites. While the non-destructive technique and the destructive G-M and autoradiography techniques are applicable for any radiological contaminant that emits beta particles and gamma-rays, the UV photography technique is not applicable unless uranium or other contamination that fluoresces under UV excitation is present. Nonetheless, the techniques developed in this project would only require minor modifications, if any, to be useful for analyzing transite or other building material samples from other DOE sites for the contamination distribution through the thickness of the samples.

This study established a reliable means of determining the penetration of contamination in flat transite panels by destructive layer removal and analysis. In addition, the novel non-destructive technique using gamma-ray spectrometry has been validated. Specific sample contamination distributions have been shown to be dependent on exposure history, as consistent with diffusion theory, and should be applicable to materials other than transite.

ACKNOWLEDGEMENTS

We would like to thank the staff of the University of Cincinnati's Accelerated Life Testing and Environmental Research (ALTER) Facility, for their assistance and the use of their facilities. We would also like to thank Nathalie Tracez, a visiting graduate student of nuclear technology from France, who provided help in literature research and data collection.

This report was prepared as an account of work sponsored by an agency of the United Stated government. Neither the United States government or any agency thereof, nor any of their employees, nor any of its contractors, subcontractors nor their employees makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, manufacturer or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States government or any agency thereof, or Fernald Environmental Restoration Management Corporation, its affiliates or its parent companies. This work was supported in part by the United States Department of Energy under the DOE-FERMCO prime contract no. DE-AC05-920R21972

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SOLID LOW-LEVEL RADIOACTIVE WASTE VOLUME PROJECTIONS AT LOS ALAMOS NATIONAL LABORATORY

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ABSTRACT

In response to regulatory requirements, the current economic environment, and diminishing on-site low-level radioactive waste (LLW) disposal capacity, Los Alamos National Laboratory (LANL) needed to develop a system to collect data on future LLW generation that would comply with DOE Order 5820.2A and be an effective facility planning tool. The LANL Volume Projections Project (VPP) was created to meet these needs. This paper describes the objectives, scope, and components of the VPP that will provide information essential to future facility planning and development.

The VPP potentially involved the accumulation and evaluation of a large volume of data. The minimum data requirements were determined by considering the waste management facility's preliminary performance assessment results, on-site disposal space limitations, treatment options, and off-site disposal options. Using these data requirements, a questionnaire was developed to obtain the necessary waste stream information from the waste generating facilities. A pilot program was implemented to evaluate the sufficiency of the data gathered and the ease-of-use of the questionnaire.

An Oracle database will be developed to manage the information. When the database is completed, the questionnaire will be distributed to the five largest waste generating facilities (accounting for 75 percent of the solid LLW disposed of at LANL). The LANL-wide implementation of this program is scheduled for the following fiscal year.

DEFINITIONS

Container. A receptacle used to hold materials or waste for shipment, storage, or disposal.

Low-level radioactive Waste. Solid waste that is radioactive and is not classified as high-level waste, TRU waste, spent nuclear fuel, or tailings from the milling of uranium or thorium ore (1).

Normal Waste. Waste that is not generated through an environmental restoration project, a decontamination/decommissioning project, a spill clean-up, or an abatement project.

Performance Assessment. A systematic analysis of the potential risks posed by waste management systems to the public and environment, and a comparison of those risks to established performance objectives (2).

Waste generator. Any person by site whose act or process produces waste or whose act first causes a waste to become subject to regulation (3).
Waste Stream. A waste or group of wastes generated by a process at regular intervals or continuously over time that varies only within a narrow range of parameters (4).

ACRONYMS

BEMR	Baseline Environmental Management Report
DOE	Department of Energy
IDB	Integrated Database
LANL	Los Alamos National Laboratory
LLW	Low-level radioactive Waste
ORNL	Oak Ridge National Laboratory
PA	Performance Assessment
TSD	Treatment, Storage, or Disposal Facility
VPP	Volume Projections Project
WPF	Waste Profile Form

INTRODUCTION

The LANL Waste Management Program Office and its facilities must provide waste generation projection data to the Department of Energy, the University of California and other stakeholders. Because of the following concerns, the need for accurate waste generation volume projections has increase substantially: projected deep budget cuts for environmental management programs, and the need to assure the availability of waste treatment, storage, and disposal (TSD) capabilities. An annual survey will be used to gather facility specific estimates of future waste stream generation volumes to compile waste volume projection data with TSD capacity data. The questionnaire will be useful in performing compliance, reporting, and planning activities.

REGULATORY DRIVER

DOE Order 5820.2A, Radioactive Waste Management states that "Generators shall provide an annual forecast in the third quarter of the fiscal year to field organizations managing the off-site disposal facility to which the waste is to be shipped" (2). Additionally, the draft DOE 5820.2B, states that "Because of the long time period involved and the nature of the events and processes affecting disposal facility performance, there will be substantial uncertainties associated with the performance projections. A Performance Assessment Maintenance Program should include a process for reducing uncertainties in predictions about the long-term performance of the facility based on experimental and model improvement efforts. Projections should, at a minimum, provide the following: a listing of the types of waste to be shipped and an estimate of the as-transported volume and weight of each waste type and the activity of major radionuclides by isotope in each waste type" (5).

DISCUSSION

Between 1992 and 1995, LANL generated approximately 2600 cubic meters of LLW per fiscal year. Therefore, the VPP involved the accumulation and evaluation of a large volume of data. Because of the large number of LLW generators and the need to ensure that the amount of data gathered was manageable, it was necessary to limit the data requested to those items that directly impact the facility's disposal capabilities and those needed for DOE reporting activities. These items are identified in the PA, the Baseline Environmental Management Report (BEMR), and in the DOE Integrated Database (IDB).

The PA requires information on the volume of waste disposed of, the radionuclide content, the waste matrix, and the type of container used for disposal. Because of space limitations, treatment options, and off-site disposal options, similar information must also be evaluated to ensure that facility operations are not impacted. The disposal facility must ensure that it has sufficient capacity and that the construction of new disposal cells are budgeted for.

Additionally, the data obtain from this project will reported to the BEMR and the IDB. The BEMR includes "life-cycle cost estimates, tentative schedules, and project activities necessary to complete the Environmental Management Program" (6). The IDB provides "radioactive waste inventories and projections, ...for use in the planning and analysis of waste management functions" (7).

Questionnaire Development

Before the questionnaire was developed, representatives from LANL's waste management facilities, pollution prevention program, and waste management program office met to discuss the data required to meet reporting requirements. The following items were identified as minimum data requirements to be provided on the VPP questionnaire:

1. The fiscal year was chosen as the reporting period, since the amount of waste generated correlates with the funding received for the fiscal year. Data will be collected in the first quarter of each fiscal year for the current year as well as the next four years.
2. The volume is reported in cubic meters. Cubic meters was chosen as the reporting unit because it is the unit requested by the BEMR and the IDB.
3. The matrix categories are summarized in Table I and are similar to those used to report to the IDB and the BEMR The matrices are important to the evolution of the performance assessment, as the waste matrix directly effects the migration of radionuclides into the surrounding environment.

Table I

4. The container type is the container used to dispose of the majority of the waste. The container types of interest are listed in Table II. The container type affects the containment of radionuclides and the efficiency of the disposal unit.
5. A confidence level (i.e., low, medium, or high) was requested to determine variance in the data.

Table II

"High confidence indicates that the volume estimates are expected to be accurate to 25 per cent; Medium 50 percent; Low 100 percent or more" (6).

Using the minimum data requirements, a questionnaire was developed (see Appendix A). The data was requested for each waste stream as identified on LANL Form No. 1346, Waste Profile Form (WPF), which describes the physical, chemical, and radiological characteristics of a waste and the generating process. The questionnaire was reviewed by the disposal facility managers and the LANL waste management program office before the pilot program was implemented.

Target Audience

LANL operates using a facility management model. Therefore, the facility managers were identified as the target audience for the questionnaire because they are responsible for the activities that take place within the facilities. Each LLW-generating facility will be requested to provide volume, matrix, and container type for each normal waste stream referenced by the WPF. LLW generators will be provided with previous

generation rates to assist them in determining future generation rates. Previous generation rates will be obtained from the LANL Chem-LLW Database.

Pilot Program

A pilot program was developed and implemented to evaluate the data gathered and the ease-of-use of the questionnaire. The pilot program was initiated at a major LLW generating facility, the Plutonium Facility. This facility generates approximately 400 cubic meters of LLW per fiscal year, making it the second largest LLW generator at LANL. The two primary waste streams coming from this facility are a compactible waste stream consisting of paper, plastic, and glass and a noncompactible waste stream consisting of building debris, metal, and wood. During the pilot program, representatives of the generating facility were given the opportunity to comment on the questionnaire in addition to providing data for the next fiscal year. A complete questionnaire for Fiscal Year 1996 is located in Appendix B

RESULTS

As a result of the pilot program, no major changes were made to the questionnaire. In the next revision of the questionnaire, dose rate information (i.e., greater than 200 mrem/hr) will be requested. This information will allow the disposal facility to identify waste that requires a more restrictive disposal location.

The questionnaire will be reevaluated annually to incorporate changes in reporting requirements and facility needs. User comments will also be considered during the review of the questionnaire.

CONCLUSION

The VPP was developed to collect LLW volume projection data essential to future planning at the LANL LLW disposal facility. Not only does the current economic environment and diminishing disposal space warrant this type of activity, it is also required by DOE Order 5820.2a (2). The questionnaire has been developed and evaluated; however, because of budget constraints the remainder of this project has not yet been completed. Once funds are available, an Oracle database will be developed to manage the data. This database will allow LANL's waste management organizations to access the information regarding future LLW generation. Upon completion of the database, the questionnaire will be distributed to the five largest waste generating facilities (accounting for 75 percent of the solid LLW disposed of at LANL). The LANL-wide implementation of this program is scheduled for the following fiscal year.

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ACKNOWLEDGMENTS
Appendix A
Appendix B

38-13
DISTURBANCES OF ENCLOSING AND SURROUNDING MEDIUM WHILE CONSTRUCTING AND OPERATING OF REPOSITORIES FOR RADWASTE MEASURES FOR THEIR MINIMIZATION AND LIQUIDATION

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ABSTRACT
Construction and operation of underground radwaste repositories are inevitably connected with disturbances of the environment, the geological media, the compact group of rocks surrounding the repository and technical effects of waste placed within them. Acceptability of technical decisions is determined by an evaluation of the disturbances, consequences of the waste impacts on the environment and measures used to minimize or to eliminate these disturbances.
To sufficiently carry out such assessments to satisfy experts and allow the public to determine the acceptability of a repository construction, it is necessary to determine:
- type and scale of environmental disturbances;
- type and scale of technogenous impact on the environment;
- measures to minimize or eliminate the disturbances.

To more fully and qualitatively carry out these assessments, general forms and characteristics of disturbances of the environment were defined in the zone surrounding the repository. These assessments include geomechanical, hydrodynamic and aerodynamic series of disturbances and contaminations.
Taking into account the importance of full disclosure of the impact of the repository on the environment, a complex classification of technogenous impacts has been developed and is being presented for discussion. It includes technological stages of construction and operation of a repository, technogenous impacts and objects of their influence, possible negative results and recommendations for their partial or full elimination. The proposed classification can serve as a methodological basis for preparation and support of technical decisions.
Construction and operation of underground facilities for radwaste disposal is inevitably connected with disturbances of the natural environment and conditions resulting from operation of different production facilities in the region. This is the reason environmental protective authorities of Russia issued regulatory requirements which it is necessary:
- To evaluate the existing condition of the environment, and the techtonic and social media in the area of construction with ecologically dangerous technology and to determine its possible reserve levels;

To define possible disturbances of the formed medium and levels of additional loads from the new production facilities and on the basis of complex long-term predictions for development of ecological situations along with the assessment of acceptability of the proposed decisions;

To predict all possible measures to minimize disturbances of the formed medium, to correct it, or at least to bring new technogenous effects to ecologically acceptable levels.

Environmental assessments are required by State regulation to assess the impact of designed facilities for use with nuclear radiation technologies.

Completion of construction is based on complex investigations concerned with disclosing technogenous effects, scale and possible consequences. The first step, while carrying out such investigations, consists of determining type and source of impact on the environment and the scale and characterization of these impacts.

At the second stage, from the scale of impact, deterioration of the ecological situation is determined.

On the basis of these activities, the technogenous impacts scale is defined and assumptions are made for methods to prevent or reduce the problem.

While constructing and operating an underground repositories for radwaste disposal, the resulting technogenous impact on the environment and the lithosphere must be taken into account.

As to technological processes being the source of geomechanical disturbances, it is necessary to:

arrange sites for drilling and other geological exploratory activities;

construct roads, communications for industrial and social purposes, buildings and constructions:

minimize drift of workings, and drilling debris from boreholes;

engineer development sites;

dig foundation pits for technological purposes and for building and construction foundations;

arrange for sewage settlement ponds;

prevent erosion of upper soil covers during sewage discharge up to the ravine formation.

The sources of hydrodynamic disturbances include activities of land development, mining and objects operation. Reference is made to:

transfer of river-beds;

drainage of surface water reservoirs;

change of level and condition of underground waters while drifting of debris through aquifers or under-working of aquifers;

depletion of aquifers over the area of an underground repository and beyond its boundaries when underground waters come in contact with mine debris with the resultant pumping out into water reservoirs.

To sources connected with radwaste placed in an underground repository, the following must be considered:

heat release connected with radioactive decay;

radioactive radiation;

chemical effect of placed matter;

Sources of aerodynamic disturbances include:

building and high structure change rate, direction and character of air flow movement;

change of relief of an engineered development on site;

ventilation of air from underground facilities;

surface evaporation of settling basins and reservoirs-accumulators.

Biomorphological sources of disturbances include engineering development of an industrial site (removal of fertile soil and its stock, site, clearing of site, removal of vegetation, etc.).

Contamination of lithosphere takes place as a result of radwaste emplacement and migration of solutions containing hazardous elements leached from them.

Sources of hydrosphere contamination refer to:

hazardous matters and radionuclides leached from waste placed in the repository which migrate into the active water exchange;

sewage discharge and water discharge from construction of underground facilities;

Run-off contaminants accumulated on site surfaces by precipitation; fall-out of contaminated precipitation and dust from the atmosphere.

Sources of atmosphere contamination include dusting of ground and rock dumps.

Biological contamination of the environment occurs when wastes from production are transferred to fertile soils and sites for housing construction.

Analysis of contamination sources from construction, operation and conservation of underground repositories is an important step for assessment of impacts of those facilities on the environment. This produces a number of important tasks:

to comprehensively inspect the construction and production facilities in the region of an underground radwaste repository;

to determine possible environmental and technogenous consequences and degree of impact resulting from construction and operation of an underground radwaste repository;

to develop measures for the improvement of ecological condition at the construction site and determine measures for barrier protection while placing waste in the repository.

Classification of impact sources and main forms of their manifestation witnesses that they present a list of indices of such manifestations allow to carry out such assessments.

Because technogenous factors disturb the ecological balance in the construction region of a repository, construction requires a complete evaluation of the impact on the natural environment, determination of possible negative results and methods for overcoming the impact for each stage of construction and operation of the underground facilities. While creating of systems for waste isolation, classification of technogenous processes and disturbances is necessary:

Flaws inherent in the classification system are:

- utilization of known reasons and their characteristics of disturbances as a basis for classification (1,2);
- scantiness of factors and stages of works (3,6).

Disposal of radwaste in geological formations is a new scientific direction in geotechnology. The characteristics of this direction consist of a widescale and complex influence of an underground repository on the area of waste emplacement beyond its boundaries. Therefore, classification must take into account:

scale and character of influences on the environment during every stage of construction and operation of a repository;

the factor that long-term existence of an ecologically dangerous repository becomes an integral interconnected element of the existing environment;

to every technological process on every stage of operation of an underground repository those or other objects of the environment correspond (earth, flora, fauna, hydrosphere, etc.).

Taking these factors into account will yield a representative model for radwaste isolation in the natural environment.

We propose a classification of technogenous impacts and disturbances of the environment which may occur during disposal of underground radwaste. The classification given below takes into account technological stages for construction and operation of repositories, factors of technogenous effects on different objects and foreseen negative results. This classification, with recommendation for minimization or elimination, can be considered guidelines for estimating repository disturbances on the environment.

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38-14

AN ARID ZONE LYSIMETER FACILITY FOR PERFORMANCE ASSESSMENT AND CLOSURE INVESTIGATIONS AT THE NEVADA TEST SITE

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ABSTRACT

Two precision weighing lysimeters were installed near the Area 5 Radioactive Waste Site (RWMS) on the Nevada Test Site to provide support for investigations of water, solute and heat fluxes in the near-surface of the soil. An outdoor facility is necessary because the moisture and thermal regimes in the upper part of the unsaturated zone in arid regions are strongly influenced by the atmospheric conditions at the soil surface. The lysimeters consist of soil tanks with a volume of 16 m3

mounted on a sensitive scale. The top of the soil tank is flush with the ground surface and access to the side of the soil tank is provided through an underground entry. During installation of the lysimeters, soil was removed in lifts, screened, and set aside for repacking. The soil lifts were repacked in order to simulate the natural stratigraphy. One lysimeter was revegetated with native shrubs whereas the other was kept bare to simulate a non-vegetated waste cover.

Data consisting of physical and hydrological properties of the lysimeter soils, thermal and moisture conditions in the lysimeters, and atmospheric boundary conditions, are being collected for calibrating and verifying computer models for simulating the flow of water and heat in the near surface of alluvium at the Area 5 RWMS. This effort will provide site-specific models for demonstration of "no migration" of constituents to the water table. Physical and hydrologic properties are being determined using standard laboratory methods on repacked soil cores. Moisture and thermal conditions are monitored daily using time domain reflectometry probes at eight depths in the lysimeters and thermocouple psychrometers at ten depths. Daily evapotranspiration is calculated from the lysimeter scales. Meteorological variables are monitored by sensors mounted on a 3-meter tower adjacent to the lysimeters. An array of soil-solution samplers to be installed through the side of the soil tank will allow studies of waste mobility under natural conditions.

Conceptual designs for closure at the RWMS are focused on using an upper layer of repacked native alluvium. However, performance of other components such as a capillary barrier can be tested by installing a scaled version in one of the lysimeter tanks.

INTRODUCTION

Weighing lysimeters are an important tool for characterizing near surface transport processes, including the measurement of evapotranspiration (1,2). Two precision weighing lysimeters were installed near the Area 5 Radioactive Waste Management Site (RWMS) on the Nevada Test Site (NTS) to provide support for investigations of water, solute, and heat fluxes in the near-surface of the soil. Data from the weighing lysimeters, physical and hydrological properties of the lysimeter soils, and atmospheric boundary conditions are being collected for calibrating and verifying computer models for simulating the flow of water and heat in the near surface of alluvium at the Area 5 RWMS. This effort will provide site-specific models for demonstration of "no migration" of constituents to the water table.

The lysimeters consist of soil tanks with a volume of 16m3 mounted on a sensitive scale. The top of the soil tank is flush with the ground surface and access to the side of the soil tank is provided through an underground entry (refer to Fig. 1) . During installation of the lysimeters, soil was removed in lifts, screened, and set aside for repacking. The soil lifts were repacked in order to simulate the natural stratigraphy. One lysimeter was revegetated with native shrubs whereas the other was kept bare to simulate a non-vegetated waste cover.

Fig. 1

SITE DESCRIPTION

The lysimeter facility is located approximately 400 m west of the Area 5 RWMS on the NTS, which is located in the northern Frenchman Flat in southern Nevada. Frenchman Flat is a closed basin. The RWMS is at an elevation of 976 m on a bajada of the Massachusetts Mountains at the intersection of three alluvial fans on a slope of about 1 (3).

The NTS lies in a region that is transitional between the Great Basin Desert and the Mohave Desert. The climate of the area is characterized by a large number of cloudless days, low precipitation, and high daily temperatures, especially in the summer. Annual average precipitation is approximately 125 mm. The majority of rain falls during two peak seasons, with a greater peak in the winter and a lesser one occurring during the summer months.

MATERIALS AND METHODS

Each of the weighing lysimeters were instrumented with eight time domain reflectometry (TDR) probes to measure volumetric soil water content (storage component of the water balance) at depths of 10, 20, 30, 50, 70, 110, 140, and 170 cm; and 10 thermocouple psychrometers (TCP's) to measure soil water potential and soil temperature at depths of 10, 20, 30, 40, 50, 70, 90, 110, 140, and 170 cm. The approximate placement of these sensors within the lysimeters is illustrated in Fig. 2. TDR and TCP data are collected daily.

Core samples were collected from the lysimeters in 10-cm increments from 0 to 2m depths for characterization of physical and hydrologic properties. The physical property analysis included dry bulk density and porosity. The hydrologic property analysis included water retention relations, saturated hydraulic conductivity, and hydraulic conductivity-saturation relations.

The boundary conditions at the ground surface are provided by collecting hourly averages of micrometeorological parameters from a 3-meter micrometeorology instrumentation stand located next to the lysimeter facility. Inputs of water from precipitation are recorded with tipping bucket rain gauges and hourly averages of evaporation and evapotranspiration are obtained from the weighing lysimeters.

Fig. 2

PRELIMINARY RESULTS

Precipitation and evaporation data have been collected for nearly two years, thus providing a clear picture of the water balance of the area for the past two years. Figure 3 illustrates monthly precipitation measured at four stations located at approximately the four corners (NW, NE, SW, and SE) of the RWMS, and monthly evaporation (measured in the bare-soil lysimeter), and evapotranspiration (measured in the vegetated lysimeter) for the period of September 1994 to August 1995.

Winter rain amounts were greater than evaporation (E) and evapotranspiration (ET) during the months of December 1994, and January 1995. Data were not available for the vegetated lysimeter in February 1995. Monthly ET totals were greater than monthly precipitation totals after February 1995, and monthly E totals were greater than monthly precipitation totals after March 1995.

The data period presented resulted in an annual precipitation total of 186 mm, which is approximately 50 percent higher than the annual average amount of 125 mm. The ET total amount was 320 mm, and the E total amount was 158 mm. These results indicate that although annual precipitation was greater than bare-soil evaporation, it was far less than annual evapotranspiration, thus illustrating that plant water use is a significant component of the water balance, and therefore that plants may prove to be an essential component in cover designs.

Fig. 3

The resulting pattern of soil water distribution from this annual cycle of wetting and drying is illustrated in Fig. 4 for the vegetated

lysimeter. The soil water content profile prior to the winter rains was consistently dry. Prior to 1995, TDR probes were not installed in the top 50 cm of the lysimeters. Following the winter rains, elevated soil water content values were found in the top meter of both lysimeters. Redistribution over the next new months resulted in increased soil water contents at depth. Later in the year, after a period of drying, the soil water content profile returned to a consistently dry state, with volumetric water content values at approximately six percent in the bare-soil lysimeter, and three percent in the vegetated lysimeter (Fig. 4). The presence of plant cover enabled considerably more soil drying than occurred with a bare-soil cover.

Fig. 4

DISCUSSION

Preliminary results indicate that actual ET rates exceed precipitation rates for this region, even though precipitation was 150 percent of normal for the test period. Weighing lysimeter data indicate that each annual input of precipitation is entirely removed by annual ET, thus indicating conditions of zero recharge, and indicating ideal conditions for location of a RWMS, and demonstration of "no migration" of constituents to the water table.

Future plans for the lysimeter facility include studies of waste mobility under natural conditions using an array of soil-solution samplers to be installed through the side of the soil tank. In addition, conceptual designs for closure at the RWMS will be tested at the lysimeter facility, because most conceptual designs are focused on using an upper layer of repacked native alluvium, as is found in the lysimeters. Performance of other conceptual designs such as a capillary barrier can also be tested by installing a scaled version in one of the lysimeter tanks.

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38-16

TRANSPORTABLE VOLUME REDUCTION SYSTEM FOR SQUARE HEPA FILTERS

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ABSTRACT

There are numerous licensed nuclear sites which have a quantity of square, contaminated filter elements in store, awaiting disposal. The possible methods of treating these filter elements using a form of volume reduction are under review for the following reasons:

The UK Conditions of Acceptance for LLW disposal prohibit this form of waste without some volume reduction. The less radioactive filters can be incinerated in the UK but a proportion of them are too actively contaminated for volume reduction by this route or have been manufactured with metal frames unsuitable for incineration.

The square filters under consideration are of such a size that they are impossible to place within a standard waste drum for high force compaction, without some form of treatment

The purpose of this review was to produce a conceptual design of a Transportable Volume Reduction System (TVRS) to manage this waste stream, specifically for those filters which have metal frames or are too radioactive for incineration

OUTLINE TECHNICAL REQUIREMENTS

The developed system had to be readily transportable so that a number of sites could be visited, with a minimum of set-up or strip-down time. Careful consideration needed to be given to the requirement for and means of, decontamination prior to moving the TVRS on to public roads.

The system had to be designed so that materials other than square HEPA filters could also be volume reduced, so far as that was practical, and the products of the developed treatment had to be collected in a sealable steel drum, suitable for transfer to other on-site facilities, if required. A means of determining the volume of the contents of the collection drum had to be incorporated into the design. Overfilling of the collection drum was not acceptable.

The TVRS was to be provided with a form of containment around the treatment equipment, incorporating a ventilation plant built to the UK standard AECP 1054 - Ventilation for Radioactive Areas. Consideration had to be given to controlling larger pieces of debris which, although they might not become airborne, could escape the containment envelope. Conventional safety issues as well as radiological matters were to be addressed

The final dimensions of the TVRS were to be such that the unit could be driven into an existing building, equipped with the necessary services, i.e., radiological changeroom, electrical and compressed air outlets and a suitable discharge point for an on-board ventilation plant.

Standard, proven equipment was to be used in the design wherever possible to minimize the cost of hardware manufacture and operational maintenance. An estimate of overall costs and construction program was to be provided.

ASSUMPTIONS

The dimensions of the square HEPA filter elements were assumed to have external dimensions of 620 mm x 620 mm x 310 mm thick, with the filter frames being constructed of plywood, glassfibre or aluminum material.

It was also assumed that the waste collection drum into which the filters were to be packed were nominally of 200 liter capacity with dimensions of 572 mm internal diameter, 610 mm external diameter over the rolling hoops with an internal height of 813 mm and an external height of 863 mm. In addition, the drum would be fitted with a sealed lid and closure ring.

A condition set by the authors for acceptance of material for treatment was that it arrived sealed in heavy duty plastic bags with radiologically clean external surfaces and that the activity within the bags was of such a level that it would be eligible for disposal as Low Level Waste.

AVAILABLE PROCESSES

In reviewing the volume reduction of contaminated square filters, the authors investigated, in detail, the following processes:

Compaction ; Creasing ; Shredding ; Cutting ; Dissolution

Although the investigation was extremely thorough, only the main points of the different systems are summarized below.

Compaction was quickly dismissed on the basis that, although all UK reactor sites are equipped with in-drum compactors, as the filters will not enter the drum to permit compaction, the existing compactors cannot be utilized on the filter frames until some pre-treatment has been achieved.

A process of volume reduction, or more realistically, dimensional change to suit the collection drum, can be achieved by creasing the filter side frames to reduce one of the 620 mm dimensions. The amount of creasing, or deformation, required is a function of the diameter of the drum. Figure 1 indicates that one dimension of the filter would need to be reduced to around 437 mm to enable it to fit into the 572 mm diameter drum. The creasing operation shown taking place on two sides of the filter frame would produce a shape which would only permit one deformed unit to be placed in each drum. To improve the packing factor, the authors reviewed the possibility of putting other forms of loose waste with the filter before the final compaction process took place. Creasing was eventually dismissed as a candidate process because of the double handling aspect and the difficulty of handling a filter with a broken frame with airborne particulate being scattered between the creasing process and the collection drum.

Fig. 1

The process of shredding material by means of multiple cutting discs results in a product, the size of which is largely determined by the size of the perforations in the mesh below the cutting discs and the time allocated to the operation. Shredders which employ two rows of cutting discs rotating in opposite directions have a good reliability record when operating in dirty and neglected conditions. Normally they are provided with an ability to reverse the direction of cutter rotation when the system becomes blocked. The cutters will then resume the original direction of rotation. The products of the shredding operation continue to be recycled through the cutting discs until they are sufficiently small to pass through the mesh under the blades. Obviously, the smaller the perforations, the longer the process. The major disadvantage of the shredder system is that it produces large quantities of dust, demanding that the process would require some efficient containment enclosure. It is, however, a single handling operation.

A saw bench style cutting table was also considered, to reduce the frame of the elements into four individual components which could be placed inside the drum. The drum would, under this system, ultimately be filled with a large quantity of frame sides measuring 310 mm wide x 620 mm long. Like the shredder, this system raises a high degree of dust, demanding extensive containment and the depositing of radioactive solid waste, i.e. the filter element material, on the saw bench table. All movement of the waste into the drum would demand manipulation through the containment wall.

Although dissolution was considered as a means of producing a reduction in volume for the filter elements, it is obvious that it means employing large tanks containing acid (for the filter frames) and water (for the filter media) together with the required neutralizing tanks, dewatering equipment, pumps, valves, etc. The possibility of then transferring the solution to a site's Active Effluent Treatment Plant for ultimate

disposal of the liquid arisings is a complication that would best be avoided.

The selected process for the TVRS for the square HEPA filters was shredding for the following reasons:

- The shredding process is a single pass operation, which will result in a volume reduction ratio of at least 3
- It requires no hands-on intervention, indicating a minimum operator radiation dose uptake.
- The shredding mechanism is shrouded and can be extended to provide radiological containment
- The system has already been proven in radioactive environments on solid waste and initial trials indicate it can handle the frame of the square filter
- It is not expensive, or complex to operate
- It is rugged, practical and maintainable
- The shredder is light in weight, compared to the other processes reviewed, and has a small envelope size, important factors when considering a transportable concept.

As a result, the Concept Design discussed below has been based on the shredding process.

CONCEPT DESIGN

The Shredder

The basis of the concept is a standard, commercially available shredder, generally as shown in Figure 2, powered by a 15 kW electric motor, which drives the contra-rotating shredder blades via a robust gearbox. The shredder blades are, in fact, thick, hardened steel discs provided with cut-outs, installed on a shaft. As well as rotating in opposite directions, the shafts also turn at different speeds and are so installed that the two sets of discs intermesh with each other without coming into contact.

Fig. 2

A feature of this design of shredder is that if a blockage of the discs should occur, the motor control system reverses the direction of rotation, serving to clear the blockage. The electric motor will then be put into the normal direction of rotation to continue the shredding process. This forward/reverse motion will continue for up to 10 attempts, when a thermal overload sensor immobilizes the motor to permit it to cool sufficiently, before recommencing the process. Normally, blockages are cleared within the scheduled 10 attempts built into the control system. If however, the blockage is not cleared, it is probably due to the fact that the shredder has not been designed to handle the material loaded into the unit. For example, a 50 mm diameter steel bar would be considered too difficult to be treated by the shredder cutting discs. Equally, a blockage could occur because of a build up of much softer material such, as sheet steel, which when re-distributed by the reversing process, permits the shredder to handle it on a more progressive basis. By varying the cutter blade arrangements and fitting a suitably perforated grill beneath the discs, the particle size produced can be controlled, from fine granules to strips of predetermined length and width. As the waste material will continue to be treated in the shredder until it drops through the chosen grill size, the small particle size will take longer to produce than a more coarse shred.

A number of preliminary trials were conducted on a simulate filter frame constructed to the correct dimensions (620 x 620 x 310 mm) but fabricated

in 1.5 mm thick carbon steel for ease of construction. The trials proved that aluminum, wood and fiberglass frames should be easily processed by the shredder, all being softer material than the simulant frame. Further tests are advocated to confirm the earlier trials, particularly on the aspect of loading configuration of the filters into the throat of the shredder. The trials indicated, for example, that the frames needed loading in a particular manner to ensure complete, successful treatment and the concept design reflects that aspect.

The result of the trials has had a significant effect on the layout of the feed hopper, to ensure that the filters are loaded correctly, but the system can also accept bags of waste when required. The hopper is provided with a recessed hinged door which when within the hopper profile, would be sealed to the hopper wall. This door was needed to restrict the airborne particulate escaping from the hopper during the shredding process, when the ventilation system described later would be providing a scavenging flow across the door catching any loose dust that might escape from the closed door.

The opening of the hopper door will serve to stop the shredder from operating, diverting the ventilation flow downwards through the shredding machine, again restricting the release of particulate via the open hinged door.

Below the shredder is a delivery hopper directing the flow of shredded material into a collection drum which is sealed to the neck of the hopper by a sliding joint arrangement. The hopper is fitted with a gate valve to control the flow of material into the collection drum, aided by the small vibrator unit installed on the hopper, to ensure that the shred material does not remain stationary within the delivery system. The delivery hopper is supplied with a connection to the ventilation plant.

The sliding joint referred to above is required to provide clearance for the collection drum to be positioned below the delivery hopper, and to have a small vertical clearance to enable the operator to remove and replace the drum lid at the appropriate moment in the process. A vertical movement of 40 mm was considered sufficient for this purpose. The Process Flow Diagram of the system is provided in Fig. 3.

Fig. 3

Ventilation System

The ventilation system provided in the concept will meet the requirements of the UK standard, AECP 1054 - Ventilation for Radioactive Areas, with the capability to produce a maximum air speed across any openings in the containment envelope of 1 m/second. The unit in the concept consists of a canister with openings at each end providing access to a pre filter at one end and a HEPA filter at the other, each of which are circular and can be replaced employing the safe change principle. Each of these filter elements is of such a size that it can be placed within the 200 liter drum for subsequent compaction, or passed through the shredder, if preferred.

The fan fitted into the outlet duct from the filter canister is a centrifugal type capable of providing a flow of 600 liters per second. The ductwork associated with the ventilation system was supplied with balance and control/isolating dampers to obtain the optimum flow and pressure conditions.

The connections between the ventilation plant and the shredder system are arranged to provide the following service:

a flow across the recessed hinged cover when the cover is inside and sealed to the feed hopper, ensuring that any particulate which might escape from the shredder containment is swept into the filters in the canister. This flow was to cease when the hopper door is opened.

a second flow pulling air down through the open feed hopper door and the shredder, exhausting via a connection in the delivery hopper below the shredder box. This flow to diminish to a predetermined low flow condition while the feed hopper door is closed and a collection drum is sealed to the delivery hopper. This reduced flow condition would ensure that a minimum of shredded paper particles get pulled into the ventilation filters thus causing early blockage of the pre filter element. The full flow will be restored when the sliding seal on the collection drum is raised for fitting of the drum lid, at which time, the flow will be both downwards through the shredder as well as sideways and upwards past the open top drum.

The exhaust from the fan is fitted with ducting to a location on the installation where it could be easily fitted with a flexible hose, as part of the TVRS supply, the other end of which will be connected to the site ventilation ducting. The shredder ventilation plant is also fitted with a small auxiliary fan and 50 mm diameter hose to facilitate the ability to clean up any active particulate which may have fallen from the delivery hopper, prior to the plant leaving a site. The debris caught by this auxiliary system will be vented into the main filter canister. Figure 3 indicates the System Process Flow Arrangements.

The Trailer

The shredder assembly, that is, the shredding machine, feed hopper and hinged door, delivery hopper, gate valve and sliding joint, is mounted on a support structure built on to the trailer chassis. The trailer, which would be specifically designed and constructed for this purpose, has been provided with a cut-out in the trailer floor to permit the drum to be positioned on the ground, immediately under the delivery hopper. This keeps the overall height of the trailer to a minimum. Naturally, the shredder support structure is designed to provide the necessary clearance over the drum top rim.

The ventilation system is also mounted on the trailer and all equipment is carefully positioned to provide the required towing hitch down load of 100 kg and to observe the maximum wheel loadings. See Fig. 4 for the trailer weight distribution.

Fig. 4

The design of the trailer is based upon employing readily available steel channel or rectangular tube, all plated with a smooth steel sheet for ease of any possible future decontamination procedure. This chassis provides the fixings for the shredder assembly, the ventilation plant, the control unit and the standard double axle rubber - in - torsion suspension units, incorporating overrun operated brakes. The trailer has also been provided with a weather containment skin of glassfibre reinforced plywood (GRP) panels, which provides an easily decontaminable surface, with suitable hinged openings for access to the following

the feed hopper
the ventilation filters
the shredder control system
the collection drum loading/unloading sequence
and for regular maintenance requirements

The Control System

An important aspect of the Outline Technical Requirements for this concept is the need for a control system to limit the quantity of shredded solid material which is placed in the collection drum. As has been stated, overfilling of the drum is unacceptable, demanding a reliable means of determining an early indication of the level of waste in the drum at any time in the filling process, as well as the final chosen level. The authors investigated a number of level sensors and finally chose to specify an ultrasonic type which is capable of providing three distinct signals over the height of the collection drum. These levels will be set to indicate:

approaching required level

at required level- above set level

The electrical control system is arranged to be fully integrated. For example, it will be impossible to initiate the start up of the shredder with the hopper feed door open. It will also be impossible to start the shredding operation if the level sensor indicates there is insufficient room in the collection drum for another shredded filter. A delay is also built into the control system to ensure the feed hopper door cannot be opened until the shredder drive has come to a complete standstill. Equally, the electrical actuators holding the delivery hopper on to the collection drum will not be able to be initiated while the shredder is rotating, even if slowing. The electrical interlocks will also make it impossible to disconnect a drum until the ventilation plant is in the correct mode, that is with an in-flow of air across the top of the drum and upwards into the delivery hopper. All of the control systems terminate at the control unit positioned over the front frame of the trailer and accessible through a hinged hatch. The electrical power supply, which will be provided by the site being visited, via a heavy duty cable connected to the TVRS control box by a suitable industrial plug and socket arrangement. This cable is part of the TVRS equipment.

Overall Dimensions

The overall dimensions of the assembled Transportable Volume Reduction System are provided in Fig. 4. They are such that the TVRS will be capable of entering a building via a 3 m wide x 3 m high doorway. The estimated overall weight of 3100 kg means that, in Europe, the TVRS could be towed by a Land Rover Defender type vehicle as it meets the Road Vehicles Construction and Use Regulations (1986)

DEVELOPMENT REQUIREMENTS

Although the performance capability of the shredding process is well documented, there are a number of areas of development required before the concept could proceed with the utmost confidence. Briefly, these are:

More tests need undertaking on the shredder to determine if it would be sufficient to only produce strips of material, with an obvious advantage of reduced cycle time, or should the shredder be employed for a longer period of time, to produce a finer material, with more dust, but with an improved volume reduction factor.

As the drums of shredded material are likely to be transferred to a site based in-drum compactor for further size reduction, there would seem to be an overwhelming advantage to producing strips rather than fine particulate, although this policy would result in a higher consumption of drums and, therefore, higher cost.

Whichever shredding policy is adopted, the effect of dust arising from the shredder and settling on the window of the ultrasonic level sensor needs further investigation.

although it has been estimated that this concept should easily treat 4 filters per hour, this throughput must be checked when undertaking the above development work.

OPERATIONS AND SAFETY ISSUES

There are a number of important safety aspects which have been addressed in this design concept and in the proposed operational procedures, the principle issues being:

- the shredding process always takes place within a closed environment
- that environment is always subject to a ventilation in-flow of 1 m/sec. when there is any breach in the primary containment
- the shredding process will not be permitted to continue to operate when the feed hopper door is open, avoiding any possible injury to the operator and minimizing the release of radioactive particulate to the atmosphere.
- any loose particulate released from the shredded square filters should be captured on the pre and HEPA filters provided.
- the shredding process is a single operation which will not require any intermediate hands-on procedures, once the process begins
- any large loose debris falling from the shredder will be deposited on the pre/HEPA filter system by the small vacuum cleaner provided.
- the pre and HEPA filters fitted to the on-board ventilation plant are circular and will be bag posted out into 200 liter for compaction or shredding.

The concept has been subject to a preliminary Safety Review, the outcome of which was the following recommendations:

- although it was accepted that the shredder system could be thoroughly cleaned prior to moving off a site it is recommended that a special drum is put in place and sealed to the delivery hopper, to catch any loose debris that might be shaken loose from within the shredder body during the road journey.
- as a second means of preventing active material falling from the shredder, it is recommended that the final shredding operation prior to final shut down be undertaken with a small bag of clean combustible waste. The resultant very low level waste could then be treated in the site's incinerator.
- The contaminated circular filters from the on-board ventilation system should be bagged out and replaced with clean units prior to travelling on public roads. The spent filters should be either sent to the site's compaction plant for volume reduction or stored awaiting the next TVRS visit.

COSTS AND PROGRAM

The budgetary costs of construction of this Transportable Volume Reduction System for Square HEPA Filters was estimated at 120,000 ($185,000) at 1994 prices.

The construction program is dominated by the lead time required by the ventilation plant suppliers, that is 10 weeks. With careful planning and project management, the overall construction program should consume a period of 15 weeks.

CONCLUSION

All of the processes involved in the concept layout are readily available and well proven. They demand little development work to make them suitable for this application. The safety aspects of the concept appear to be acceptable for the levels of radioactivity involved.

The next phase of this project is to investigate the potential savings in disposal costs due to the volume reduction obtained and compare them with the overall finance required to implement this concept. A decision to proceed to construct this Transportable Volume Reduction System will depend on the outcome of that investigation.

ACKNOWLEDGEMENT
The authors would like to express their gratitude and thanks to Nuclear Electric plc and Golder Associates (UK) Ltd for the help and assistance provided in the preparation of this paper.

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CSTREAT - SUPERIOR ION EXCHANGE MATERIAL FOR THE REMOVAL OF RADIOACTIVE CESIUM FROM NUCLEAR WASTE EFFLUENTS: PRESENT EXPERIENCE IN PLANT OPERATION AND FUTURE POTENTIAL

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ABSTRACT
CsTreat is a granular hexacyanoferrate-based ion exchange media suitable for use in fixed bed columns for the removal of radioactive cesium ions from nuclear waste solutions. IVO CsTreat System was put into use at IVO's Loviisa NPP (VVER-440), Finland, in 1991 for the removal of 134,137Cs from high-salt evaporator concentrates and a total of 700 m3 (185,000 gallons) has been purified using only 80 liters (2.8 cu.ft) of CsTreat media. This corresponds to an average treating capacity of 8,750 L of waste solution by 1 L of CsTreat media (66 000 gal/cu.ft) in this high-salt waste stream. Two other commercial applications have been sold. CsTreat has also been used in Paldiski, Estonia, to purify low-salt waste solutions resulting from the operation of training reactors by the ex-Soviet Navy. In this purification campaign, a total of 760 m3(200,000 gallons) of waste solution originating from different waste tanks was purified with 12 liters (0.42 cu.ft) of Cstreat media. This corresponds to treating capacity of 63,300 L/L (474,000 gal/cu.ft) of CsTreat media. Since no indication of the exhaustion of the CsTreat media was observed during the purification campaign, the actual treating capacity for this solution is even higher.
So far, full-scale operating experience has been gained in the treatment of both high-salt evaporator concentrates and low-salt solutions arising from the operation of nuclear reactors. These solutions were low-active, the activity concentration of 134,137Cs ranging from 1.5 kBq/L to 200 kBq/L (410-5 - 5.510-3 mCi/ml). Laboratory testing of the material has been carried out for other types of simulated solutions: alkaline high salt effluents from reprocessing plants, medium-salt waste from reprocessing plants and low-salt NPP effluents. Results obtained confirm the high efficiency of CsTreat also in these waste streams. Tentative

process flowsheets have been outlined for these applications. In order to support new applications, thermodynamic modelling studies are underway to predict the capacity of CsTreat in feeds containing most common interfering macro-ions (Na, K, Ca, Mg).

A comparison of the performance of CsTreat media with those of other Cs-selective ion exchange medias (zeolites, silicotitanates, Cs-selective resins), as published in open literature, shows that CsTreat has superior efficiency.

INTRODUCTION

Insoluble transition metal hexacyanoferrates are highly selective ion exchangers for cesium. These have been used as precipitants for the removal of radiocesium from nuclear waste effluents at reprocessing plants and research centers (1). In these processes, pre-formed hexacyanoferrate slurry, or pre-cursor reagents of an insoluble hexacyanoferrate, have been added to the waste solution. Using hexacyanoferrate in packed-bed columns is much more efficient, but difficulties have arisen in manufacturing hexacyanoferrates in granular forms suitable for this type of operation.

The purpose of our research and development is to produce highly selective ion exchange materials for the separation of radioactive elements from nuclear waste solutions in order to obtain high reductions in final waste volumes and in radioactive discharges into the environment. This report describes experiences with both the laboratory testing and the actual industrial use of CsTreat, and also presents the modeling of cesium ion exchange for predicting the performance of CsTreat in actual processes.

EXPERIENCE OF INDUSTRIAL-SCALE USE OF CsTreat

CsTreat is a granular hexacyanoferrate-based ion exchange media suitable for column operations. The first system utilizing this media was put into use at the Loviisa NPP, Finland, for the removal of radiocesium from high-salt evaporator concentrates (2,3). These solutions contain high concentrations of $NaNO_3$ (2-3 mol/L), KNO_3 (0.1-0.2 mol/L) and $Na_2B_4O_7$ (0.5-0.8 mol/L). $^{134,137}Cs$ are the dominating waste nuclides in the waste solution, typically containing more than 90% of the total activity in solutions stored for a few years. The activity concentration of $^{134,137}Cs$ has typically been in the range of 40-200 kBq/L (1-510-3 mCi/mL). Since 1991, 700 m^3 (185,000 gal) of evaporator concentrate has been purified using 80 L (2.8 cu.ft) of CsTreat media. In normal operation, the breakthrough capacities (1% breakthrough) have been in the range of 50-180 m^3 per one 8 L (0.28 cu.ft) column (47,000-170,000 gal/cu.ft). Decontamination factors (DF) obtained for $^{134,137}Cs$ at Loviisa have ranged from 1000 to 2000.

In Paldiski, Estonia, CsTreat has been used to purify low-salt waste waters originating from the operation of nuclear training reactors by ex-Soviet Navy. In 1995, a total of 760 m^3 (200,000 gal) of these solutions, stored in four different tanks, were purified with a single 12 L (0.42 cu.ft) column (7.5 kg of CsTreat media), and no indication of the exhaustion of the column was observed when the purification campaign was completed. The activity concentration of $^{134,137}Cs$ in these waste waters was initially in the range of 1.5 -80 kBq/L (410-5 - 2.210-3 mCi/ml) and 0.001-0.01 kBq/L (2.710-8 - 2.710-7 mCi/mL) after purification.

TESTING OF CsTREAT IN SIMULATED WASTE SOLUTIONS

Experience obtained in these large-scale operations indicates that CsTreat has potential for the treatment of a wide range of nuclear waste

solutions. Laboratory-scale testing has been carried out batch- and columnwise in various types of simulated solutions. Some results obtained in column experiments are described below.

High salt reprocessing waste

Composition: 250 g/L $NaNO_3$, pH = 10, 0.5 ppm Cs + 134Cs tracer

Results: See Fig. 1:
DF=15,000 - 30,000 (detection limit)
Breakthrough capacity: No breakthrough detected at 4,000 bed volumes when experiment stopped

Medium salt decontamination waste

Composition: Na 5 g/L, NH_3 0.05 g/L, Na-oxalate 0.05 g/L, Nitrilodiacetic acid 0.05 g/L, Na-polyphosphate 0.05 g/L, pH = 10, 134Cs tracer

Results: See Fig. 1:
DF = 1,000-3,000
Breakthrough capacity: No breakthrough detected at 4,200 bed volumes when experiment stopped

Medium salt regeneration solution

Composition: Na 6.7 g/L, K 3.3 g/L, pH 6, 134Cs tracer

Results: See Fig. 2:
DF = 6,000 - 7,000 (detection limit)
Breakthrough capacity: No breakthrough detected at 2,000 bed volumes when experiment stopped

For zeolite Zeolon 900 (mordenite) as comparison:
DF < 50
Breakthrough capacity: instant breakthrough, column exhausted at 500 bed volumes

Low-salt NPP waste water

Composition: Na 320 ppm, Ca 17 ppm, pH = 6, 134Cs tracer

Results: See Fig. 3:
DF = 2,000-20,000, average 5,000 (high bed)
2,000-15,000, average 2,500 (low bed)
Breakthrough capacity: No breakthrough detected at 30,000 bed volumes (low bed), experiment continues

In all the above tests, CsTreat showed very efficient performance. For practical reasons, most of the experiments were discontinued after a flow of 400-4,000 bed volumes of solutions. Rather high activity concentrations of 134Cs (0.01 mCi/mL) had to be used in the solutions to measure accurately the high decontamination levels. In order to avoid high build-ups of activity in the columns experiments were discontinued even though no breakthrough of 134Cs from the columns was observed.

Fig. 1

Fig. 2

Fig. 3

COMPARISON OF CsTreat WITH OTHER CESIUM-SELECTIVE MEDIA

Comparison with other commercially available ion exchanger materials (Table I) indicates that CsTreat is a superior exchanger for cesium. Distribution coefficient (for definition, see later), which is directly proportional to the column performance, is at least two orders of magnitude higher for CsTreat than for mordenite zeolite and the organic resin. Zeolites, which are used in many nuclear applications, are usually synthetic mordenite, chabazite or natural clinoptilolite. Chabazite and clinoptilolite have ion exchange properties comparable to those of

mordenite. As it can be seen from Table I, none of these materials is very effective for the removal of cesium from concentrated salt solutions. Also in more diluted solutions CsTreat works much more efficiently. For example, in a clinoptilolite column 1% breakthrough occurred even below 10,000 bed volumes when fuel pond water having 0.004 mol/L sodium ions was treated (4). In the purification of a comparable solution at the Paldiskij Base with CsTreat, no breakthrough was observed at all after the termination of the purification at 63,300 bed volumes.

Table I

A new sodium titanate impregnated zeolite (IONSIV IE-96) product has also been recently reported for the purification of high-active concentrated waste solutions (5). From a 2.1 M Na+ solution the 1% cesium breakthrough occurred at about 120 bed volumes. This is close to what one could expect for conventional zeolite based on the Kd-value presented in Table I (see also Eqs 1-3 later). Estimated from the Kd-value, the capacity of CsTreat would be about 1,000 times as high in this solution.

A new organic resorcinol formaldehyde resin has been developed at the Savannah River Laboratory (6) and is also available commercially. This exchanger is much more selective for cesium than other organic resins and exhibits a cesium removal efficiency comparable to that of zeolites. For example, the 1% breakthrough of cesium from a resorcinol formaldehyde resin column occurred at about 100 bed volumes in a high active waste simulant having 5.6 mol/L of sodium ions.

Durasil 230 is an inorganic oxide-based cation exchanger used at nuclear power plants for cesium removal from diluted waste solutions. It has a high throughput capacity (100,000 gal/cu.ft) but the decontamination factor is modest (250) compared with 1,000-10,000 obtainable with CsTreat (7).

IONSIV IE-911 is a crystalline silicotitanate product, which has a better ion exchange performance for cesium compared with plain titanates. 50% breakthrough values from two waste simulants, Hanford DSSF-5 and ORNL W-27, were 540 and 500 bed volumes (8), which are rather modest compared with those typical of CsTreat.

MODELLING OF ION EXCHANGE EQUILIBRIA

Fundamental studies of ion exchange equilibria of radioactive cesium and macro-ions commonly present in solutions (Na,K,Ca,Mg) is underway with a view to modeling the equilibria and to predicting the capacity of the CsTreat exchanger in different waste effluents. Here the results obtained in the three-component system (Cs/Na/K) are described, and corresponding selectivity data are used to predict the capacity of CsTreat in "model" solutions.

General Theory

What is of interest in the purification of nuclear waste effluents is the capacity of the ion exchanger in terms of solution volume (V liters) that can be treated with a given amount (m kg) of ion exchanger. This capacity is unambiguously given by distribution coefficient KD of the radionuclide. Distribution coefficient KD is defined as the equilibrium ratio:

Eq. 1

where [M]r and [M] are the equilibrium concentrations of ion M in the ion exchanger (mol/kg) and in the solution (mol/L), respectively. For column operation, KD gives the total column capacity in terms of L/kg (volumetric capacity). The distribution coefficient is determined by two factors: selectivity and ion exchange capacity. Taking the Cs/Na exchange

as an example, the equilibrium is determined by selectivity coefficient kCs/Na of the exchange reaction, i.e.

Eq. 2

where subscript r refers to the ions in the exchanger phase. Inserting [Na]r = Q - [Cs]r (Q = ion exchange capacity) into Eq.2 and solving for the KD of Cs gives

Eq. 3

It can be seen that when [Na]/kCs/Na >> [Cs], the volumetric capacity of the ion exchanger (in L/kg) is independent of the Cs concentration in the solution. This condition is fulfilled when radioactive Cs ions are present in the solution containing a large excess of Na. For instance, the chemical concentration of Cs in solution corresponding to the activity concentration of 1 mCi/L is 8 x 10-11 mol/L. Selectivity coefficients kCs/Na are typically on the order of 10 in common organic resins and on the order of 10-100 in zeolites. Thus, unless the Na concentration in solution is very low ([Na] << 10-8 mol/L), the KD (and volumetric capacity) of the exchanger is independent of the concentration of Cs in the solution and the familiar relationship is obtained for the KD, in the logarithmic from

In other words, the KD of the trace ion is inversely proportional to the concentration of the macro-ion in the solution. Selectivity coefficient can be assumed constant in this case as the loading of Cs in the exchanger is very low.

Selectivity of CsTreat for Cesium

Distribution coefficient KD of trace Cs in CsTreat has been determined in NaNO3 and KNO3 solutions of different concentrations for the Na- and K-forms of the exchanger, respectively. Binary selectivity coefficients for the Cs/Na and Cs/K exchanges have been determined from these measurements. The following values have been obtained:

kCs/Na = 1.56106

kCs/K = 4.8104

Selectivity coefficients usually decrease as a function of ion loading in the exchanger. Experiments carried out with higher concentrations of inactive Cs show that the selectivity coefficients remain in these high values at least up to the Cs loadings of 0.2 mmol/g. The capacity of the Na-form exchanger for Cs is 0.3 mmol/g and that of the K-form exchanger is 0.5 mmol/g. Experiments carried out in the ternary system (134Cs/Na/K) show that in the mixtures of NaNO3 and KNO3 solutions at a given constant total concentration, logKD is a linear function of the amount of exchangeable K (or Na) in the exchanger (Fig. 4). This makes it possible to estimate the KD in mixtures of Na- and K-salt solutions, when the corresponding Na and K contents of the exchanger are known, eg. from the binary Na/K exchange data.

Fig. 4

Table II shows some KD-values and volumetric capacities of CsTreat, predicted from the equilibrium data above for different model solutions and for two batches of evaporator concentrates processed at Loviisa NPP. The predicted volumetric capacities obtainable with CsTreat are very high. These values (Table II) represent total exhaustion capacities of CsTreat columns (which corresponds to the solution volume at 50% breakthrough for a symmetrical breakthrough curve). Actual breakthrough capacities are somewhat lower, depending on the kinetics of the exchange reaction and on operating conditions (solution flow rate, exchanger grain size, column depth, temperature etc.).

Table II

At Loviisa NPP, batch KD-values of 137Cs have been determined for the evaporator concentrates in each tank prior to purification campaigns to get an estimate of the column capacity. Data show (Table III) that breakthrough capacities (1% breakthrough) are about 40% of the total capacities estimated from the measured KD-value (Q(BT)/KD (obs) = 0.37-0.38) at relatively slow flow rates of 10 BV/h. In dilute salt solutions, where the exchange rate is faster, it may be possible to obtain breakthrough capacities that are about 50% of the total capacities. Table III also shows some KD-values for the evaporator concentrates, predicted from the ternary Na/K/134Cs exchange data. The model calculations yield somewhat lower values than what are measured for the evaporator concentrates. The model is being further developed., e.g. considering the ion activities in solution, instead of concentrations, since it is clear that solution phase activity correction cannot be ignored in such concentrated solutions. The aim is to develop a model that could be used to predict accurately the capacity of CsTreat in waste streams of differing chemical compositions. Other ions, e.g. H, Ca and Mg, will be added to the model at a later stage.

Table III

FLOWSHEETS FOR SPECIFIC APPLICATIONS

CsTreat can be utilized in many different systems and applications. Some possible flowsheets for applications are described in Fig. 5.

The most straightforward application is a tailored system to purify liquid from an existing waste tank (Fig. 5a). If the waste liquid is highly acidic (pH < 1) or alkaline (pH > 13), pH control as a first step may be advantageous to improve performance. As normal, particles are filtered up-stream to avoid the fouling of the CsTreat bed.

CsTreat media can also be utilized in a loop configuration, e.g. to purify fuel pond waters (Fig. 5b). High ion exchange capacity minimizes waste amounts. In addition to particle filtration and ion exchange by CsTreat, some additional treatments may be required before water is recirculated to the pool.

A typical demineralizer system at a nuclear power plant (Fig 5c) may include coarse filtration, carbon filtration (removal of organics and oils), anion, cation and/or mixed bed with organic resins and a cesium specific ion exchange bed. In this application, CsTreat can be used to replace conventional cesium-specific materials like zeolites. Use of CsTreat, which has extremely high selectivity for Cs, results in a remarkable decrease in waste amounts. In many cases, no additional construction is needed, since existing systems can be fully utilized.

For the purification of high- and medium-active liquids, a system can be constructed in a hot cell (Fig 5d). High levels of activity can be loaded into the inorganic CsTreat material. Removal of other radionuclides, like 90Sr, can be connected to the same system. Typically, such treatment would result in reclassification of a large volume of liquid waste as low-active waste. Only a very small amount of ion exchange media remains as high-active waste. This gives high cost savings in further treatments and especially in the final disposal

Fig. 5

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DEVELOPMENT OF INDUCTION HEATED HYBRID TYPE MELTER WITH PLASMA TORCH FOR THE TREATMENT OF LOW LEVEL RADIOACTIVE SOLID WASTE FROM NUCLEAR FACILITIES

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ABSTRACT

Melting process has been discussed to be applied to the treatment process in Low Level Radioactive Solid Waste (LLSW) in its advantage of volume reduction and stability of processed solid form. Hybrid type Melting Process (HMP), which combines induction heating with plasma torch has been developed and investigated in this study. Non-radioactive simulated waste is fed to the melting furnace to investigate the characteristic of solidified products and off-gas. It is demonstrated that HMP can melt or decompose any waste, even concrete or PVC. Plasma arc gives high temperature and agitation to assist melting and increases NOX content in off-gas.

INTRODUCTION

In Japan, final disposal of Low Level Radioactive Uniform Waste (LLUW) which is cement solidified liquid waste has already been started, and as the second stage, Low Level Radioactive miscellaneous Solid Waste (LLSW) disposal facility is being designed. In this situation, the melting

process is one of the most promising processes for the treatment of such solid waste in its advantage of volume reduction and chemical/physical stability of processed solid form.

Since the radioactive solid waste consists of various materials, a melting technique to be adopted for the waste treatment needs to have flexibility in accepting various constituents of waste from different sources. To realize the flexibility of the melter, the Hybrid type Melting Process (HMP), which combines induction heating with plasma torch, has been developed and investigated in this study. Its main advantages are:

1. applicability to various waste materials
2. high uniformity and high stability of processed solid form
3. high volume reduction ratio
4. separation of harmful nuclides for disposal
5. simplicity of pretreatment
6. high processing efficiency
7. ability to take representative samples for the measurement of radioactivity from the melted waste

LLSW contains various waste materials, such as steel, aluminum, copper, ash, concrete, PE (PolyEthylene), PVC (PolyVinyl Chloride), glass, clothing, paper and so on. The radioactive levels vary among waste container and within each container. Such differences would make the heavy burden for the final repository, and give very high cost for disposal. It is not rational for the design of final repository and it is better that the waste is uniform. Moreover high volume reduction gives the elongation of life of the disposal site, and decreases the cost of disposals. Non-radioactive simulated waste is fed to the hybrid type melting furnace to investigate the characteristics of solidified products and off-gas. Some test results are summarized in the following sections.

TEST PROCEDURE

The test facility investigated in this study is shown in Fig. 1. It has a induction heating furnace with plasma torch. The furnace has 7-litter volume, and the induction heating capacity is 100kW - 2500Hz, the plasma torch capacity is 30kW respectively. The waste element of concrete, ash, heat insulator, PVC, PE are selected to represent the typical waste contents in nuclear facilities for the tests. Each waste element of non-radioactive simulated waste is initially prepared in a 2.4-litter steel container separately. Then the steel container is directly charged into the furnace and melted separately. After that the mixture of the waste elements are put into the same steel container and melted at one time.

Fig. 1

The weight of solidified products and slag composition are investigated after melting. The temperature of some important points and the off-gas composition are monitored during the operation. The situation during the melting operation and after melting operation is observed.

TEST RESULTS

The test conditions and main test results are summarized in Table I.

Table I

1. All of the inorganic compounds except concrete can be melted easily. The coarse aggregates in concrete are very difficult to melt only by induction heating, and those can be melted by using plasma torch concurrently as a result of heating and agitation of plasma arc.

Organic compounds like PE and PVC are easily decomposed by heat from melted waste, and caused much black smoke. It indicates that organic compounds burn imperfectly.
2. Volume reduction ratio for inorganic materials is observed to be 1/6-1/10. Organic materials are almost burned out and only little residue is observed.
3. The basicity (CaO/SiO_2) of the generated slag from inorganic compounds is analyzed to be low (0.2 to 0.8).
4. Typical results of Off-Gas measurement is shown in Fig. 2.
The evolution of NOX gas is observed when plasma torch is used. Plasma arc causes NOX content increasing in off-gas.
Fig. 2

DISCUSSION

Melting of Concrete

Concrete is not uniform compound but mixture of cement and aggregate that consists of sand and gravel. The melting point of cement and mortar is low, that is about 1400 and 1450 respectively and also cement and sand are very fine particles. Because of that cement and mortar are melted easily. But the melting point of gravel is more than 1700 and the reaction surface area of coarse aggregates is small comparatively. Therefore, usual method can not melt coarse aggregates. According to the phase diagram (1), which is shown in Fig. 3, if agitation of molten phase is enough, the same content of concrete requires a temperature of more than 1600 for melting. The usual melting process can heat up to very high temperature at the center of the liquid, but at the surface the temperature is far lower than the center. It is the reason why such process cannot melt concrete. HMP gives high temperature surface and strong agitation to the liquid, and in this condition concrete can be easily melted. The coarse aggregates in concrete can be melted by using the plasma torch concurrently as a result of heating and agitation of plasma arc.
Fig. 3

DECOMPOSITION OF PVC

PVC is very difficult to be burnt because of its chlorine, but PVC can be decomposed at more than 200, and carbon particle and hydrochloride appear in off-gas, then a part of carbon particle is oxidized by air into carbon monoxide (CO). Carbon monoxide in off-gas is measured and the result is shown in Fig. 4. From this result, peak decomposition rate is observed, and it is about twice of the average rate. The temperature in off-gas system was also measured. The temperature did not increase by decomposition and burning. This proves difficulty of burning of PVC.
Fig. 4

NOX by Plasma

NOX was also measured during the melting operation, and the typical result is shown in Fig. 5. Maximum concentration of NOX is about 40 ppm at the condition of 20.8% O_2. This oxygen content is quite higher than usual incineration. Plasma heating does not make wastes burn, just only decomposed, then oxygen will not decrease by plasma. In this reason, NOX content should be appreciated.
Fig. 5

CONCLUSION

The following conclusions are provided through this investigation.
1. HMP is the superior method to melt LLSW.
2. HMP gives very high volume reduction rate for LLSW.

3. HMP can melt or decompose any LLSW, even concrete or PVC.
4. Plasma arc increases NOX concentration in off-gas.
5. The slag generated by melting of LLSW is 0.2 to 0.8 in basicity.
6. The slag is produced as a stable glass form.

Hybrid Melting Process will be the best way for processing the low level radioactive miscellaneous waste. Further investigation will be conducted to apply this technology to other types of waste treatment.

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BITUMEN IMMOBILIZATION OF AQUEOUS RADWASTE BY THIN-FILM EVAPORATION

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ABSTRACT

In the early 1980's, AECL, at the Chalk River Laboratory (CRL) site, built a Waste Treatment Centre (WTC) for managing low level solid and aqueous liquid wastes. The objective was to demonstrate processes for converting Canadian Deuterium Uranium (CANDU) wastes to a form suitable for disposal while meeting or exceeding current environmental regulations.

At present, two liquid waste streams are being treated at the Waste Treatment Centre. The liquid waste streams are volume reduced by a combination of continuous crossflow microfiltration (MF), spiral wound reverse osmosis (SWRO), and tubular reverse osmosis (TRO) membrane technologies (6). The concentrate produced from the TRO system and the volume-reduced MF backwash solutions are evaporated while simultaneously adding bitumen in a thin-film evaporator. A water-free product of chemical and radiochemical salts and bitumen is removed in 200 L galvanized steel drums for storage and eventual disposal in the CRL Waste Management Area.

The feed stream to the thin-film evaporator typically has a b/g activity of about 1 - 3 mCi/mL. This radioactivity would be concentrated by a factor of about 10, while simultaneously being immobilized. The radiation field of product drums on contact typically has a value of 0.5 to 3 R/h depending upon the feed concentration of radioactivity to the evaporator. The total solids content in the 200 L drum ranges from 25% to 35%. Encapsulated in the bitumen matrix are a variety of non-radiochemical salts which comprise the bulk of the total solids which are in the product drum. The drum contains less than 1% of free water, and the product passes the USNRC guidelines for a solidified wasteform.

The paper will discuss the volume reduction capability of the plant, with an emphasis on the immobilization of the aqueous waste with bitumen in a thin-film evaporator. Operations experience gained from over 200 campaigns are documented.

Samples of bituminized wasteforms from the Waste Treatment Centre have been leached in accordance with the ANS/ANSI 16.1 leach test. In this test, the wasteform is immersed under water for an extended period of time and the leachate is periodically removed and chemically analyzed.

The leachability index varied between 7 and 9 for the emulsified bitumen wasteforms produced at the Waste Treatment Centre, but could be improved significantly with the addition of protective overcoats of either Portland cement or oxidized bitumen.

INTRODUCTION

In 1975, the need for a Waste Treatment Centre (WTC) at Chalk River Laboratories (CRL) was identified. The WTC was built to demonstrate systems capable of converting liquid wastes to a stable and leach-resistant form. Current operations in the facility involve the treatment of the low-level CRL liquid waste using a 3-stage membrane plant to accomplish most of the volume reduction (comprised of microfiltration, spiral wound reverse osmosis, and tubular reverse osmosis unit operations). Further volume reduction is achieved with a thin-film evaporator (without bitumen addition), and another thin-film evaporator whose primary function is for bitumen immobilization.

Fig. 1

Figure 1 shows a flowsheet and volumetric balance for the primary unit operations at the WTC. For a basis of 100 units of liquid waste (or 100# in the figure), the quantities of liquid at various points in the plant are indicated. A total volume of 93.5# is released by the 2-stage reverse osmosis plant (per 100# treated), while a combined 99.9# is released from both the reverse osmosis and thin-film evaporator operations. A total volume of 0.4# (including emulsified bitumen volume) is the secondary waste product from the thin-film evaporator used for bituminizing (TFE-1). This results in an overall volume reduction factor of 250 for the plant, based on feed liquid volume to immobilized secondary waste volume, in the product drum (that is, 100 0.4).

The TFE-1 unit is used to immobilize liquid waste concentrates with bitumen. The other (TFE-2) is used to volume reduce microfiltration backwash concentrates and tubular reverse osmosis retentate prior to immobilization. The bituminized product from TFE-1 is collected in 55 US gallon (210L) galvanized steel drums. A full drum weighs approximately 240 kg. The drums are capped, transferred to a shielded container and then shipped to the site waste management area for storage and eventual disposal. The evaporation/immobilization process is carried out remotely in a shielded facility that is designed to handle product drums with contact gamma radiation fields as high as 10R/h. In practice, however, the radiation fields are maintained below 5 R/h.

At present, two radioactive CRL liquid waste streams are being processed by the thin-film evaporator used for bitumen immobilization (TFE-1). One stream originates from the central Decontamination Centre (DC). The other, an Active Drain (AD) stream, is produced from a large and diverse number of research laboratories and radioisotope production facilities. The two waste streams, totaling about 3000 m3 per year are volume reduced by a factor of 80 prior to immobilization by three membrane systems and thin-film evaporation. The performance of the thin-film evaporation process used for bitumen immobilization, and the characteristics of the wasteform after standard leach tests (using the ANS/ANSI 16.1 protocol), represent the focus of this paper.

Not much data is presently available on the performance characteristics of the current wasteform being produced at the WTC. The early work on leaching was on phosphate-based wastes encapsulated in oxidized bitumen. The reason for this is that experiments carried out about a decade ago on waste product simulants indicated that there was little difference

between the leaching properties of distilled (soft), versus oxidized (hard) bitumen. There are currently about 200 product drums of secondary immobilized secondary waste which have their own blend of contaminants and solids, and the performance of this product is currently being evaluated.

The leaching work reported here is the first step in the evaluation of the waste product currently being produced at the WTC. The ANS/ANSI 16.1 leach test (1) is an aggressive procedure that reflects unrealistically poor conditions for disposal in the unsaturated zone. For the tests carried out here, bitumen samples were unconfined and subjected to immersion and frequent leachate replenishment. The results of these leach tests will be a lower bound for the performance of the bitumen waste product in an unsaturated environment. Further work under more realistic conditions will provide a better estimate for the performance of the bitumen waste product in an unsaturated environment.

TECHNOLOGY DESCRIPTION

Membrane Plant

A flowsheet of the integrated plant for volume reduction and immobilization of aqueous radwaste is shown in Fig. 1. Liquid waste feed to the WTC is sampled and analyzed for pH, conductivity, a and gross b/g activity, tritium (3H), turbidity, total solids, gamma-emitting radionuclides, and various non-radioactive chemical species at the source tanks. Waste is then sent to a 45 m3 feed tank in the WTC and it is pH adjusted to the alkaline region (8 - 10); an adequate conditioning time of 12 hours is allowed to maximize precipitation of metals, prior to treatment with MF for the removal of suspended solids.

During microfiltration (MF), concentrate is recycled; filtrate is continually removed from each bank of filtration modules at about 25 L/min, and is directed to the spiral wound reverse osmosis (SWRO) feed tanks. When the backwash (concentrate) solution exceeds 10 g/L, it is sent to the volume reduction evaporator, TFE-2. The MF filtrate, free of suspended solids, enters the SWRO feed tank at a rate equal to the permeation rate of the SWRO. The concentrate from the SWRO membrane process is bled to the TRO feed tanks.

The TRO membrane system operates in a batch process mode, where the final tank volume is reduced to approximately 10 to 15% of the initial feed volume. The concentrate, containing about 50 to 80 g/L of dissolved solids, is transferred to one of three 7100 L tanks in the active tank room, to await further volume reduction by evaporation with TFE-2.

Liquid Waste Immobilization Process Description

The evaporator has a 10.8 ft2 (1 m2) heated surface, with superheated steam at about 100 psig (720 kPa) applied to the jacket. Waste feed containing between 5-15% total solids is introduced at a rate of about 2 ft3/h (1 L/min), along with bitumen emulsion flowing at about 0.6-1 ft3/h (0.3-0.5 L/min). The emulsified bitumen and the waste are introduced via a distribution ring to the top of the evaporator and fall along the heated surface. The evaporator has internal rotating blades, which sweep within 1 mm of the vertical heated surface. The blades in TFE-1 rotate at 900 rpm.

The evaporator does not operate full of product; the liquid or slurry forms a thin film or annular ring of product from the feed nozzle to the product outlet nozzle. Holdup or liquid inventory in a thin-film evaporator is very low, typically about 5 kg of material per square meter (0.5 pound per square foot) (3). A thin-film evaporator, which is an

inherently low-pressure-drop device, has mechanical turbulence and, therefore, good heat transfer properties over a wide range of viscosities.

DESCRIPTION OF BITUMEN USED FOR IMMOBILIZATION

Bitumen Feed

Bitumen is the name given to a wide range of hydrocarbons with high molecular weight that are commercially available as a residue of petroleum or coal-tar refining. Its two major components are asphaltene compounds, which give bitumen colloidal properties, and malthene compounds, which impart viscous liquid properties. Most bitumen is obtained during the distillation of crude oil and is called "direct distilled" bitumen. This bitumen has a high viscosity and must be heated to make it suitable for mixing. The bitumen used in the evaporator has been emulsified in water (contains 45% H_2O) and can be added directly to the evaporator without being heated. Table I shows the properties of typical bitumen used in the Liquid Waste Immobilization System.

Table I

Rationale for Selection of Bitumen Matrix

Solidification of wastes in bitumen has been tested extensively in Europe, and has been applied more recently at Virginia Power in the United States (5) . Experience has demonstrated that bitumen is also suited to most streams generated by nuclear power plants and by industry (4). The stability of bitumen with respect to radiation is a property of primary importance when bitumen is considered as a waste matrix material. The main factors influencing the radiation stability of bitumen are the dose-rate and the total absorbed dose. A total absorbed dose limit of 108 to 109 rad is commonly reported for the radiation resistance of bitumen-waste forms (4).

Bitumen generally provides superior leach resistance in comparison with cement, which could be important for certain species and disposal scenarios. One of the potentially more significant advantages provided by bituminization processes is volumetric efficiency, which is particularly important if waste must be stored for extended periods before disposal. Cement solidification can result in as much as a 100% increase in volume, whereas, depending on waste stream concentrations and the type of equipment used, bituminization can provide volume reduction factors of 5 or more, because associated water is driven off during the solidification process.

OPERATIONS PERFORMANCE OF TFE-1

Effect of Rotor Change

Data have been collected from TFE-1 after a new rotor was installed and operated at higher speeds. Table II summarizes the performance of TFE-1 before and after the rotor change. The new rotor operates at a speed of 900 rpm, compared to 600 rpm prior to its installation. A total volume of 26.9 m^3 was immobilized with TFE-1 in 1994 with a volume reduction factor (VRF) of 2.92. A volume reduction factor (VRF) is defined by Eq. (1).

Eq. 1

Prior to the installation of the new rotor a total volume of 13.2 m^3 of liquid had been processed with a VRF of 2.67. After the installation of the new rotor a total volume of 13.7 m^3 was processed with a VRF of 3.21. The cleaning frequency was reduced by a factor of three after the installation of the rotor operating at higher rpm. The higher VRF after the rotor change boosted the average volume of waste concentrate immobilized per drum from 535 L to 642 L. Over the duration of the year

(including data for both rotors) an average volume of 585 L of waste concentrate was immobilized per drum. This represents an overall VRF of 2.92. On average over the duration of the year, there was about 64.2 kg of waste solids immobilized in each drum. The drums contained approximately 30.3% solids by weight. Inert solids (associated with grit and suspended matter removed by microfiltration in the backwash stream) represent about 4 to 5% of the weight of the product drum. The immobilized waste solids are comprised primarily of sodium nitrate, sodium phosphate, sodium sulphate, and sodium carbonate. The average concentrations of the primary radioactive and chemical contaminants immobilized in the product drums are given in Table III. The data are expressed as a quantity of contaminant per unit mass of immobilized secondary waste, from TFE-1.

Table II

Chemical characterizations of the ammonium hydroxide and citric acid-based cleaning solutions used for TFE-1 (prior to the installation of the new rotor), have shown that there is severe scaling of the evaporator by phosphate- and sulphate-based salts. About 6 kg of sodium phosphate and 1 kg of sodium sulphate were deposited on the 1 m2 of evaporator surface area. Significant quantities of calcium, magnesium, iron, and silica were also detected in the ammonium citrate-based cleaning solution. Visual observations indicated that large deposits of hardened scale were present on the heat transfer surfaces, particularly within about 25 cm of the feed inlet. These observations indicated that all of the water associated with the waste solids was flashed off within approximately 0.3 m of the TFE-1 feed inlet. The length of the heated surface is 1.5 m.

Crusty deposits of scalable species are thought to be responsible for the reduction of the heat transfer area, and lower product temperatures prior to the installation of the new rotor. Product temperatures were recorded at one hour intervals during a processing campaign, which typically lasted approximately 14 hours. At the start of a campaign a process temperature of 155 oC was attainable but could not be maintained at this level for the duration of the run, due to the chemical scaling.

The chronic fouling observed prior to the rotor change has now been minimized. The frequency of chemical cleaning has been reduced by a factor of three after installation of the new rotor and this has minimized down time. It is now possible to maintain an outlet product temperature of between 135 - 145 oC for the duration of a 4000 L campaign. This ensures that there is no residual liquid in the product drum. There is less than 1% free water in a product drum based on mass balance.

For operations with the old rotor, the average combined distillate rate (evaporated from the waste and the bitumen) varied between 0.4 and 1.0 L/min. When the new rotor was initially installed the distillate rate increased to about 1.5 L/min, before subsequently decreasing back to an average of 1.0 L/min. However, the present distillate rate of 1.0 L/min can be maintained at an average product temperature of about 135oC over the duration of a 14 hour run.

Decontamination Factor of Radionuclides Treated by TFE-1

The removal efficiency of a radionuclide or other chemical contaminant in an evaporator is usually expressed as a decontamination factor (DF), which is defined by Eq. (2).

Eq. 2

Figure 2 shows the evaporator performance for the removal of the most abundant isotopes present in the feed stream to the evaporator. The data were compiled from about two years and more than 50 operational runs. The first bar of each histogram represents the total radioactivity of the specified radionuclide in the feed stream in curies. The second bar is the total radioactivity of the radionuclide in the distillate stream, also in curies. Finally the line plot, which refers to the right ordinate, is the decontamination factor for each radionuclide.
Fig. 2
The first set of two bars shows the evaporator performance for the removal of 60Co. Over an operating period of about 500 days, about 10 Ci of 60Co was introduced into the solidification plant. Of this, less than 0.002 Ci was released, representing a DF of about 6000 in the thin-film evaporator. If the fraction of suspended particles to the evaporator for a given batch of waste is high it is probable that a large removal of 60Co will be achieved, since a large fraction of this radionuclide is adsorbed to the surfaces of suspended particles.
The most problematic radionuclides from a decontamination perspective are 137Cs and 134Cs. About 80% of the cesium is present as 137Cs, but both isotopes are equally difficult to remove upstream of the evaporators in the membrane plant, due to their low valence. The cesium isotopes are rejected at about 98% by the membrane plant, compared with 99.5% for the other b/g emitters (6). The bulk of the cesium is soluble and is not retained by the suspended solids in the backwash concentrate stream. The decontamination factors for 137Cs and 134Cs are 4000 and 800 respectively.
The decontamination factors for the two cerium isotopes (141Ce and 144Ce) are 105 and 2 x 104 respectively. The large DF for these two isotopes is partially attributed to the fact that the bulk of the cerium is adsorbed on suspended solids removed by the MF system. Since the retention of suspended solids in the evaporators is very high, it follows that there is very efficient removal of the cerium isotopes.
The lower DF for the two cesium isotopes may be related to the fraction of total cesium that is dissolved. By comparison with 141Ce and 144Ce, where the decontamination factors are greater than 20 000, and 60Co (DF 6000), most of the cesium is dissolved. About 60% of the cobalt isotopes and 75% of the cerium isotopes are associated with the suspended solid phase. These adsorbed radionuclides are subsequently removed by the MF membranes as backwash concentrate. The results suggest that the removal efficiency of the evaporators for radioactivity decreases when the dissolved fraction of the isotope increases. It is known from operating experience at CRL that, of the radionuclides discussed here, the cerium isotopes are the least soluble, and the cesium isotopes are the most soluble.
Organic Carryover in Evaporator Distillate
Although the effluent quality from the evaporator meets all discharge criteria for radioactivity, there is some carryover of organics in the distillate. The source of these organics is primarily the distillation of lighter components in the emulsified bitumen in TFE-1, which is used for immobilization of the mixed aqueous waste concentrate. The concentration of dissolved organic carbon (DOC) in the distillate can vary between 1 and 300 mg/L, while the concentration of phenolics in the distillate stream ranges from 0.02 to 2 mg/L. The oil and grease concentration in

the distillate stream (T-13) is usually less than the 15 mg/L Federal Discharge target (2), but has exceeded 100 mg/L on occasion.

It has been observed that a titanium dioxide photocatalytic reactor is capable of reducing both EPA 624 (volatile) and EPA 625 (extractable) priority contaminants present in the evaporator distillate to below the method detection limits of the GC/MS analytical equipment (7). Dissolved oxygen was sufficient to remove the color associated with organics from the wastewater, even though not all of the organic carbon was removed. A concentration of 500 mg/L of hydrogen peroxide was found to be sufficient for the removal of the organics present in the evaporator distillate stream. Phenolics were effectively reduced to well below the Canadian Federal Discharge limit of 20 mg/L with the technology, and oil and grease was reduced to below the 15 mg/L imposed guideline for Federal Establishments (2). Other aromatic compounds, including naphthalene and methyl-substituted naphthalene derivatives, were removed or converted to other less toxic constituents. The dissolved organic carbon was not all converted to carbon dioxide in the catalytic reactor; some intermediate oxidation products were formed, which included organic acids.

Product Drum Radiation Field

The radiation field on the product drums is a function of the concentrations and distributions of the specific b/g emitters immobilized in it. The contact field on a product drum varied between 100 and 1000 mR/h for the first 100 drums produced at the WTC, during which time the weight per cent solids in the product drum was maintained below 25%. Thereafter, with more radioactive waste introduced into the WTC, and by maintaining the weight per cent solids at about 35%, the average contact radiation field increased significantly. There have been two runs to date where 5 R/h (contact) drums were produced. The new rotor operating at the higher speed of 900 rpm has increased the volume reduction potential of the evaporator (Table II). This is because there is better heat transfer through the vessel wall.

Fig. 3

A semi-empirical correlation was employed which allows the radiation field to be computed if the concentration of gamma emitters in the feed tank is known. Figure 3 shows the correlation between the radiation field (as measured on contact on the side of the drum), and the estimated radiation field. The correlation is accurate to within 7% on average, assuming that the radiation field on contact with the drum exceeds 500 mR/h. The correlation is accurate enough to allow fine tuning of the liquid and bitumen emulsion flowrates. This ensures that the radiation field on any drum does not exceed about 3 R/h on contact.

The estimated radiation dose to the bitumen was computed over its lifetime. The gross beta/gamma loading per kg of product varied between 5 x 107 and 5 x 108 Bq/kg. This resulted in an estimated dose to the bitumen ranging between 1 x 106 to 1 x 108 rads over its lifetime. This was well within the maximum allowable dose of about 5 x 108 rads per drum (4).

Fig. 4

The estimated dose to the bitumen (from the known beta/gamma emitters) for the 40 product drums generated in 1994, is shown in Fig. 4 on the left ordinate. A large fraction of the dose to the bitumen is due to Co-60 radioactivity (shown on the right ordinate in mCi/drum).

Radiological and Chemical Characteristics of Product Drum

The total inventory of radioactivity and salt loading immobilized in the 40 product drums was computed. These totals were divided by the total mass of immobilized secondary bitumen waste produced from the WTC in 1994. Table III shows the distribution of radiological and chemical contaminants expressed in the units of mCi/mL (of immobilized product) and mg/kg (of immobilized product), respectively. Of the b/g emitters immobilized in the product drums, the most abundant were: Ce-144 (0.91 mCi/mL), Co-60 (0.44 mCi/mL), Cs-137 (0.90 mCi/mL), and Cs-134 (0.94 mCi/mL). On average, there was a total of 0.023 mCi/mL of gross a present, about half of which could be accounted for by the two radionuclides: Am-241 (0.0072 mCi/mL) and Pu-239 (0.0033 mCi/mL). The gross a accounts for about 1% of the total radioactivity in the product drum.

Table III

Five chemicals have been identified as contributing towards the majority of the reactive solids loading in the product drum. These include the sodium-based salts of nitrate (92 g/kg), phosphate (36.5 g/kg), sulphate (18.4 g/kg), chloride (11.2 g/kg), and carbonate (10.1 g/kg). Concentrations of sodium, iron, and calcium are also given in Table III.

LEACH RESISTANCE OF EMULSIFIED BITUMEN WASTEFORM

The emulsified bitumen wasteform from the WTC was evaluated with the ANS/ANSI 16.1 test. In this standard procedure the wasteform is immersed under water for an extended period of time and the leachate is periodically removed and chemically analyzed. The test protocol is aggressive and not indicative of disposal conditions. The test is meant to be used as a relative indication of product quality.

Leach resistance of a wasteform can be expressed in terms of a leachability index (LI) as follows:

Eq. 3

The units of LI are dimensionless, while those of D (diffusivity) are cm2/s. A high value of LI implies a slow leaching rate (and a low effective diffusivity). A minimum LI of six is required by the USNRC, and a value of 10 is considered to be excellent.

The leachability index using the ANS/ANSI 16.1 standard test varied between 7 and 9 for the various salts present in the WTC wasteform. Subsequent tests have shown that the LI of the emulsified bitumen wasteform is relatively independent of the processing conditions used at the WTC. Step changes in process temperature did not have any significant effect on the results obtained.

To determine if the leach resistance of the emulsified bitumen wasteform product from the WTC could be improved, a series of experiments have been carried out using a simulated waste (prepared in the laboratory), immobilized in emulsified bitumen. The bitumen wasteform created in this fashion was itself further encapsulated in other binding agents including oxidized bitumen (used for liquid waste immobilization at the Surrey Radwaste Facility in Virginia), and ordinary Portland cement. Both encapsulating materials surrounding the bitumen core were evaluated at three wall thicknesses: 0.25 cm, 0.50 cm, and 0.75 cm. Each sample, prior to leach testing, had a diameter of 3.5 cm and a height of 7.0 cm. This sample size was maintained to provide the same surface area to volume for all encapsulated and unencapsulated samples. In addition, three different salt loadings were investigated. Only results for the maximum loading are reported in this paper.

Fig. 5

Some of the data is shown in Fig. 5, where the leachability index (from the ANS/ANSI 16.1 test) is plotted for the uncoated emulsified bitumen wasteform (zero wall thickness), and at the other three wall thicknesses. The data are plotted for both the cement and oxidized bitumen encapsulations of the emulsified bitumen wasteform. Phosphate is among the most abundant chemicals present in the WTC emulsified bitumen wasteform (Table III). The leachability index is shown as a function of the wall thickness for both the cement (open symbols) and the oxidized bitumen (solid symbols) encapsulations. For chloride with cement encapsulation, the LI increases from about 8.6 for the pure emulsified bitumen product (without additional protective coatings), to about 11.5 at a cement wall thickness of 0.75 cm. By comparison, for the oxidized bitumen overcoat, the LI for chloride increases from 8.6 (with no overcoat), to 13 for an oxidized bitumen wall thickness of 0.25 cm. No further increase of LI with a wall thickness greater than 0.25 cm is observed.

For phosphate, the LI increases from about 9.2 to 12.5 (at a wall thickness of 0.25 cm) for both the cement and the oxidized bitumen protective overcoats (Fig. 5). There does not seem to be much improvement of the LI for wall thicknesses greater than 0.25 cm, for either the cement or the oxidized bitumen encapsulations.

Fig. 6

Figure 6 shows the leach resistance of the two encapsulation matrices for cesium and strontium. The leachability index for the emulsified bitumen wasteform (without protective overcoats) is about 8.4, and increases to about 9 with coatings of cement or oxidized bitumen. For strontium the LI without any overcoat is about 11. For the cement overcoat the LI increases to about 11.8 at a wall thickness of 0.25 cm, but shows no further increase with higher wall thicknesses. This could be due to the fact that natural strontium was also being leached from the cement matrix itself, which might bias the results. For the oxidized bitumen overcoat, the LI increases from 11 (for emulsified bitumen wasteform), to about 15 for a wall thickness of 0.25 cm.

The results from Figs. 5 and 6 show that the addition of protective overcoats of either cement or oxidized bitumen can significantly improve the leach resistance of the emulsified bitumen wasteform. In general, the oxidized bitumen encapsulation was superior to the cement encapsulation for most of the chemical and radiological contaminants that were evaluated. The cesium leachability was about the same for both encapsulation materials at all wall thicknesses.

CONCLUSIONS

Product drums from the thin-film evaporator facility at the CRL Waste Treatment Centre have an on-contact radiation field of between 500 mR/h and 3000 mR/h, depending upon the radiological and chemical characteristics of the feed. The dominant b/g emitters in the product drum include 137Cs, 134Cs, 144Ce, 141Ce, and 60Co. On average, there is a gross / loading of 2.6 mCi/mL in the immobilized bitumen product. The concentration of a emitters constitutes about 1% of the total radioactivity, and the principal isotope is 241Am. The radiation dose to the bitumen varies between 106 and 108 rads per product drum (over its lifetime), which is less than 20% of the maximum allowable dose for the matrix.

The decontamination factor of the thin-film evaporator varies with the radionuclide. It is about 20 000 for 144Ce, 6000 for 60Co, and about 4000

for 137Cs. Those radionuclides more strongly adsorbed to the inert solids associated with the MF backwash solids are more efficiently removed in the evaporator.

Installation of a new rotor, and increasing its speed from 600 rpm to 900 rpm, has significantly improved the operation of the thin-film evaporator. The cleaning frequency of the unit has been reduced by a factor of three, and the solids loading has been boosted from about 25% to 35%. The volume reduction factor increased from 2.67 to 3.21 after the installation of the new rotor operating at 900 rpm.

There is some distillation of the organic fraction from the emulsified bitumen used for immobilization of the mixed waste concentrate. Photocatalytic oxidation can effectively remove the organics volatilized in the distillate.

The addition of protective overcoats of cement or oxidized bitumen can significantly improve the leach resistance of the emulsified bitumen wasteform. Based on the ANS/ANSI 16.1 test it was shown that the diffusivity of many of the chemical and radiological species can be reduced by up to three orders of magnitude with protective layers of these encapsulating agents. The oxidized bitumen matrix was superior to the cement encapsulation of the emulsified bitumen wasteforms.

ACKNOWLEDGEMENTS

Partial funding for the experimental program was provided through the CANDU Owners Group Low-Level Radioactive Waste Working Party 49. The authors express their gratitude to process operators in the Waste Treatment Centre for their efforts in operating the equipment and collecting data.

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38-20

DEVELOPMENT OF THE BORIC ACID RECYCLE SYSTEM BY REVERSE OSMOSIS

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ABSTRACT
MHI has developed the Boric Acid Recycle System which can recycle the boric acid in the liquid wastes by using reverse osmosis and then can reduce the volume of concentrated liquid wastes. This system can enable to reduce the capacity and cost of the volume reduction/solidification system of the concentrated liquid wastes.
The current PWRs in Japan use the evaporators to process the liquid wastes which contain the boric acid and use the large capacity solidification system to reduce the volume and to solidify the concentrated liquid wastes. The reverse osmosis in the Boric Acid Recycle System can recycle 85% of the boric acid in the liquid wastes. Then this system can reduce the volume of the concentrated wastes by the factor of about 7.

PURPOSE
To reduce produced amount of waste, various types of volume reduction equipment have been introduced and great results have been obtained so far. However, there is a tendency that the scale of the facility is getting larger. Here is a study which simplifies the treatment facilities not by reducing the produced volume of waste but by reducing the amount produced at the production source.
Currently, the number of drums produced by concentrating and solidifying waste liquid from PWR type nuclear power plants depends on the boron content contained in the waste liquid. Thus, simplification of the volume reduction equipment, solidification equipment, etc., are performed by recovering only boron in advance and effectively concentrating the waste liquid.
[Purpose] Reduction of waste by boron recovery
Simplification of treatment facilities
Fig. 1
PRINCIPLE
When waste liquid (drain) containing boron, radionuclides, etc. is supplied to the high pressure side of reversible osmosis membrane (RO membrane), the boron moves to the recovered liquid side by permeating through pores of the RO membrane, as boron exists in molecular status and its molecule diameter is small.
On the other hand, since radioactive materials (Co, Cs, etc.) and sodium (Na) exist as hydrates after ionization and they cannot permeate through pores of RO membrane as the diameters of hydrates become larger, they remain on the waste liquid side.
As a result, the boron can be recovered by separating it from the other materials.
Fig. 2
FACILITY
Figure 3 shows the relation between the PWR plant and the boron recovery system. As universally known, the boron is circulated in the cycle of Boric Acid Tank (BAT), RCS Coolant Holdup Tank (HT), Boric Acid Evaporator (EVAP) and is reused, and a part of it is contained in the drains from various sections of the plant and collected in the waste liquid holdup tank (WHT) of the WDS. The boron recovery system recovers

selectively only the boron from the liquid in this WHT, sends it to CVCS to reuse the boron, and, simultaneously, this system reduces the boron amount in the waste liquid, thus reducing the produced amount of the waste.

Fig. 3

Figure 4 shows the abstract system configuration of the boron recovery system. After removal of the impurities in particle state of the waste liquid from WHT by the pretreatment filter, the waste liquid is transferred by pressure to the reversible osmosis membrane (RO). The borated water which has permeated through the first stage RO is transferred to CVCS and reused.

Though the boric acid is contained in the liquid on the non-permeable side, the concentration of impurities rises. For this reason, the impurity concentration in the recovered borated water rises also, if the boric acid is recovered from this liquid. To avoid this, the borated water which impurity concentration is reduced is returned to the circulation tank, and is transferred by pressure to the first stage RO again, and in this way, the improvement of boric acid recovery rate is attained.

On the other hand, the waste liquid on the non-permeable side of the third stage RO, which impurities are concentrated, is cement-solidified after it is concentrated up to the concentration limit by the evaporator.

Fig. 4

Figure 5 shows an example of the boric acid recovery features of the system shown in Fig. 4. Only the boron can selectively permeate through the RO membrane, and a little impurities (SiO_2, etc.) can permeate through it. Therefore, the limit appears to the recovery rate of boric acid, by the limitation of the impurity concentration in the recovered boric acid.

Fig. 5

In the example of Fig. 5, the recovery rate of boric acid is approximately 90%, if the concentration limit of SiO_2 is 0.5 ppm. If the recovery rate of boric acid is 90%, the produced amount of waste becomes 1/10 in comparison with the case without boron recovery system, as the remaining 10% of boron is contained in the waste liquid side. Similarly, if the boron recovery rate is 80%, the waste amount becomes 1/5.

EFFECT

The comparison of facility scale (number of components and installation space), number of produced drums and maintainability was made between the following two methods:

Case A . . . Conventional Method

Concentration + Volume reduction + Solidification (EVAP) (cementation equipment with high volume reduction rate)

Case B . . . Method combining boron recovery and simple solidifying equipment (vacuum injection type cementation)

Boron recovery + Solidification

(RO) (vacuum injection type cementation)

The following is a summary of comparisons between both cases :

1. It can be said that the effect of facility simplification of case B is rather high, though the number of drums is slightly more.

2. While case A has lot of high temperature components and, further, consists of components for powder handling, case B has less high temperature components and consists only of liquid system processes.

Therefore, it can be said that case B has excellent operability and maintainability.
As above-mentioned, it is supposed that the introduction of boron recovery may contribute to the rationalization of the total system including the range from the treatment in power plant.
The following is the features of boron recovery:

- Reduction of produced amount, by tracing back to upstream
- Simple facility configuration /reduction of installation space
- Easy operation and maintenance
- Less dynamic component
- No high temperature component
- No powder handling component

DEVELOPMENT SCHEDULE
The development of boron recovery system has a long history, and this system was developed using tubular type modules which are of slightly old type RO, in 1977 and 1978, and even the site test was carried out at that time.
After that, the RO technology made great progress, and the spiral type RO module has become mainstream, which contributes to planning of size down and reliability enhancement of the device.
During the period from 1984 up to now, the boron recovery were actually carried out 13 times in total for the waste liquid from decontamination of S/G water chamber as an object, and excellent results were obtained every time. Thus, the reliability of this technology has been confirmed and, at the same time, the technology has been accumulated.
Based on this experience, the system study and various kinds of test have been conducted for the introduction of the full-scale boron recovery system since 1990 and a positive prospect for the practical use was obtained.
In 1992, the tests of long term durability was performed. Now, it is planned to apply to the Japanese APWR.

38-21
PRACTICAL USE OF RADIOACTIVE SPENT ION EXCHANGE RESIN TREATMENT SYSTEM
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ABSTRACT

In Japanese PWR plants, radioactive spent ion exchange resins have been stored in tanks in their sites, since they have a relatively high level radioactivity and there had been no way to treat appropriately for the final disposal. Therefore, an effective treatment technique had been desired, which converted the spent resin into a stable form suitable for the final disposal.

We have performed many research and development works on the spent resin treatment technique, and finally we have successfully developed a new type of spent resin treatment system which is suitable for the final disposal.

The system removes radioactivity by eluting radioactive substances from resins.

H_2SO_4 is used for an eluant. The eluant is recovered by diffusion dialyzer for reusing. The resultant inactivated resins are incinerated. The ashes and eluted radioactive substances are solidified with inorganic matrix.

In 1994, the system has been installed in the Ohi Nuclear Power Plant No. 1, 2 of the Kansai Electric Power Co., Inc. Further the system is planned to apply to some other plants.

DEVELOPMENT OF RADIOACTIVE SPENT ION EXCHANGE RESIN TREATMENT SYSTEM

System Concept

The basic system concept is illustrated in Fig. 1. The elution process is the main part of the system where radionuclides are removed from the spent resins by the use of $2N\text{-}H_2SO_4$ as an eluting solution(this is termed eluant). The resulting low activity resins are incinerated to produce the inorganic ashes, while the effluents containing the eluted radionuclides are led to the diffusion dialyzer. The diffusion dialyzer recovers more than 90% of H_2SO_4 as eluant. The unrecovered eluants containing a major part of the eluted radionuclides are solidified by cementation with the inorganic material. Thus, the organic spent resins are converted to the inorganic forms suitable for the final disposal.

Fig. 1

Major Nuclides In Spent Resin

Knowledge of the type and quantity of radionuclides absorbed in spent resins is indispensable to develop the spent resin treatment system. Spent resins stored in spent resin storage tanks over a period of 4 to 8 years were sampled and analyzed to determine the constituents of the absorbed radionuclides. The result is that the major constituents are Co-60, Ni-63 and Cs-137 as shown in Fig. 2.

Fig. 2

Elution

Ion exchange resins used in Japanese PWR plants consist of cation exchange resins with sulfonic acid radicals as substituents and anion exchange resins with ammonium radicals as substituents. The major radionuclides absorbed in spent resins are cations such as Co^{2+}, Ni^{+} and Cs^{+}. These cations are absorbed in cation exchange resins.

The cations in the resins can be replaced with hydrogen ions in contact with the sulfuric acid solution as follows:

$$R\text{-}(SO_3^-)_2Co^{2+} + 2H^+ \quad R\text{-}(SO_3^-)_2(H^+)_2 + Co^{2+}$$

This process is termed elution.

Elution characteristics of Co and Cs were tested in relation to the concentration of the sulfuric acid solution. The result was that $2N\text{-}H_2SO_4$ could elute more than 99.9% of Co and 99% of Cs by passing through the resins with about a 20 times volume of the resins as shown in Fig. 3.

Fig. 3
Recover Of Eluant
The concept of the recover of the eluant with the diffusion dialyzer is shown as Fig. 4. The diffusion dialyzer is provided with an anion exchange membrane and is divided into two chambers by the membrane. The eluant with the eluted radionuclides is introduced to the left chamber in the upward flow and the pure water is fed to the right chamber in the downward flow. While the eluant is passing trough the chamber, SO42- and H+ move from the left chamber to the right chamber through the membrane. However, major radionuclides such as Co2+, Ni+ and Cs+,which are cations, flow out of the chamber without moving to the right chamber due to the characteristics of the anion exchange membrane. Accordingly, the eluant(sulfuric acid) is separated from the radionuclides.
Fig. 4
The test was done to examine the characteristics of the diffusion dialyzer with 2N-H2SO4 containing Cs and Co. The relationship between the eluant recovery rate and the radionuclides leakage rate is shown in Fig. 5. The eluant recovery rate has a relation to the area of the membrane. We have decided that the design eluant recovery rate is 90%.
Fig. 5
Demonstration
A pilot plant was constructed and tested to demonstrate the system performance. The plant has a capacity of 0.05 m resins per batch which is a fifth scale capacity of the practical plant. A schematic flow diagram of the pilot plant is shown in Fig. 6.
Fig. 6
In the test, a batch of simulated spent resins were transferred from the spent resin receiving tank to the measuring tank by a slurry pump until an ultrasonic level sensor detected the predetermined level. The elution column received the resins from the measuring tank by gravity ,and dumped them to the eluted resin receiving tank after the elution. All operations including handling of resin and conditioning of eluant were performed automatically.
The radionuclide concentration at the outlet of the diffusion dialyzer is shown in Fig. 7.
Fig. 7
PRACTICAL USE OF SPENT RESIN TREATMENT SYSTEM
The developed spent resin treatment system has been applied to the Ohi Nuclear Power Plant No. 1/2 of the Kansai Electric Power Co., Inc. in 1994 at first in Japan.
The system flow diagram is shown in Fig. 8.
Fig. 8
Design Base
a. Processing design capacity
The processing design capacity is 6 m3/year, which is over the averaged generation rate in the Ohi Nuclear Power Plant No. 1/2. The operation way of the system is batchwise manner and one batch is to be processed in one week. One batch processing capacity is 0.25 m3 so that the system operational availability is below 50%.
b. Design decontamination The design decontamination (DF) is 1000 for 60Co and 100 for 137Cs under the design activity level of each nuclides as bellows.
60Co : 3.7 X 1013/ 6 m3
137Cs : 1.8 X 1013/ 6 m3

The above activity level is based on the analysis of the sampled spent resin activity.

Design Features

a. The existing incinerator and solidification system can be used.
b. The system is operated automatically.
c. The system scale is compact. (easy to apply to the existing plant.)
d. The amount of secondary waste is minimized.
e. The running cost is low.

CONCLUSIONS

The spent resin treatment system employing a radionuclide elution technique coupled with an eluant recovery by diffusion dialysis has proved adequate for practical application.

This system is suitable for the final disposal, since the organic spent resins can be converted to the inorganic materials.

38-23

PROCESS FOR TREATMENT OF INTERMEDIATE LEVEL RADIOACTIVE WASTE BASED ON RADIO-NUCLIDE SEPARATION

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ABSTRACT

During reprocessing of spent nuclear fuel, intermediate level liquid waste (ILW) along with other type of wastes like high level waste (HLW), zircalloy hulls etc. are generated. ILW streams are basically concentrated condensates and are alkaline on account of neutralization step prior to storage in carbon steel tanks . The specific radioactivity varies from 2 to 100 mCi/l due to Cs137/Sr90 besides traces of Ru106. The waste also contains unrecovered traces of uranium and other actinides. A new process has been developed which employs alkali precipitation for removal of uranium and other actinides, ion exchange for Cs137/Sr90 removal followed by treatment for decontamination with respect to Ru106. The ion exchange matrix employed is resorcinol formaldehyde (RF) condensation polymer with imino-diacetic acid chelating group. This resin has been synthesized and studied in detail at BARC laboratories in India for its application for treatment of ILW. The resin is found to be highly selective for both Cs137/Sr90 even in presence of high concentration of salts (about 25% $NaNO_3$) under alkaline pH condition.

In this process, as a first step, the uranium and other actinides are precipitated after carbonate elimination. The carbonates are present in the waste since these are used for neutralization before the storage. The supernatant is then passed through the RF resin column for cesium and strontium removal. The effluent from the resin column which essentially contains Ru106 and trace amount of Cs137/Sr90 is treated further by effluent polishing. Ru106 is removed by using zinc activated charcoal

system after pH adjustment, whereas Cs137/Sr90 are removed by standard chemical treatment processes involving precipitation by barium sulfate and copper ferrocyanide, before discharge.

The salient features of this process are as follows:

1) During precipitation step, as the quantities of uranium and actinides are small, sludge is not withdrawn after each batch of operation. The sludge concentration is allowed to be built up leading to more effective decontamination and is withdrawn only after 12-15 batch precipitation operations. This concept of multiple precipitation has helped significantly in low volume of sludge generation.

2) The loaded RF resin columns are regenerated using dilute nitric acid and, in this process, Cs137/Sr90 get stripped. The regenerant solution is concentrated and passed through an ammonium-molybdo-phosphate (AMP) capsule for cesium recovery which can be used in medical and irradiation applications.

3) The resin is conditioned with alkali and is used in subsequent operations. Hence, the total amount of radioactive solid waste is very significantly reduced and an overall volume reduction factor of about 40 is achieved taking into consideration the volume of conditioned RF resin column and conditioned chemical sludge in cement matrix.

An industrial scale plant based on above process is in operation at Waste Immobilization Plant (WIP), Tarapur, India. The main ILW treatment plant consists of 1) a process cell housing process equipment and vessels, liquid sampling system and various remotization gadgets and viewing aids; 2) vessel off-gas system for treatment of process off-gases by scrubbing and filtration; and 3) chemical additive and services systems. Various processing steps and remote maintenance works are done using master slave manipulators, in- cell crane and pneumatic wrenches with the help of viewing aids like shielding windows, mirrors and closed circuit television.

The successful treatment of alkaline radioactive liquid waste on industrial scale has demonstrated the usefulness of this process. The process not only helps in separation of Cs137 and Sr90 radionuclides but also helps in decontamination with respect to Ru106.

INTRODUCTION

Intermediate level liquid waste (ILW) is generated during reprocessing of spent fuel as a second cycle raffinate, evaporator condensate and during concentration of low level waste (LLW) from nuclear power plants (1). As a practice, this category of waste is being neutralized using alkali for storage in carbon steel tanks. Sodium carbonate is used for neutralization to keep the traces of uranium and actinides, if present, in dissolved form. The specific activity of ILW ranges from 2 - 100 millicurries per liter which is predominantly due to Cs137/Sr90 with traces of Ru106. The treatment of ILW involves use of various matrices like cement, polymer or bitumen. Among these, cement is the most cost effective matrix. Polymer based processes are comparatively costlier though the product has higher leach resistance. Bitumen is another matrix used for immobilization of ILW. It is low in cost and processing temperatures are moderate. In the process developed by BARC, India, emphasis is on separation of Pu, U, Cs and Sr resulting in high volume reduction factors. In this process, the uranium and actinides are first removed by alkali precipitation after carbonate elimination. Subsequently, for removal of Cs137/Sr90, the supernatant is passed through an ion exchange column filled with resorcinol formaldehyde

condensation polymer having imino-diacetic acid chelating type of side chain (2). The effluent from the IX column, which essentially contains Ru106 and traces of Cs137/Sr90, is further polished by passing through second set of RF resin columns. Ru106 is removed using zinc-activated charcoal system after pH adjustment (3), whereas Cs137/Sr90 are removed by standard chemical treatment using barium sulfate and copper ferrocyanide before discharge.

PROCESSING SCHEMATIC

The process is broadly divided into three stages which are preconditioning, ion exchange processing, post treatment and conditioning. The other steps involved are, disposal of spent IX columns and separation of Cs137 from regenerant solution. The detailed engineering flow diagram is given in Fig. 1.

Fig. 1

Pre-conditioning

The pre-conditioning of waste in this process involves carbonate elimination and alkali precipitation. The details are given below:

Carbonate Elimination

Intermediate level wastes generally contain traces of unrecovered uranium and actinides. In order to keep these in dissolved state, neutralization (pH-9) is carried out with sodium carbonate. In order to obtain good decontamination factors (DFs) with respect to alpha, carbonates are eliminated by acidification using nitric acid.

Precipitation/ Sludge Separation

After carbonate elimination, ILW is transferred to the precipitation tank, (Tk-10). This is a conical bottomed tank with 3 drain lines located at different elevation for draining of supernatant. Sodium hydroxide is added to the waste and the resultant precipitate is allowed to settle for a period of 8 to 10 hours and drained. Once the clear supernatant is obtained, it is passed through the RF resin column.

Ion-exchange Processing

The supernatant solution is passed through the RF resin column from top to bottom. The effluent coming out of the column is monitored for specific activity. Once the specific activity of LLW reaches a pre-set value, the next column is taken on line and the loaded column is shifted to a location where it is regenerated.

Post Treatment and Conditioning

Zinc-activated charcoal (ZAC) treatment process (3) is found to be effective for ruthenium removal. The LLW is acidified to a pH of 2 and passed through the ZAC column. Finally, the waste is subjected to chemical treatment process involving precipitation with barium sulfate and copper ferrocyanide.

RESORCINOL FORMALDEHYDE BASED CHELATING RESIN

The synthesis, characterization and successful testing of a resorcinol-formaldehyde polycondensate resin for selective removal of radiocesium from alkaline salt-loaded waste solutions has already been reported (2,4). These studies also established the chemical, thermal and radiation stability of the resin. The resin contains phenolic -OH groups which ionize under alkaline conditions and serve as cation exchange sites that have high selectivity for cesium even in the presence of large concentration of sodium ions. Incorporation of imino-diacetic acid functional groups into a resorcinol-formaldehyde polymer matrix has been reported to give it the additional capability of strontium sorption by chelating mechanism (5). Based on these earlier developments,

investigations were carried out which finally resulted in the successful bulk synthesis of a suitable resin containing cesium-selective phenolic - OH and strontium-selective imino-diacetic acid groups in the required proportion. Laboratory and bench-scale tests with this resin using actual waste solutions gave encouraging results (6). This was followed by engineering scale evaluation to study the phenomena of channeling and finalization of procedures for remotized handling of loaded IX columns and disposal aspects of spent IX columns.

OPERATIONAL EXPERIENCE

This process has been adopted on industrial scale for treatment of alkaline liquid wastes at Waste Immobilization Plant (WIP), Tarapur. The engineering flow diagram is shown in Fig. 1. All process equipments are installed inside a concrete shielded cell of about 4 M X 4 M X 10 M high with an in-cell crane, shielded viewing window, master slave manipulators and other remotization gadgets. The waste feed and other process solution transfer pumps as well as off-gas handling equipments are housed separately in areas having restricted access. Instrumentation is provided to monitor various process parameters like pressure, level, density, flow rate and temperature to facilitate process and inventory control.

The facility described above has been used for conditioning of different streams of ILW having specific activity in the range of 2-15 millicurie per liter of gross beta-gamma activity. The pH of the waste was in the range of 9 to 12.5 due to presence of - OH- and CO3-- ions. The salt content varied from 190 to 280 grams per liter which is essentially due to sodium nitrate. The salient features of the process operations are as follows:

Addition of dilute nitric acid was controlled in such a manner that excessive foaming and pressurization during carbonate elimination was completely avoided.

In place of withdrawal of sludge after each precipitation operation, withdrawal after multiple precipitation operation helped in reduction of over-all sludge generation. The average decontamination factors (DFs) with respect to alpha during precipitation was of the order of 250.

The processing rate through ion-exchange column ranged from 3 to 5 bed volumes per hour (BV/H) for acceptable column performance. The DFs achieved for gross beta-gamma activity were in the range of 200 to 600. The volume of processed waste varied from 500 to 850 bed volumes between two successive regenerations for 1% breakthrough.

The zinc-charcoal system was found effective for Ru106 removal giving DF in the range of 10-30.

The loaded ion-exchange resin was effectively regenerated using 25-30 bed volumes of dilute nitric acid with about 90% removal efficiency.

Ammonium molybdo phosphate supported on asbestos fibre was found to have good column characteristics and good efficiency for removal of Cs137 from acidic regenerant solution.

The spent ion-exchange columns after final wash are placed in carbon steel drums and immobilized by using cement concrete matrix prior to disposal as solid radio-active waste.

Various remotization gadgets have been used extensively for handling of ion-exchange columns, liquid sampling and disposal drums resulting in minimum exposure to O&M personnel.

As a result of safe and successful conditioning operations, over-all volume reduction factor of the order of 40 to 50 has been obtained. Due

to ambient temperature processing, the discharges through gaseous route are also very low.

CONCLUSION

This process is essentially based on separation of uranium and other actinides by chemical precipitation followed by Cs137/Sr90 separation by ion exchange method. Presence of uranium in waste helps in removal of actinides which are present in trace quantities. The RF resin, due to its high specificity for Cs and Sr in alkaline condition, was found to be effective on industrial scale for decontamination of waste containing high percentage of sodium ions. The separation of ruthenium from waste solution by zinc-activated charcoal was found to be effective. This process offers advantage of high volume reduction factor, high DF and low man-rem expenditure to O & M staff and is, therefore, very useful for conditioning waste streams of this nature.

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38-24

ADVANCED CEMENT MATERIALS FOR RADIOACTIVE WASTE MANAGEMENT

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ABSTRACT

Improvements in cement properties have been investigated from the viewpoint of better radioactive waste management. The basic concept for realization of the improvements is the addition of small amounts of admixtures to conventional cement; small amounts ensure the cost increase is minimized. Fundamental experiments first showed that carbon fiber reinforcement enhanced long term durability of waste forms because it increased tensile and flexural strengths of the hardened cement. Secondly, addition of natural zeolite and silica powder to cement improved sorption ability for Cs-137 and C-14, respectively. Furthermore,

lithium nitrate reduced hydrogen gas generation to 1/20 of its typical value during aluminum waste solidification, because the additive functioned as a corrosion inhibitor.

Application to actual waste management was also investigated based on these results. High performance cement allowed various low level radioactive wastes to be solidified into stable waste forms by one simple facility.

INTRODUCTION

Cementitious materials are widely used for solidification of low and intermediate level radioactive wastes and construction of waste disposal sites because these materials have long term durability and good sorption ability for radioactive nuclides, and they are low cost and can be handled easily (1). Improvements of the cement properties have been investigated from the viewpoint of better radioactive waste management. The basic concept for the improvements is the addition of small amounts of admixtures to Ordinary Portland cement (OPC), so as to minimize the cost increase.

This paper first describes some results of fundamental experiments. The points studied are:

1) fiber reinforcement to ensure long term durability of waste forms;
2) enhancement of sorption ability for Cs-137 and C-14; and
3) prevention of hydrogen gas generation during aluminum waste solidification.

Application to actual waste management is investigated based on the experimental results. A high performance cement solidification system for low level radioactive waste is presented and its features are discussed.

IMPROVEMENT OF CEMENT PROPERTIES

Fiber Reinforcement

Effect of carbon fiber: Mechanical strength of hardened cement is one of the most important parameters for long term durability. Tensile and flexural strengths of the hardened cement are relatively low, while compressive strength is quite high (Table I.) Fiber reinforcement is an effective method to increase the former two strengths. Various materials such as glass, steel and carbon can be used for the reinforcing fiber. However, glass fiber tends to dissolve and deteriorate in an alkaline atmosphere (pH13) such as present in the cement. Steel fiber is not easily dispersed uniformly in the cement mortar, because its density is high (~7g/cm3) and the fiber tends to sink downwards. Therefore, carbon fiber (18mm x 3mmL) was selected for study.

Table I indicates the effect of the carbon fiber. Tensile and flexural strengths increase in proportion to the fiber content and double by adding 3wt% fiber. Carbon fiber improves durability of the hardened cement, because resistance to cracking can be enhanced.

Application to solidification of spent ion exchange resin: Spent ion exchange resin comprises a major fraction of the low level radioactive wastes generated from nuclear power plants. The resin content in a cementitious waste form is typically controlled below 25kg-dry resin/200L, because the waste form tends to swell and crack under a water immersion condition when resin content is higher (1). The deterioration mechanism is as follows. When water penetrates into the waste form, spent resin swells by absorbing the water and tensile stress is exerted on the cement matrix around the resin particles (2,3). The waste form is cracked in water, if the tensile stress is higher than the tensile strength of the cement.

Water resistance of the waste form can be improved by use of the fiber reinforced cement, so as to raise the spent resin loading. Figure 1 shows typical results. The waste form with reinforcing carbon fiber does not deteriorate even at such a high resin content as 55kg-dry resin/200L, while the conventional cement without fiber cracks in water and its compressive strength decreases to almost 0kg/cm2 within a few days.
Table I
Fig.1
Enhancement of Sorption Ability for Radioactive Nuclides
Radioactive cesium: Hardened cement absorbs various kinds of radioactive nuclides, such as C-14, Co-60 and transuranic elements, and retards their release from the disposal site into the environment. However, the sorption ability for radiocesium (Cs-137) is low, so that potential Cs adsorbents were surveyed.
Although the highly alkaline environment and coexistence of calcium ions in the cement limited the choices of Cs adsorbent, a natural zeolite, whose main constituent is clinoptilolite, has good performance (4). Batch sorption experiments indicated that the distribution coefficient for Cs-137 increases from 2 to 120mL/g by adding 5wt% adsorbent to the cement.
Radioactive carbon: Hardened cement absorbs C-14 and its distribution coefficient is on the order of 1000mL/g. Enhancement of the sorption ability for C-14 is desirable to ensure long term safety after land burial because of its long half life (5730 years). While C-14 adsorbents were surveyed first, no suitable adsorbent could be found. The fundamental aspects of the C-14 sorption mechanism by the OPC were examined for next. The following results were obtained.
1) Carbon-14, in the form of CO32-, is adsorbed by positively charged sites on the cement particle. These sites were produced during the cement hydration process due to the reaction between CaO and SiO2 contained in the cement materials.
2) The OPC does not contain enough SiO2 compared with its CaO content, to produce sufficient numbers of C-14 adsorption sites.
3) Silica powder was added to the OPC and the mixture was hydrated.
Results of batch sorption experiments for the hardened cement are plotted in Fig. 2. The C-14 distribution coefficient increases from 2,000 to 7,000mL/g by adding 20wt% silica.
Fig. 2
Prevention of hydrogen gas generation from aluminum wastes
Selection of inhibitor: Miscellaneous solid wastes include aluminum-containing materials. They generate hydrogen gas during the cement solidification process, because the aluminum is corroded under a highly alkaline condition:
Eq. 1
Possible inhibitors were investigated to prevent this corrosion. While sodium silicate is well established as an inhibitor under an alkaline condition (5), it can not be used in the cement for the following reason. Calcium ions in the cement mortar react with the sodium silicate:
Eq. 2
Therefore, calcium silicate is precipitated and the sodium hydroxide produced accelerates the aluminum corrosion.
Galvanic current measurements and alkaline corrosion experiments were performed to find inhibitors effective in the cement (6). Soluble lithium compounds were selected, with lithium nitrate as the best. Figure 3 compares the hydrogen gas generation during the cement solidification

process. Addition of 3wt% lithium nitrate to the OPC reduces hydrogen gas generation to about 1/20 regardless of aluminum waste content.
Reaction mechanism of lithium nitrate: Microscopic observations and X-ray diffraction measurements show that the aluminum surface was coated by an thin Li-Al double salt film (~5mm, LiAl(OH)3Al(OH)4xH20) as is shown by the SEM photograph in Fig. 3. The double salt was produced by reaction between lithium ions and aluminate ions:
Eq. 3
This insoluble film stopped aluminum corrosion and hydrogen generation. While the function of the counter ion, namely nitrate ion, is not very clear, it probably prevents pitting corrosion.
Fig. 3
APPLICATION TO SOLIDIFICATION OF LOW LEVEL WASTES
A high performance cement (HP-cement) has been developed based on the fundamental experiments and a centralized cement solidification system was designed to solidify low-level radioactive wastes into stable waste forms. Figure 4 shows an outline of the system. It offers the following features.
1) Applicability to various wastes: Various kinds of wastes, concentrated liquid waste, spent ion exchange resin, incineration ash and miscellaneous solid waste, can be solidified into waste forms by one facility.
2) Stable waste form: The HP-cement contains carbon fiber, natural zeolite and lithium nitrate. These additives allow stable waste forms to be produced independently of the waste type.
3) Simple system: The HP-cement is supplied as a ready mixture, hence special equipment, such as a high-shear mixer, is unnecessary. That is, wastes are simply mixed with the HP-cement paste using a conventional mixer.
A full-scale pilot plant was constructed to demonstrate its industrial applicability. The treatment capacity was 2 drums/hour. Table II summarizes the test results. It was confirmed that all wastes can be solidified into stable waste forms.
Fig. 4
Table II
CONCLUSIONS
Improvements of the cement properties have been investigated from the viewpoint of better radioactive waste management. The following results were obtained.
1) Carbon fiber reinforcement is an effective method to increase tensile and flexural strengths of the hardened cement. This ensures long term durability of waste forms.
2) Addition of natural zeolite and silica powder to cement enhanced sorption ability for Cs-137 and C-14, respectively, so that their release from the disposal site into the environment could be retarded.
3) Aluminum wastes generate hydrogen gas during the cement solidification process. Addition of lithium nitrate reduced hydrogen generation to 1/20 of the usual value, because the additive functioned as a corrosion inhibitor.
4) Based on these fundamental results, a high performance cement was developed to solidify various low level wastes into stable waste forms by one simple facility. Its industrial applicability was demonstrated by full-scale pilot plant tests.
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38-25

INCINERATION OF HIGHLY CONTAMINATED GRAPHITE WASTE FOR KEEPING CLEAN ENVIRONMENT

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ABSTRACT

Flameless incineration under laminar gas flow, was used to dispose of highly radioactive graphite from a nuclear reactor. The radioactive ashes were collected on filters, while the carbon dioxide formed, went through a freezing-sublimation process to achieve a better separation of the CO2 from aerosols, which might be carried by the off gas of the incineration. Optimization of the incineration process was sought by studying the effect of gas flow speed, oxidation temperature and furnace rotation speed. Sublimation of the frozen CO2 reduces its activity by a factor of 108.

The following conclusions were drawn from the research. The graphite was ground to particles of size 22.5 microns and specific area of 15 m2/gr. Since the rate of oxidation was constant, a parameter K(hour-1) was defined, equals to the part of graphite incinerated in one hour. The dependence of K on oxidation temperature, gas flow speed and furnace rotation speed was investigated. The results are as follows : by changing the incineration temperature from 6000C to 7000C under air flow speed of 0.034 m/sec, the oxidation time reduced from 7.5 hours to 2.5 hours. At 7000C the graphite powder ignited. The furnace rotation speed did not have a strong effect on incineration time in air. When the incineration was done under pure oxygen, ignition started at 5600C. Changing pure oxygen flow rate from 0.025 m/sec to 0.051 m/sec at temperature of 5600C, reduced the incineration time by 20%. Changing the furnace rotation speed from 0 to 4.2 rad/sec (at 5400C) reduced the incineration time from 30 to 5 hours. At the end of the process, after freezing the off gas and the CO2 sublimation, the specific activity of the CO2 was less than 4 10-16 curie/liter

INTRODUCTION

An alternative to treatment of high level radioactive waste, with the objective of decreasing its volume, is storing it in its original form,

in depositories. Treatment becomes expedient when storing cost of the nonradioactive part of the waste, exceed the cost of its treatment. Such depositories had not been built yet, so that other methods should be developed. The present work describes the development of a technique of flameless burning of highly contaminated graphite which was used as a construction material in large, water-cooled, graphite reactors - LWGR and high temperature gas cooled reactors - HTGR.

In case of disruption of fuel rod cladding in LWGR and a flaw in the coolant vessel, contamination of the inner surface of the graphite mould will result, by fuel and fission products. The contaminated graphite layer turns "spongy" and a highly radioactive powder. The fuel and the radioactive fission product might constitutes 0.1 - 1%.of the graphite. HTGR fuel element cladding disruption releases fissile and fertile materials particles, and results in a more complicated contamination of the graphite shell. In regular oxidation of high level waste, filtration of the highly radioactive aerosols is complicated because of the necessity of using a large number of filters, which increases aerodynamic resistance of the purification system by the large mass of the filtering material, which later on undergoes treatment as a contaminated material. Reduction of aerosols quantity released to the atmosphere may be obtained by partial rebound of the off gas for further oxidation. (1).

Oxidation of highly radioactive waste may be carried out in concentrated sulfuric and nitric acids (2) or in molten salt (3). An interesting method of catalytical oxidation of contaminated reactor graphite is described in ref (4). It is an alternative to converting CO_2 into $CaCO_3$. A two - stage process is used to oxidize and reduce contaminated graphite, according to the reactions :

$$C + CO_2 = 2CO \quad (1)$$

$$2CO = 2C + O_2 \quad (2)$$

A partial oxidation of CO provides a source of CO_2. It should be noted, that treatment of graphite by this process results in forming a fine dispersion of carbon in form of aerosol.

THE FLAMELESS INCINERATION METHOD

However, the techniques mentioned above do not provide the possibility of reducing the volume of high level radioactive waste in form of contaminated graphite, without contamination of the environment, since these techniques include release of at least a part of the aerosols, to the atmosphere. This release is avoided if oxidation is carried out in a closed volume, as described in the following scheme :

1) Oxidation of waste.
2) cooling off the off gas.
3) Freezing the CO_2 from the off gas.
4) continuous oxygen feed and
5) return of the gas (without CO_2) to the beginning of the process for further oxidation of waste.

The principal technological stages are 1) and 3). It should be noted that freezing - sublimation process is widely used in purifying of substances in radiochemical technology. The effectiveness of such process is known to be 108 (5). The amount of continuously fed oxygen (4) is determined by its consumption in the carbon oxidation (1).

Accumulation of the isotope 14C in the active zone of LWGR during 20 years of operation is on the average 3.5×10^{-3} Curie per kg graphite. After oxidation, the concentration of the isotope 14C in the gas amounts to 1.9×10^{-6} Curie/l, while the permissible level is 1.2×10^{-10} Curie/l

(6). Therefore after sublimation of the solid carbon dioxide, the gas must be convert into solid, i.e. thermodynamicly stable substance (7).
A system was designed and built for graphite oxidation under controlled conditions, according to a scheme, in which very small quantity of graphite powder leaves the oxidation volume. The rate of the process and its safety correspond given conditions. The system is shown in Fig. 1. It consists of a tube furnace 1 containing a stainless steel reactor 2. An aluminia crucible was installed in the reactor, it contained the graphite powder 3. The reactor was rotated by an electric motor and a gear box 4, on bearing 5 and stationary plug 6. Furnace 1, reactor 2, bearings 5 and 6 were hitched on the chassis 7 which with the aid of supporting tube 8, was fixed at an angle of 150 to the horizontal table. The air flow through the oxidation zone was at a constant rate, it was controlled by a compressor 9 (when pure oxygen was tested, an oxygen cylinder was used), manostat 10 and a flow rate meter 11. Inlet 12 and outlet 13 were used, in the fluorplastic plug 6, which serves simultaneously, as a sliding bearing for the reactor 2. Reactor graphite containing (0.11 0.04)% ash was burnt in the system. It was crumbled to particles of average dimension of 22.5 microns and specific surface 15m2/gr. The purpose of using fine powder graphite was to get good simulation of dispersity of LWGR waste. Too fine graphite powder can expose "hot" spots in the graphite (4) and therefore increase the release of radioactive powder from the furnace.
Fig. 1
RESULTS AND DISCUSSION
The kinetics of the process was studied by the time required to complete the oxidation of a given quantity of powder, (tc hours). Since the rate of oxidation was constant in time, then instead of tc , a parameter K , hour-1 = 1/tc was used , (the relative part of reacted graphite in 1 hour). Figure 2 represents the dependence of oxidation rate K - on temperature T0C. In the interval 600 to 7000C, the rate of oxidation increases more then three times. The time of complete oxidation, tc, changed from 7.5 to 2.5 hours. The ignition of the graphite powder was seen at 7000C.
Fig. 2
Graphite powder oxidation in pure oxygen, at temperatures interval 500 to 6000C with a constant flow rate of 0.034 m/sec and reactor rotation speed, with graphite powder, of 2.09 rad/sec, showed a complete cycle of oxidation time change from 25 to 3 hours. At the interval 560-6000C, ignition of separate particles of graphite powder was seen. At a constant temperature of 5400C, increasing the oxygen flow rate from 0.025 to 0.051 m/sec caused 20% acceleration of oxidation rate only. At the same temperature, increasing reactor's rotation speed from 0 to 4.2 rad/sec, decrease the complete cycle of oxidation by more then 6 times from 30 to 5 hours.
In case of graphite powder oxidation in air at temperature of 6500C and flow rate of 0.042m/sec, changing the reactor rotation speed from 0 to 3 rad/sec, essentially did not affect the rate of oxidation, as seen in Fig. 3. It is concluded therefore that the rotation of the reactor is not necessary. That facilitates keeping the system hermetical and enhances its safety.
Fig. 3
Figure 4 shows the dependence of oxidation rate K(h-1) on air flow velocity v m/sec. At velocity interval from 0 to 0.04 m/sec,

reproductibility of the results sharply deteriorates. That indicates dependence on additional factors (might be turbulence). Based on the data obtained, the following conditions for oxidation are recommended : the temperature of the graphite should not exceed 6800C; the velocity of air flow should not exceed 0.04 m/sec; stirring the graphite powder by reactor rotation is not necessary. Under such conditions the complete oxidation cycle was of the order of 5 hours.

Fig. 4

For evaluation of the extent of graphite powder release from the furnace, tests were carried out at constant rotation speed of 2.09 rad/sec and at different air flow velocities (without heating). At flow velocity of 0.042 m/sec, the amount of powder carried by the air was 0.15% per hour. This parameter depends on nozzle's distance and angle, relative to the powder surface. This small amount of graphite must be oxidized before it continues to the system's next stage namely the freezer. For more effective oxidation in that case it is desirable to use a catalyst. Separating the high level radioactive impurities from material deposited on carbon (crystalline or amorphous) , was done as follows: 1 kg of waste of activity 1.5 Curie was crumbled and fed into a stationary reactor. It was oxidized at temperature of about 6500C with air flow velocity of 0.04 m/sec. The complete oxidation cycle under these conditions was 5 hours. The activity released from the reactor with the gas flow, in form of aerosols, was 1.5 Curie x 0.0015 h-1 x 5h = 1.125 x 10-2 Curie. The products of oxidation together with the aerosols were fed into a reheating furnace at the same temperature. The extent of aerosols release after reheating was 0.15% per hour, that is: 1.125 x 10-2 Curie x 0.0015 h-1 x 5h = 8.44 x 10-5 Curie. 1.87 x 103 l, of CO2 were formed, hence the concentration of radionuclides was 4.5 x 10-8 Curie/l. The effect of the sublimation on reduction of gas activity is a factor of 108, therefore the concentration of radionuclides in the sublimated carbon dioxide gas did not exceed 4.5 x 10-16 Curie/l. This value coincides with permissible concentration of a mixture of unknown radionuclides composition 4x10-16 Curie/l (6).

The Authors wish to thank Dr. V. Tokarevsky for his technical help during the experimental work.

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38-26

COST EFFECTIVE WASTE MINIMIZATION USING HEPA BAG VENTING FILTERS

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ABSTRACT

Nuclear waste volume reductions of 50-70% and more have been achieved by evacuating trapped air within PVC and polyethylene liner bags. The method is to vacuum trapped air out of plastic bags through a snap-tight high efficiency particulate air (HEPA) filter prior to placing the bag into the steel box, drum or other container. Estimates indicate that the mass of waste in each container may be doubled by removing the trapped air, thereby significantly reducing containerization, storage, transportation and disposal costs of low level, transuranic (TRU) mixed and asbestos waste. Nuclear waste volume reduction by evacuating trapped air through a high efficiency particulate air (HEPA) filter offers a cost savings of $45 to $65 per cubic foot of soft compressible waste with little additional cost or loss in productivity.

INTRODUCTION

Radioactive TRU and low level waste is placed in 8 mil, 55 gallon polyethylene bags, then placed in type B-25 containers or 55 gallon drums for storage and eventual disposal. Each bag is sealed before placing it into the container. Soft compressible low level waste such as cellulosic materials, gloves, towels etc., is stored in rigid fixed volume containers. The mass loading limit of containers that store soft compressible material is not typically reached. The limiting factor to the amount of material that is stored in the container is the restricted volume of the container. Volume reduction of the waste bags, then, allows a greater mass of waste to be stored in the container. Since storage and disposal cost are primarily dependent on volume, the challenge is to reduce volume safely and efficiently.

Air evacuation of PVC or polyethylene bags is accomplished safely because the trapped air is evacuated through a HEPA grade snap-tight filter. The geometry of the filter housing allows the nozzle of a vacuum hose to be snugly attached prior to evacuation, then easily removed once evacuation of resident air is achieved. Tape placed over the filter top once the vacuum hose is released prevents bag re-inflation.

The cost of poly-bag modification to include a HEPA grade snap-tight filter is about $7.00 per bag, material and labor included. Complete evacuation of a typical 55 gallon, 8 mil polyethylene or PVC bag is achieved in 2.5 to 5 minutes, therefore, throughput of waste processed bags is not compromised. The cost savings of doubling the mass of loading in waste boxes such as the B-25, by minimizing the volume of each bag of waste, may exceed $5000 to $6,000 per container. These savings are realized by reducing disposal costs alone; additional savings will be realized by reduced interim storage and transportation costs. Based on disposal cost of $70 per cubic foot, disposal of a 90 cubic foot B-25 container is $6,300 (1).

DESCRIPTION

Evacuating trapped air through the NucFil snap-tight HEPA filter greatly reduces the volume of stored wastes. Evacuation of air from waste bags is effective in TRU, low level, mixed, and asbestos waste minimization efforts.

Laboratory studies indicate that trapped air in storage bags accounts for as much as 50 to 75% of the total volume of stored waste. By eliminating

all possible air from each bag of TRU, low level, or asbestos waste, container loading may be increased by two to three times.
Shown below is simulated radioactive waste in a sealed 8 mil 55 gallon polyethylene bag. Notice the large volume of the bag.
Fig. 1
The simple low cost solution for evacuating trapped air from storage bags is demonstrated below. The technician simply applies a 0.5 to 3 horsepower vacuum pump to the flange extension of the NucFil snap-tight bag filter. All trapped air is safely removed within 2.5 to 5 minutes and the bag is compressed to 50% to 75% of its original volume. The NucFil snap-tight filter provides high efficiency particulate air (HEPA) grade particle removal efficiency when challenged with 0.3 to 0.7 micron DOP aerosol. This assures virtually no contamination or exposure health hazards to technicians involved in waste operations. The filter media utilizes patented carbon-bonded-carbon technology.
Fig. 2
The completely compressed bag of waste is easily deposited into a rigid storage container such as a B-25 steel waste box. Given the 90 cubic foot capacity of the B-25 container, and disposal cost estimated at $70 per cubic foot, cost savings of $5000 to $6000 per container are realized. Disposal of the waste that normally fills two B-25 containers costs $12,600. By compressing the volume of bags that normally fill two containers into one, the cost is reduced to $6,300, representing a savings of $6,300. Additional cost reductions are realized by reduced containerization, storage and transportation costs. At the Savannah River Site, engineers are presently testing 100 polyethylene bags configured with the NucFil snap-tight filter. Approval for use has been granted from SRS quality assurance and health departments.
Fig. 3
FIELD USE AT SAVANNAH RIVER SITE
Engineers at the Savannah River Site (SRS) have 100 8 mil, 55 gallon polyethylene bags that were modified with the NucFil-030 snap-tight polyethylene filter. The snap-tight filters are tested for aerosol removal efficiency and delivery capacity of 800 ml/min at 1 inch water column. SRS engineers conducted additional leak tests using Emery 3004 aerosol to verify filter and bag integrity.
About 20 bags of waste usually fit into a 90 cubic foot B-25 box. An objective of the study at Savannah River Site is to determine the increase in the number of bags that will fit into the B-25 container. It is anticipated, based on laboratory model studies, that about two to three times as many bags of waste will fit into the steel B-25 Box. Complete data will be available from the authors in Spring 1996
LONG TERM LIFE CYCLE COST SAVINGS
It is estimated that during the next 75 years, low level waste alone, from existing waste in storage, waste generated from future operations and waste generated from facility stabilization, nearly 647 million cubic feet of low level waste will be generated (3). Much of the low level waste in the DOE complex may be incinerated. Assuming that roughly 50% will not be incinerated leaves 323.5 million cubic feet of low level waste that must be disposed of in landfills. Of the 323.5 million cubic feet of non-incinerated waste, it is reasonable to assume that 50% is soft compressible waste accounting for 162 million cubic feet of low level waste that may be stored in air evacuated waste bags.

Under the present status quo, based on a disposal cost of $70.00 per cubic foot, and 90 cubic feet per container, total cost for disposal alone could approach $22 billion requiring 4 million B-25 boxes. Storage of toxic air will cost $15 billion and take up 226 million cubic feet of space. By evacuating the 50% to 70% of air trapped in bags, much volume and money will be saved for better uses. By evacuating trapped air from the 162 million cubic feet of non-incinerated, compressible low level waste, 80 million fewer cubic feet of landfill space will be required and about $10 billion will be saved.

CONCLUSION

The practice of removing trapped air from bags of TRU, low level, mixed and asbestos waste through the high efficiency particulate air filters saves up to $6000 per 90 cubic foot, B-25 box. Laboratory studies indicate that 50% to 70% of the volume of a typical waste storage bag is trapped air. Evacuation of trapped air saves $45 to $65 per cubic foot of soft compressible waste. Long term savings may approach $10 billion over the next 75 years. The Savannah River Site is presently investigating the snap-tight HEPA bag filters for use in evacuating trapped air from 55 gallon polyethylene bags and complete data from the study will be available from the authors in the Spring of 1996.

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38-27

DEVELOPMENT OF A SCINTILLATION FLOW-CELL DETECTION SYSTEM FOR ENVIRONMENTAL RESTORATION AND WASTE MANAGEMENT APPLICATIONS*

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ABSTRACT

A flow-cell detection system was developed utilizing a coincidence circuit and tested with BaF2, CaF2:Eu and scintillating glass. The coincidence detection system reduced the background from ~200 cps to ~0.5 cps. The detection efficiencies for these cells ranged from 0.38 to 0.66 for 45Ca beta particles (Emax=0.257 MeV) and from 0.45 to 0.52 for 233U alpha particles (Ea = 4.8 MeV). The minimum detectable activity was calculated for a 30 s count time and determined to be in the range of 1-2 Bq.

INTRODUCTION

The use of ion chromatography (IC) for radiochemical separations is a well established technique. IC is commonly used in routine environmental monitoring applications as well as in specialized research applications. Typical usage involves the separation of a single radionuclide from the non-radioactive constituents. During the past decade, a limited amount of research has been conducted using automated IC systems in actinide

separation applications (e.g., (1). More recently, separation procedures for common non-gamma emitting activation and fission products were developed utilizing a high performance liquid chromatography (HPLC) system (2,3) (Fig. 1). In addition, a separation procedure for six common actinides has been developed using a HPLC system (4). These latter systems used on-line flow-cell detectors for quantification of the radioactive constituents of the effluent stream. Figure 2 is an example of the actinide separation and on-line flow-cell detection of convenient activities (20-80 Bq/radionuclide) (5). In order to apply HPLC with on-line detection to environmental samples, sample preconcentration and a lower detection limit are a requisite.

Fig. 1

Fig. 2

Flow-cell scintillation detection systems have been developed over the past 30 years. Although other designs have been evaluated, the most common is a translucent tube in close proximity to two photomultiplier tubes (PMTs) coupled in coincidence. The coincidence counting system is used to reduce the background count rate. Radioactive samples come in contact with the scintillator in the flow-cell and the scintillation photons are measured by the PMTs. The flow-cell can contain either a homogeneous (liquid) scintillator or a heterogeneous (powder) scintillator. In a homogeneous detection system the sample is mixed with the liquid scintillation cocktail upstream of the flow-cell, and the mixture passes through the flow-cell for quantification. The major advantages of a homogeneous flow-cell are high probability of interaction and relatively low background count rate. Disadvantages include variable quench, relatively low luminosity, and increased complexity resulting from the additional pump and mixing apparatus. The heterogeneous flow-cell typically consists of an inert inorganic scintillator that is crushed and sieved into small (50-100m) particles. Advantages of a heterogeneous flow-cell are ease of use and relatively high luminosity. Disadvantages are higher background count rates and a high probability of self-absorption resulting in an overall detection efficiency that is lower than for a homogeneous flow-cell, particularly for low energy beta emitters.

OBJECTIVES

A project has just been initiated at Clemson University to develop a HPLC/flow-cell system for the analysis of non-gamma emitting radionuclides in environmental samples. An important component of this project is development of a low background, flow-cell detector that counts alpha particles and beta particles separately through pulse shape discrimination (PSD). The objective of the work presented here is to provide preliminary results of an evaluation of the following scintillators: CaF2:Eu, scintillating glass, and BaF2. Both CaF2:Eu and scintillating glass are common heterogeneous flow-cell detector materials. The advantage of CaF2:Eu is the higher luminosity while the advantage of the glass is its inertness. BaF2 was chosen as a new material for investigation, with potential advantages during later parts of the project.

PROJECT DESCRIPTION

Flow-Cell Detectors

CaF2:Eu and glass scintillators (GS-20: cerium activated lithium glass) were purchased from Bicron; BaF2 was purchased from Optovac, Inc. CaF2:Eu and BaF2 were purchased as rough crystals that were subsequently crushed

and sieved to a 63 to 90 mm particle size range. GS-20 scintillator was purchased as 63 to 90 mm particles. The scintillators were individually packed into 3.0 mm OD X 1.5mm ID X 120 mm polytetrafluoroethylene tubing and coiled to 1.5 turns to yield an approximate active volume of 0.08 mL.

Radioactive Sources

Slightly acidic aqueous solutions of an alpha emitter, 233U (Ea=4.8MeV), and a pure beta emitter, 45Ca (Emax = 0.257 MeV), were used to evaluate the flow-cell. 233UO2(NO3)2 solution at pH 5.5 and concentration of 475 Bq/ml was used. 45CaCl2 was dissolved in deionized water at pH 5.5 at a concentration of 670Bq/ml.

Electronic Circuit

A schematic diagram of the electronic circuit used to acquire the data is presented in Fig. 3. The electronic modules were all standard Nuclear Instrument Module electronics. The flow-cell detector resides in a bath of silicon oil positioned between two Hamamatsu R292 PMTs that were separated by 1 cm. The anodes of the PMTs were grounded through a 50 W resistor and used for timing. The timing signal, generated by the Ortec 935 constant fraction discriminator, was fed into an Ortec 567 time-to-amplitude converter (TAC). The TAC range was set to 50 ns, and the output gated the pulse height and pulse shape inputs to the analog-to-digital converter (ADC, Aptec MCA card). The pulse height signal was acquired from dynode 11 and had a 1MW load resistor. The pulse height outputs from the PMTs were connected to Ortec 113 scintillation preamplifiers which were connected to Canberra 2021 amplifiers with 3 ms shaping times. The unipolar outputs from the amplifiers were connected to an Ortec 533 sum amplifier which is output to the pulse height ADC and the data stored on a personal computer.

Fig. 3

Minimum Detectable Activity

In radiation detection applications, the traditional approach for quantifying detector sensitivity is through the lower limit of detection (LLD). LLD is defined, on the basis of statistical hypothesis testing, as the smallest amount of activity that will yield a net count for which there is a confidence at a predetermined level that activity is present (6). For 5% risks of false detection and false non-detection, LLD is given as:

Eq. 1

where CB is the background count rate and t is the residence time of the sample in the detector.

Minimum detectable activity (MDA) is a function of the theoretical LLD, count time, and detection efficiency:

Eq. 2

where e denotes detection efficiency. To lower the MDA, t and/or e could be increased, and/or CB could be reduced. For applications involving HPLC, t is limited by the resolution of the chromatographic peaks. A 30 second residence time is typical while 60 seconds would be an upper limit. For a heterogeneous flow-cell, e is limited by the particle size of the scintillator. The range of the 233U alpha particles in water is 43 mm. This range is on the order of the interstitial spacing in the flow-cell. A smaller particle size would yield smaller interstitial spacing and hence higher efficiencies resulting from less self-absorption, but is limited by increased back pressure. The remaining variable parameter affecting the MDA is CB. For this paper, coincidence detection techniques

are used to reduce the background events that are attributed to thermonic emissions of electrons from the photocathode of a PMT.

Coincidence Detection

The count rate of a coincidence detection system is related to the background rate in each detector in the following manner:

Eq. 3

where

Rcoin = the coincidence count rate,
R1, R2 = the single detector count rates, and
t = resolving time of the detection system.

As stated earlier, t was set to 50 ns. With the typical background count rate for a PMT at 500 counts per second (cps), the expected count rate of the coincidence detection system is 0.025 cps. The theoretical lower limit will be obtained only in the case when there are no electrical or magnetic interactions between the PMTs, when no optical cross-talk occurs, when the scintillator is not inherently radioactive, and when the shielding from external radiation is sufficient.

RESULTS

A typical pulse height distribution for a flow-cell detector is displayed in Fig. 4, shown here for the CaF2:Eu flow-cell. Tailing of the 233U alpha full-energy peak is due to self-absorption effects. In the cases of GS-20 and BaF2 the amplifier gains were adjusted to X3 and X2, respectively, the setting used for CaF2:Eu. Increasing the gain compensated for the reduced luminosity of the latter scintillators thus keeping the alpha peak channel approximately the same.

Fig. 4

The concept of using coincidence detection as a means to reduce the background count rate appears to have potential, but care must be taken in selecting a scintillator with a low intrinsic background. Table I summarizes the background count results for the CaF2:Eu flow-cell. Note that the measured background count rate is considerably above the theoretical background count rate (typical for the other scintillators as well), thus giving an indication of the margin for improvement. The elevated background count rates of CaF2:Eu and BaF2 were attributed to the insufficient inertness of the materials resulting in adsorption of the radionuclide onto the scintillator. Elevated background count rates of BaF2 and also GS-20 were attributed to intrinsic radioactivity of the scintillation material. By gamma-ray spectroscopy it was determined that the intrinsic background in BaF2 was due, at least in part, to radium contamination. The elevated background of GS-20 was attributed to thorium, which is a common contaminant of glass.

Table I

Flow-cell detector volumes were determined by a conductivity measurement using a NaCl solution that filled the detector. The detector volumes were all determined to be nominally 0.08 ml. Acidic 233U solution (233UO2(NO3)2, pH 1) at a concentration of 130 Bq/ml was used to determine the efficiency of the flow-cells. An acidic solution was necessary to ensure minimal adsorption of the radionuclide onto the scintillator. The CaF2:Eu and GS-20 flow-cells had a detection efficiency of 0.54 for 233U. The GS-20 and BaF2 flow-cells had detection efficiencies of 0.38 and 0.66, respectively, for 45Ca. But since the uranium in the aqueous solution interacted with the BaF2 and the 45Ca interacted with the CaF2:Eu, the efficiency could not be directly measured. Table II summarizes these results. The MDA was calculated using

the experimentally determined background count rate (Table I) and detection efficiency, and a 30 second count time.

Table II

APPLICATION

Based on the results obtained in this study, coincidence detection should be used to reduce the electronic noise associated with the photomultiplier tubes. The reduction in the background count rate was several orders of magnitude (from a single PMT rate of 200 cps to a background count rate of 0.5 cps). Despite the relatively high background count rates, the minimum detectable activities calculated for these flow-cells were 1 Bq for CaF2:Eu and BaF2, and ~1-2 Bq for GS-20.

FUTURE ACTIVITIES

Presented here were preliminary results of the first task of the project. Additional work on this task includes the evaluation of additional scintillators and the addition of pulse shape discrimination, active shielding, and passive shielding to the detection system. Other tasks include the identification of sample interferences in the chromatographic portion of the apparatus, the development of sample processing protocols, and laboratory testing of the entire system using surrogate environmental and waste samples.

ACKNOWLEDGMENTS

The authors wish to thank David D. Brown for his assistance in preparing the radioactive solutions as well as measuring the detector volumes, Dr. John D. Leyba for assisting in the intrinsic radioactivity analysis of the scintillators, and Jagdish L. Malhotra, METC Contracting Officer's Representative.

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38-29

THE BEHAVIOR OF TRITIUM IN SOLID WASTE MATERIALS

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ABSTRACT
At Ontario Hydro's Darlington Nuclear Generating Station (DNGS), the Darlington Likely Clean Waste Management Program qualifies bagged solid wastes for free release. As part of Likely Clean waste processing, the wastes are monitored for total gamma emitters in a bag monitor and hand-frisked with gas proportional instruments to detect alpha and beta-gamma contamination. Neither of these methods of contamination monitoring is appropriate for the detection of tritium oxide (hereafter referred to as tritium), because it is a low-energy beta emitter. A system of multiple checks and balances has been established to minimize the risk of free release of tritiated wastes. The system involves the exclusion of areas that are likely to generate tritiated wastes from the Likely Clean Program. As well, bagged Likely Clean wastes are evaluated for the presence of tritium by storing them for 24 hours after collection to allow any tritium in the waste to reach an equilibrium concentration in the waste bag air. The waste bag air tritium concentration is then measured. If the waste bag air contains less than 0.5 MPC(a) tritium (1.85 x 105 Bq/m3), the waste bag is eligible for opening and the hand-frisking of each individual piece of waste. Wastes that contain no detectable radioactive contamination are eligible for free release.
A series of experiments was conducted to test the appropriateness of the tritium detection protocol. The conclusions of the experiments were:
1) The safety of workers performing hand-frisking of the wastes is assured by using the maximum bag air tritium concentration of 0.5 MPC(a). However, bag air tritium concentration does not provide a quantitative evaluation of the tritium present in the bagged wastes.
2) 24 hours of storage prior to tritium sampling was sufficient time for establishing an equilibrium bag air tritium concentration for non-absorbent materials at room temperature, but was not sufficient time for absorbent materials.
INTRODUCTION
The Darlington Likely Clean Waste Management Program has been in operation in Zone 3 of the Darlington Nuclear Generating Station (DNGS) since the fall of 1992. Zone 3 of DNGS, which consists of all areas that contain radioactive systems and material, is divided into areas eligible for Likely Clean waste collection and those excluded from the Likely Clean program. This division was made on the basis of radioactive source term investigation. Excluded areas include the areas where tritiated wastes are likely to be generated.In the eligible areas of Zone 3, waste generators segregate solid wastes into either "Active" or "Likely Clean" waste cans. Likely Clean waste bags are sealed at collection and, if they contain no visible free liquid, are stored at room temperature for 24 hours to allow any tritium in the waste to reach an equilibrium concentration in the waste bag air. After 24 hours, the Likely Clean waste bag air is sampled with a Triton III tritium monitor, and waste bags with a tritium air concentration of less that 0.5 MPC(a) are classified as eligible for further processing as Likely Clean waste. At Ontario Hydro, 1 MPC(a) for tritium is defined as 3.7 x 105 Bq/m3. The choice of 0.5 MPC(a) as the eligibility level for hand frisking was based on two factors:

1) The Ontario Hydro requirement for respiratory protection applies for air tritium concentrations above 1.0 MPC(a). Therefore, workers performing the hand frisking step would not be required to wear respirators, and
2) 0.5 MPC(a) was felt to be the lowest level of air tritium concentration that could be measured with confidence using the Triton III.

Further processing begins with a gross gamma reading of each Likely Clean waste bag in a waste bag monitor. If the waste bag meets the program's criterion for gross gamma, it is eligible for hand-frisking. The waste bag contents are placed onto a HEPA-ventilated sorting table, with a 944 L/s airflow established across the sorting table. Each piece of waste is hand frisked with a gas-proportional instrument to measure the levels of alpha and beta-gamma contamination. If no radioactive contamination above the Minimum Detectable Activity (MDA) for the instrument is detected, the wastes are "free released" (1). Free release of these wastes makes them eligible for recycling or disposal at any landfill, and puts no restriction on the end use or destination of the material.

In 1994, the Darlington Likely Clean Waste Management Program requested that a series of experiments be conducted to determine:

1) whether 0.5 MPC(a) is an appropriate maximum bag air tritium concentration to determine the eligibility of the wastes for hand frisking.
2) whether 24 hours was sufficient time for an equilibrium bag air tritium concentration to be reached at room temperature. If not, what percentage of the equilibrium concentration was reached after 24 hours.
3) whether the degree of absorbency of the waste had an effect on the rate of establishment of the equilibrium bag air tritium concentration.

PHASE 1

The objective of Phase 1 was to determine whether tritium air concentration inside empty waste bags is dependent on the specific activity of tritium in the water inside the bag, or on the total tritium activity in water inside the bag. This would indicate whether bag air tritium concentration values in MPC(a) can be used to estimate the tritium inventory of bagged solid waste.

Two tritium solutions were prepared by adding tritium to distilled water. Solution 1 contained 3.7 x 107 Bq/l tritium, and Solution 10 contained 3.7 x 108 Bq/l tritium. A schematic drawing of the equipment setup for all phases is provided in Fig. 1. In all phases, baseline tritium air concentration and ambient gamma doserate measurements were performed, and fumehood air temperature was recorded. The volume of a typically-full Likely Clean waste bag was estimated at about 59 liters by pumping a known volume of air into an empty waste bag. Using the assumption that about 2% of the empty bag volume will be water vapor at room temperature, 1 ml of Spike Solution should have provided enough water vapor to saturate the air contained in an empty Likely Clean Waste bag.

Fig. 1

One bag was labelled as Waste Bag Empty-1, and the other as Waste Bag Empty-10. A 1.5 liter glass beaker was installed in the bottom of each bag. Five milliliters of Solution 10 were added to the beaker inside Bag Empty-10. The tritium air concentration in the bag air was recorded upon addition and every 15 minutes thereafter (Fig. 2). Further additions of 10 ml, 20 ml, 30 ml and 60 ml, 125 ml, 250 ml and 500 ml were made. Time

was allowed between each addition for a stable air concentration reading to be established and recorded.

Fig. 2

The same injection protocol was followed for Waste Bag Empty-1, using Solution 1. The waste bags were left overnight with the tritium monitor's pump circulating bag air. Bag air tritium concentration was observed and recorded the following morning.

From Fig. 2, after the injection of 35 ml of Solution 1, and 3 hours and 15 minutes after the start of the experiment, a stable tritium air concentration of 1.4 MPC(a) was reached and maintained throughout the addition of a further 965 ml of Solution 1, with a final reading of 1.2 MPC(a). After the addition of 35 ml of Solution 10, a stable bag air tritium concentration was reached, this time at 11 MPC(a). The bag air tritium concentration was still at 11 MPC(a) the next day with 1 liter of Solution 10 in the bag.

Once the air inside the empty Likely Clean waste bags became saturated with tritiated water vapor, one would expect that the saturation water vapor content of the bag would remain constant at constant temperature and pressure regardless of how much water was added to the bag. Thus, for a given Spike Solution tritium concentration, one would predict that the concentration of tritium in the bag air would reach a stable MPC(a) value at water vapor saturation and remain at that value for all further volume additions of tritium solution. Given sufficient time, only 1 ml of tritium solution was sufficient to saturate the Likely Clean waste bag air with water vapor. Thus, it is not surprising that the bag air tritium concentration reached a stable value that did not change regardless of the addition of further tritium solution volumes. This phase of the experiment was conducted to illustrate the fact that the stable bag air tritium concentration was dependent of the specific activity (Bq/l) of tritium in the tritium solution but not on the total activity (Bq) of tritium present in the bag. This result has significant implications for tritium detection in bagged waste, as it indicates that sampling bag air would be an indication of the specific activity of tritium in liquid absorbed in solid waste, but would not indicate the total tritium activity present in the waste bag. Thus, using the Darlington Likely Clean Waste Management tritium detection protocol, it is not possible to quantify the total tritium inventory contained in a bag of Likely Clean waste. If unconditional release were based on the IAEA Unconditional Release levels (Bq/g), as is the case for other Clearance programs in Canada, sampling the bag air would not give the necessary quantitative information about the total tritium content of the waste.

PHASE 2

The objective of Phase 2 was to compare the time taken for waste bag air tritium concentrations to reach equilibrium inside bags of absorbent material (paper towelling) and in bags of non-absorbent material (plastic) by adding a known quantity of tritium to new material. A second objective was to determine what percentage of the anticipated bag air tritium concentration was reached after 24 hours, based on Phase 1 results.

Two large rolls (2.5 kg) of paper towel were torn into individual sheets, and 2.5 kg of paper towel were placed in each of two Likely Clean waste bags. An injection line was taped into the center of the paper towel as it was emplaced. This allowed for the injection of tritium solution into

a "worst-case" location at the center of the waste bag. The two bags were labelled Waste Bag Paper-10 and Waste Bag Paper-1.

An addition of 15 ml of Solution 10 was calculated for Waste Bag Paper-10. This 15 ml addition would result in a specific activity of 2.22 x 103 Bq/g in the waste, and the IAEA (2) unconditional release criterion for tritium in solid materials is 3 x 103 Bq/g for tritium. Although slightly less than the release criterion, 15 ml of tritium solution were added because its behavior in an empty waste bag had been studied in Phase I. For Waste Bag Paper-1, 200 ml of Solution 1 were added, which equals a tritium specific activity of 3 x 103 Bq/g of solid material. The experiment ran for over 6 days.

In the second part of Phase 2, two bags of Likely Clean waste containing mostly plastic were selected. The bags had been evaluated for tritium using the Likely Clean program's protocol, and had <0.5 MPC(a). The first waste bag, which weighed 3 kg, was labelled Waste Bag Plastic-10. The second waste bag, which weighed 4.5 kg, was labelled Waste Bag Plastic-1. Ten milliliters of Solution 10 were injected into Waste Bag Plastic-10. Thirty-five milliliters of Solution 1 were injected into Waste Bag Plastic-1, and the bag was examined for the presence of liquid. When none was observed, an additional 90 ml of Solution 1 was injected into the bag. At this point, free liquid was observed in the waste bag. Since the presence of free liquid in a Likely Clean waste bag makes it ineligible for Likely Clean processing, no more tritium solution was added. The bags were left overnight with bag air circulating through the tritium monitors.

The data from Phase 2 are shown in Fig. 3. In the bags of paper towelling, a detectable rise in bag air tritium concentration did not occur within 24 hours of the injection of the tritium solutions. For Waste Bag Paper-1, the bag air tritium concentration took 5 days to rise to 0.4 MPC(a), which is below the tritium criterion used at DNGS for the rejection of Likely Clean waste for hand-frisking. This level of tritium contamination in absorbent waste would not likely pose an occupational risk to personnel performing hand frisking, as offgassing concentrations are low and the ventilated sorting tables have good airflow. However, the paper towel was spiked near or at the IAEA (2) Unconditional Release Levels, so using 0.5 MPC(a)as the only tritium release criterion for release-level based waste segregation programs would not be advisable. Based on Phase 1 results, one would expect the maximum bag air concentration that could be reached in Waste Bag Paper-1 would be 1.4 MPC(a). Water present in the bag due to the natural water content of the paper towelling would have exchanged with the tritium solution water and vapor, resulting in the dilution of the specific activity of tritium in the waste bag. It is not likely that the maximum bag air tritium concentration for an empty bag (from Phase I) would ever have been reached.

Fig. 3

The bag air tritium concentration in Waste Bag Paper-10 took 5 days to rise to 0.5 MPC(a) above baseline. The saturated bag air tritium concentration in the equivalent empty bag in Phase 1 rose to about 11 MPC(a). These results indicate that paper towelling, and likely any other absorbent material, requires a considerable length of time in a closed bag at room temperature to yield enough tritiated water vapor to exceed the DNGS Likely Clean Program's tritium criterion of 0.5 MPC(a). Either a longer laydown period (greater than 24 hours) prior to tritium

monitoring, or some method of enhancing the offgassing of tritium from the absorbent waste prior to waste bag air sampling would be advisable. Figure 3 indicates that tritium solutions added to bagged plastics reached levels in excess of 0.5 MPC(a) within 24 hours of the addition. Since the plastics contain little or no water, there was no dilution of the tritiated water in the bag. The bag air tritium concentration inside Waste Bag Plastic-1 reached the level predicted by Phase 1 of 1 MPC(a) within the 24 hours. The bag air tritium concentration inside Waste Bag Plastic-10 rose to 6 MPC(a) within 24 hours, as compared to a predicted value of 10 MPC(a). The reason for the lower-than-predicted reading may have been the presence of a small amount of absorptive material, as a cotton glove was observed in the bag. These results indicate that the 24 hour laydown prior to sampling the bag air for plastics, and likely any other non-absorbent material, is sufficient.

PHASE 3

The objective of Phase 3 was to confirm Phase 2 results by performing Phase 2 in reverse order. Tritium solution was poured into a plastic glove bag and allowed to reach a stable bag air tritium concentration. The tritium solution was then soaked up using known quantities of paper towel and Tyvek to demonstrate any effect on bag air tritium concentration.

Two plastic glove bags were labelled Waste Bag Paper-1 and Waste Bag Tyvek-1. A sealed plastic vial containing 15 ml of Solution 1 was placed in each glove bag. For Waste Bag Paper-1, 210 g of new paper towelling was sealed into a zippered plastic bag and then placed into the glove bag. The 15 ml of Solution 1 would spike this amount of paper towel to a level of 2646 Bq/g, which is slightly below the IAEA (2) unconditional release level of 3,000 Bq/g. For glove bag Tyvek-1, a tyvek disposable suit was cut into pieces, each of which was weighed and labelled, and then the tyvek material was put in another plastic zipper bag and installed in the glove bag. The tyvek material was spiked to a level of 2282 Bq/g, which is also slightly less that the IAEA (2) unconditional release level.

The vials were opened using the gloves, and Solution 1 was poured into the glove bags. The next morning, the gloves were used to open the zipper bag containing the paper towelling inside Waste Bag Paper-1, and the tritium solution was absorbed using the paper towelling. The procedure was repeated in Waste Bag Tyvek-1, using a known weight of Tyvek material to soak up the solution, and then emptying the rest of the Tyvek into the bag. Tritium bag air concentration for both bags was recorded continuously on the chart recorder.

The volume of an empty glove bag was measured at about 120 liters.Using the same assumptions as in Phase 1 for the saturation of bag air with water vapor, and 2 ml of tritium solution would provide enough water to saturate the air in an empty glove bag.

The results of Phase 3 are shown on Fig. 4. Tritium bag air concentrations in the empty glove bags rose to the values predicted by Phase 1, about 1 MPC(a), before the tritium solution was absorbed into the paper towel and Tyvek material. Absorbing the Spike Solution into the paper towel resulted in a drop in tritium bag air concentration to 0.1 MPC (a) within 2 hours. Absorbing the tritium solution into Tyvek material did not result in any decrease in bag air concentration. The slight rise in bag air MPC(a) values after the addition of Tyvek material was likely due to the observed beading of the tritium solution over a

large surface area of material, making evaporation of the tritium solution more efficient. These results confirm the Phase 2 conclusion that the presence of absorbent material in bags of waste does decrease the anticipated tritium equilibrium concentration in the bag air. Therefore, enhanced sampling methods are required to ensure that a significant percentage of the tritium present in the absorbent waste is detectable in the waste bag air within 24 hours.

Fig. 4

CONCLUSIONS

There are two bases upon which the appropriateness of using a bag air tritium concentration of 0.5 MPC(a) to determine the eligibility of Likely Clean wastes for hand-frisking must be evaluated. They are:

i) The assurance of a low probability of the free release of tritiated wastes must be established. Assurance of the low probability of free release of tritiated wastes is achieved by the combination of the exclusion of areas of Zone 3 that would generate tritiated wastes, and by the utilization of 0.5 MPC(a) tritium in waste bag air as a pass/fail criterion for further processing of the waste as Likely Clean. As a result, no changes are required for the tritium detection protocol for the Darlington Likely Clean Waste Management program. However, the results of these experiments clearly illustrate that bag air tritium concentration is dependent on the type of waste material and the specific activity of the tritium solution. The bag air tritium concentration does not provide a quantitative evaluation of the tritium present in the bagged wastes.

ii) A consideration of the safety of personnel performing waste segregation. Ontario Hydro policy does not require the use of respiratory protection for tritium at air concentrations below 1.0 MPC(a), so the choice of the setpoint for bag air tritium concentration of 0.5 MPC(a) is appropriate. Additional worker safety is assured because adequate airflow is established across the ventilated sorting tables where the hand-frisking of the waste occurs.

With respect to whether the 24 hour storage period prior to tritium monitoring of the bagged wastes, 24 hours was sufficient time for establishing an equilibrium bag air tritium concentration for non-absorbent materials at room temperature. However, 24 hours was not sufficient time for establishing an equilibrium bag air concentration for absorbent materials at room temperature. The percentage of equilibrium concentration reached after 24 hours for paper towelling was not sufficient to estimate the equilibrium bag air concentration with confidence.

ACKNOWLEDGEMENTS

The financial support of CANDU Owner's Group Working Party 49 (Low and Intermediate Level Waste) is gratefully acknowledged.

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38-30

REAL-TIME ALPHA MONITORING OF A RADIOACTIVE LIQUID WASTE STREAM AT LOS ALAMOS NATIONAL LABORATORY

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ABSTRACT

This paper discusses the development, installation, and testing of a real-time radioactive liquid waste monitor at Los Alamos National Laboratory (LANL). This detector system was designed for the LANL Radioactive Liquid Waste Treatment Facility (RLWTF) as a means of real-time monitoring of influent to the plant. By knowing the activity of the influent, the plant operators can better monitor the treatment process, potentially segregate waste better, and monitor the regulatory compliance of users of the LANL Radioactive Liquid Waste Collection System.

This detector system uses long-range alpha detection (LRAD) technology, which is a non-intrusive method of characterization that determines alpha activity on the liquid surface by measuring the ionization of ambient air above the surface. Extensive testing has been performed to ensure long-term reliability with a minimum amount of maintenance. The final design is a simple cost-effective alpha monitor that can be modified for monitoring waste streams at various points in the LANL Radioactive Liquid Waste Collection System. The detector system has been used to monitor low-level waste streams at LANL. Typical activities of influent measured at the RLWTF are tens of nCi/l due to alpha-emitting contaminants.

INTRODUCTION

Design and installation of a working prototype was begun in 1994 to meet the needs of the RLWTF at LANL for monitoring alpha contamination in their influent stream. The detector is based on the long-range alpha detection sample monitor design and monitors the airborne ionization created by alpha radiation (1).

The normal activity level for the alpha component of radioactive liquid waste at the RLWTF is tens of nCi/l. The goal of this project was to install a real-time monitor at the point where the influent enters the treatment facility so that spikes in activity could be detected. The LANL waste acceptance criteria for the main RLWTF plant is 500 nCi/l for alpha contamination. Therefore a real-time alpha monitor would alarm for large changes in activity and allow the operators to closely monitor daily fluctuations in influent activity. This could lead to a better understanding of waste disposal at LANL and perhaps lead to segregation of waste before treatment. Current regulations mandate that both influent and effluent be monitored to determine the overall effectiveness of the treatment process. Traditional methods require up to a day and a half for analysis.

Benchtop tests were made in 1994, and the detector was able to monitor liquid at the 100 pCi/l level. Modifications were made to the detector and it was installed in the pH balancing tank in mid-1995. The detector system was mounted in a non-intrusive manner on the lid of the tank, several inches above the liquid surface. The pH balancing tank is equipped with a mixer that ensures the liquid is homogeneous and a baffle that provides a calm surface of liquid for monitoring. Recent tests with an inline detector system have shown sensitivity at the several nCi/l range. The LRAD-based system is therefore a factor of 10-100 times more sensitive than the alarm limit of 500 nCi/l. This technique should save both time and money, and could be applied to effluent monitoring before discharge into the environment.

DESIGN

Traditional alpha detectors monitor the alpha particles directly, requiring that the detectors be close to the object being monitored. It is difficult to design a detector to do this because of the changing levels of the liquid stream, humidity, and other engineering concerns. The radioactive liquid waste monitor being developed by LANL is not subject to these constraints because it monitors the airborne ionization created by alpha particles generated on the liquid surface. Tests at LANL have shown that this ionization can be transported via an electrostatic field or by airflow up to distances in excess of several meters, depending on the application. Each 5-MeV alpha particle can create approximately 150,000 ion pairs, and this technology has proven to be highly sensitive for monitoring alpha contamination (2).

The Radioactive Liquid Waste Monitor (RLWM) design is based on the electrostatic long-range alpha detector (3). Initial tests were made by inserting trays of radioactive influent into the LRAD Sample Monitor. This simple electrostatic design consists of a metal enclosure and a high-voltage signal plane. The signal plane is a metal plate maintained at 300 V DC, and a highly sensitive electrometer is used to detect changes in current to the plate. The box is maintained at ground potential, and so the ions produced by alpha contamination are collected on the plate and box, according to their polarity. By using standard alpha sources for calibration, the resulting current in the system can be linearly scaled to give surface activity. The calibration was made using a set of National Institute of Standards and Technology (NIST) traceable Pu-239 sources with strengths ranging from 100 to 1100 dpm-alpha. For this prototype, the conversion for dry samples is 1 mV = 59 dpm, and the monitor is able to discriminate dry samples with an activity less than 100 dpm-alpha.

In the first design for the RLWM, a signal plane and guard plane were suspended above a stainless steel sink. Because the radioactive liquid was shown to contaminate the sink, a stainless steel insert was used to hold the liquid. Therefore a background could be taken between samples without concern for cross-contamination. This benchtop model was able to monitor changes in volume activity of the liquid and the measurements were compared to the gross alpha measurements taken by drying 10 ml samples and monitoring them in a traditional alpha detector.

The next design (see Fig. 1) was built to fit into an opening in the pH balancing tank at the RLWTF. This model had a smaller surface area than the benchtop model. A preliminary calibration showed that it had a good linear response for the dry Pu-239 source set. However, because of background ionization in the pH balancing tank, the sensitivity of the detector was greatly reduced. It is believed that this background was primarily caused by contamination on the sides of the tank and the charging of the Lexan lid. In addition, the guard plane proved inadequate for shielding the signal plane from large variations in current that were not directly related to the activity of the influent.

Fig. 1

In mid-1995 the RLWM was removed from the pH balancing tank to implement and test modifications. The final design, shown in Fig. 2, had several important changes over the original design, and it was tested in a benchtop configuration with a stainless steel sink. The guard plane was shaped so that the acceptance of the signal plane was reduced, causing the signal on the signal plane to come primarily from the surface beneath

the detector and not from the Lexan lid or the sides of the tank. Another important modification was the use of additional insulated standoffs between the signal and guard planes. By increasing the path lengths over surfaces between the ground, guard, and signal planes, leakage current was reduced and the detector was less sensitive to humidity in benchtop tests. During tests in which high-activity sources were placed in various geometries with respect to the detector, it was found that the signal wire, running from the battery to the signal plane, could collect large amounts of ionization. The resulting contribution to the signal current was reduced significantly by shielding the signal wire with an insulator; however, the signal remained noisy due to fields induced by charging of the insulating material. This noise was eliminated by adding a stainless steel mesh shield around the insulator and connecting the mesh to the guard plane.

The RLWM was re-installed in the pH balancing tank in November 1995. The detector initially showed a very poor correlation to liquid activity measured by taking grab samples near the detector. The current from the signal plane was correlated strongly with liquid level, most likely due to heavy contamination of the tank walls. As the liquid level rose, covering more of the wall surface, the signal would drop proportionally. The level is monitored continuously by the treatment facility, and it is possible to record this data simultaneously with our detector signals. By subtracting the effect of change in the liquid level, the correlation between the detector signal and the grab sample activity increased dramatically. Periodic fluctuations on the order of seconds in the signal current, induced by the motion of the liquid, were also encountered. These were damped by placing a suitably chosen capacitor across the electrometer to form a low-pass filter.

Fig. 2

RESULTS

Figure 3 shows the corrected RLWM signal over a 24-hour period. The baseline obtained with the conversion remains constant over the several weeks in which data was obtained. Figure 4 shows the data points obtained from the corrected signals plotted against the activities of grab samples in counts per minute, with the bold line giving a linear fit to the corrected signals. The raw signals decrease with activity, due to the rise of liquid levels associated with a contaminated influent. By including a level correction, the correlation is much improved.

The grab samples were filtered to remove particulate matter that contributes little to the RLWM signal because of wetting of the particle surface. The maximum values for activity obtained correspond to 100 nCi/l, while the non-filtered values ranged from 120% to 150% of the filtered values. The range of activity shown is only 20% of the action level of 500 nCi/l, and further data at higher activity levels would be useful for obtaining better calibration of the signal correction factors.

Fig. 3

Fig. 4

CONCLUSIONS

The results of monitoring influent at the RLWTF at LANL show that the RLWM is sensitive to changes in alpha activity in the influent liquid stream at the 10 nCi/l level. There is a great deal of uncertainty remaining because of the procedures used for drawing off grab samples. The calibration for LRAD applications, such as this, is always difficult when the range of samples is not great. A much improved calibration could

be made by including data for high-activity samples; however, this sort of calibration will have to be made over time by drawing off samples when the detector alarms. Therefore, over time, a much more sensitive calibration could be made. Even with a less-sensitive calibration, the detector has been shown to have a stable baseline over time and large changes in activity can clearly be measured, giving the operators at the RLWTF important information. Further development with standard liquid sources would allow improvements to be made to this detector design, resulting in a much more sensitive alpha monitoring system for liquids. Continuing development of this detector could lead to a sensitive monitor for treated effluent streams as well. This sort of application would be very useful for monitoring low-level radioactive liquid streams before discharge into the environment. A more sensitive design could ensure regulatory compliance and be used to monitor samples in the field: saving time and money over other sampling methods.

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MIGRATION OF SR-90, CS-137, AND PU-239/240 IN A CANYON BELOW A LOS ALAMOS OUTFALL*

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ABSTRACT

Technical Area-21 (TA-21) of Los Alamos National Laboratory (LANL) is on a mesa bordered by two canyons; DP Canyon and Los Alamos (LA) Canyon. DP Canyon is a small semi-arid watershed with a well defined channel system where the stream flow is ephemeral. TA-21 has had a complex history of waste disposal as research to determine the chemical and metallurgical properties of nuclear materials occurred here from 1945-1978. Due to these operations, the TA-21 mesa top and bordering canyons have been monitored and characterized by the LANL Surveillance and Environmental Restoration Program. Results identify radionuclide activities at a clay pipe outfall identified as Solid Waste Management Unit (SWMU) 21-011(k) which exceed Screening Action Levels, and at locations along DP Canyon which exceed regional background levels.

The radioactive contaminants considered in this study are strontium-90, cesium-137, and plutonium-239/240. This paper examines sediment transport

and contaminant migration from the source at SWMU 21-011(k), which is located on a slope above DP Canyon. Three dimensional surface plots of data from 1977-1994 are used to portray the transport and redistribution of radioactive contaminants in an alluvial stream channel. Contaminant activity has decreased since 1983, in part due to more stringent laboratory controls, and also due to the removal of the main plutonium-processing laboratories to another site.

INTRODUCTION

Technical Area-21 (TA-21) of Los Alamos National Laboratory (LANL) lies on the northern section of the Laboratory, at an elevation of 7140 ft. It is located on the Pajarito Plateau,midway between the steep slopes of the Jemez mountains to the west and the Rio Grande river to the east. TA-21 lies on DP Mesa, immediately east-southeast of the Los Alamos town site. TA-21 was the plutonium processing facility for nuclear weapons research and development from 1945 to 1978. After 1978, operations were scaled back dramatically but not stopped completely. Liquid radioactive waste was treated at a site treatment plant from 1969 to 1995. TA-21 is currently being decontaminated and decommissioned.

The Liquid Radioactive Waste Treatment Plant processed aqueous effluent from TA-21 chemical laboratories. The effluent included the isotopes plutonium-239/240, cesium-137, and strontium-90, among other radionuclides. After processing, the effluent was discharged into DP Canyon via an outfall now referred to as SWMU 21-011(k). A map of TA-21 and DP canyon is shown in Fig. 1. DP Canyon is the small east-west oriented canyon which flanks TA-21 on the north. The walls of the canyon are steep and rocky. The stream channel on the floor of the canyon receives ephemeral flow from precipitation and runoff (summer storms and winter snowmelt). Sediments in the stream channel consist of sand and illitic clay. DP Canyon joins Los Alamos Canyon approximately 1.5 km downstream from SWMU 21-011(k).

Fig. 1

SWMU 21-011(k) received processed waste effluent streams from 1969 until 1978. From 1978 until 1984, it received chemical waste containing various radionuclides. A 400 L aqueous spill of untreated effluent probably containing various contaminants, occurred at the outfall in 1991 (1). The clay pipe at 21-011(k) released effluent into a precipitous rock strewn slope with minimal soil depth.

Sediment Sampling Technique

This investigation examined temporal trends in available data for radionuclides at sediment sampling stations in DP Canyon. The sampling locations included the SWMU site, two locations (DPS-1 and DPS-4) in DP Canyon, and the site of well LAO-3, near the confluence of DP and Los Alamos Canyons (See Fig. 1). The first site was sampled in the summers of 1992 and 1993 (2). The last three sites have been sampled repeatedly by the Environmental Protection Group (3).

Stream bed samples were taken from sampling stations along DP Canyon shown on Fig. 1. A consistent sampling procedure for the sediments was repeated annually to ensure comparability of the results. Samples from the bed of DP stream were collected by digging a line of uniform depth in an undisturbed area across the main channel (4). Using a trowel, 500 ml of sediment was collected at a depth of one-half to three-fourths inch. Soil from SMWU 21-011(k), in contrast with sediment, samples were taken at a depth of one to six inches and six to eighteen inches in the soil. Care was taken to collect the clay-rich fraction with which the

radionuclides are commonly associated (5). The sediment sample is mixed prior to analysis and analyzed at the LANL Environmental Measurements Lab and each year the field samples were taken within a period from late April through May (5).

The Environmental Chemistry Group at LANL used a documented Quality Assurance and Sampling Procedure for the sample analysis and data verification. The analyses of radioactive constituent concentrations were determined by gamma-ray spectrometry.

Table I portrays the regional background concentration, the Screening Action Level (SAL), and half life of the three elements.

Table I

A SAL is set on potential contaminants in various media based on conservative calculations of human health risk. Soil SALs for radionuclides are calculated using a residential scenario at an annual dose limit of 10 millirem (10 mrem/a) above regional background.

Plutonium-239 decays by alpha- and gamma-emission with a half life of 24,100 years (a). Plutonium-240 is also an alpha- and gamma-emitter, with a half-life of 6560 a. Cesium decays by beta-and gamma-emission, with a half life of 30.17 a. Strontium decays by beta-emission, with a half-life of 28a.

Clay minerals are very effective at retaining these radionuclides. As a monovalent alkali cation, cesium readily exchanges with potassium, and is commonly strongly bound to surface and interlayer sites of clay particles (6). Thus, contaminant cesium is expected to be associated with the finer fractions (especially the <0.053 mm fraction) in soils and sediments (7). Transport of cesium commonly occurs in association with soil and sediment erosion (7). There is evidence of substantial drainage erosion from the SWMU 21-011(k) clay pipe down the slope to DP stream in the form of two drainage channels.

Strontium is a divalent alkali earth cation which readily substitutes for calcium in mineral lattices and on surface sites. Although transport of strontium will also be largely through sediment transport, sorption on mineral grains is generally less strong for strontium, and hence, transport in runoff may also be significant.

Because of their similarity to potassium and calcium, respectively, cesium and strontium are also bioactive (readily absorbed by plants, soil microbes, and animals). Plants and microbes whose growth is limited by calcium and/or potassium will tend to take up larger amounts of strontium and/or cesium (8).

RESULTS AND DISCUSSION

This section discusses the present concentrations in soil and sediment at SWMU 21-011(k), and the variations with time of radionuclide activities at the sediment sampling stations downstream. The three radionuclides discussed are Pu-239/240, Sr-90, and Cs-137. (Note that the standard measurements for plutonium do not distinguish the two isotopes, so the value reported is for the sum). At the source [SWMU 21-011(k)], Pu-239/240 activities range from 7.2 to 46,000 pCi/g,

Cs-137 activities ranged from 8.23 to 2675 pCi/g, and Sr-90 activities ranged from 1.96 to 1800 pCi/g. Figs. 2,3, and 4 show the variations in activities as a function of time and location in DP Canyon. Table II summarizes the data on radionuclides activities in DP Canyon.

The plutonium chart (Fig. 2) portrays the maximum activities occurring at DPS-1, with values at DPS-4 and LAO-3 consistently below 1.0 pCi/g. After 1986, the DPS-1 activities are likewise below 1.0 pCi/g. The outfall

clearly displayed high activities (1900 times the soil SAL) after the decline in levels at DPS-1, so it would appear that transport of plutonium was facilitated by the effluent from the outfall. Without the continued effluent, relatively small degrees of transport occurred to the stream channel, and the remaining contamination at DPS-1 was moved downstream and mixed with less contaminated soil by runoff.
Table II
Fig. 2
Plutonium does not appear to be migrating to DPS-1 or DPS-4 at levels above the SAL. Cesium historically has had high levels at DPS-1, and DPS-4 but currently the levels are well below the SAL. The strontium data is similar. There is an evident decrease in the three radionuclides downstream.
The cesium data (Fig. 3) indicate that concentrations were above the SAL at all three stations until the late 1980s, but have subsequently declined to activities at or near background levels. The high levels seen at SWMU 21-011(k) are not found in the stream channel, so dispersion and mixing of contaminated sediment appears to have diluted the concentration of cesium. In addition, the cessation of effluent release appears to have permitted contamination levels in the channel to decline through transport and/or dilution. However, the activities remained higher for cesium than for plutonium, and cesium is the predominant radionuclide contaminant in DP Canyon. This conclusion is corroborated by the EG&G aerial radiological gamma surveys in 1975 and 1982 (9). These surveys show that cesium is the dominant anthropogenic radionuclide found outside of TA-21, and that it had migrated approximately 600 meters down DP Canyon.
Fig. 3
The strontium data (Fig. 4) show a pattern similar to that for cesium, with high values at the source, values above the SAL at DPS-1 until the mid-1980s, and values below the SALs, at all stations since then. The values are lower than those for cesium, which may reflect the composition of the original effluent, but may also represent the lower sorptivity of strontium. Strontium may have been mobilized into the surface water and alluvial ground water, and transported deeper, or further downstream.
Fig. 4
Table III shows sorption coefficients for all three elements. Strontium is less strongly sorbed onto clay mineral surfaces than plutonium and cesium, but appears to have behaved in a manner similar to cesium in the sediment of DP Canyon. The pattern for plutonium appears unusual in the light of its estimated sorption coefficient, as it is expected to sorb slightly less strongly than cesium, yet it appears hardly to have been transported at all. It is conceivable that plutonium in the effluent was sorbed onto or precipitated in extremely fine colloidal particles which percolated into the soil to some depth. The cesium and strontium, which sorbed rapidly onto clay or silt particles at the surface may have been more susceptible to transport in surface runoff. Further analysis is underway to test this concept.
Table III
Strontium is less strongly sorbed onto clay mineral surfaces than plutonium and cesium, but appears to have behaved in a manner similar to plutonium in the sediment of DP Canyon. Strontium is likely to remain stored in the banks of the stream channel, but to a smaller extent (10). Strontium sorbs to the LA Canyon sediments less strongly than cesium by a

factor of 12. Batch strontium sorption experiments were conducted on 20 soil and 16 channel sediment samples collected in TA-21, and upper Los Alamos Canyon. Cesium sorption coefficients (Kd) ranged from 162.3 to 1444.3 ml/g in soil and 77.5 to 1034.1 ml/g in sediment (11). Strontium sorption coefficients ranged from 15.8 to 67.7 ml/g, with a mean of 35.7 ml/g for soils. For strontium in LA Canyon channel sediments sorption coefficients ranged from 8.8 to 41.3 ml/g, with a mean of 21.4 ml/g (11). These data suggest that strontium is a non-conservative solute that is partially removed from solution through cation exchange.

CONCLUSIONS

The outfall at SWMU 21-011(k) is a major source for radionuclide contamination in DP Canyon. Plutonium has not migrated from the source in concentrations sufficient to exceed the SAL, and activities declined very rapidly after effluent ceased to be released into the SWMU. Cesium and strontium have been transported in quantities sufficient to exceed the SALS, although only cesium activity exceeded SALS at station DPS-4 after 1977. Both declined in activity after effluent ceased to be released into the SWMU. Recent year samples have all shown activities below the SAL values. The systematic relationships among multiple contaminants are complex, involving interactions among surface and ground water hydrologic and geochemical processes, and understanding may also require detailed knowledge of the effluent discharge and surface water history.

ACKNOWLEDGMENTS

This work was funded by the US Department of Energy through the Los Alamos National Laboratory Environmental Restoration Project and Field Unit-1 Project leader- Garry Allen. The authors thank Robert K. Greene, Thane Hendrix, Ningping Lu, Maxwell N. Maes, Gary McMath, David B. Rogers, Ines Triay, William R.J. R. Turney, and Douglas Walther.

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RESULTS AND COST SAVINGS FROM RADIOLOGICAL SURVEYS OF BURIED DRAIN LINES WITH THE PIPE EXPLORERTM SYSTEM*

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ABSTRACT

Science and Engineering Associates, Inc. (SEA) has developed a unique and versatile method of transporting characterization tools into pipes, drain-lines, and ducts. The system, called the Pipe ExplorerTM, uses a pressurized inverting membrane to tow sensors such as radiation detectors, video cameras, and pipe locator beacons through pipes. In addition to towing characterization tools, such as a radiation detector, the membrane also lines the pipe to provide a clean conduit through which the detector travels. This protects expensive sensors by preventing removable contamination from coming into contact with them. Cross contamination is thus eliminated allowing the system to obtain a true measurement of contaminant activity as a function of distance. The system is currently capable of transporting detectors up to 250 feet in pipes 2 inches in diameter and greater. It can also negotiate multiple elbows and obstructions. The DOE Morgantown Energy Technology Center sponsored the development of this technology as a characterization tool to support the decommissioning and decontamination of DOE nuclear process facilities. Recently the system was demonstrated at a site being remediated under the DOE Formerly Utilized Sites Remedial Action Program (FUSRAP). The Pipe ExplorerTM successfully characterized a total of 8, 4-inch diameter buried drain-lines, over lengths as long as 120 feet. The interior surfaces of the drain-lines were coated with a thick oily sludge that contained residual amounts of U-238. The drain-lines were continuously characterized over the entire lengths and were found to have contamination levels ranging from below 3000 dpm/100cm2 to over 1,000,000 dpm/100cm2.

By using the Pipe ExplorerTM system the DOE FUSRAP was able to save in excess of $1,000,000 in this single demonstration. These cost savings were realized through reduction of secondary waste and by avoiding more labor intensive and less accurate survey methods. A description of the system and samples of the data obtained from the FUSRAP site are presented. In addition, data substantiating the cost savings by the DOE are included.

INTRODUCTION

The U.S. Department of Energy's nuclear facility decommissioning program needs to characterize radiological contamination inside piping systems before the pipe can be recycled, remediated, or disposed. Historically, this has been attempted using hand held survey instrumentation, surveying only the accessible exterior portions of pipe systems. Difficulty, or inability of measuring threshold surface contamination values, worker

exposure, and physical access constraints has limited the effectiveness of this approach. Science and Engineering Associates, Inc. under contract with the DOE Morgantown Energy Technology Center has developed and demonstrated the Pipe Explorer system, which uses an inverting membrane to transport various characterization sensors into pipes. The basic process involves inverting (turning inside out) a tubular impermeable membrane under air pressure. A characterization sensor is towed down the interior of the pipe by the membrane.

Advantages of this approach include the capability of deploying through constrictions in the pipe, around 90 bends, vertically up and down, and in slippery conditions. Because the detector is transported inside the membrane (which is inexpensive and disposable), it is protected from contamination and cross-contamination is eliminated. Characterization sensors that have been demonstrated with the system thus far include: gamma detectors, beta detectors, video cameras, and pipe locators. Alpha measurement capability is currently under development.

A remotely operable Pipe Explorer system has been developed and demonstrated for use in DOE facilities in the decommissioning stage. The system is capable of deployment in pipes as small as 2-in-diameter and up to 250 ft long. This paper describes the technology and presents measurement results of a field demonstration conducted with the Pipe Explorer system at a DOE site. These measurements identify surface activity levels of U-238 contamination as a function of location in drain lines. Cost savings to the DOE of approximately $1.5 million dollars were realized from this one demonstration.

PROBLEM

By their nature, the interiors of pipes and ducts are difficult to access. In many cases, even the exteriors are inaccessible. For example, drainlines are buried or encased in concrete and duct work is often elevated or enclosed. To access these structures for characterizations such as radiological surveys, requires significant effort and cost. These costs are further increased if the characterizations are carried out in a radiological control zone, where greater personal protective measures and support crews are required.

Furthermore, for alpha and beta emitting contaminants, such as U-238 and Pu-239, it is necessary to take unobstructed measurements of contaminated surfaces. Thus, external measurements through pipe walls are inadequate and the only way to gather data is to get an instrument inside of the pipe.

Alternative methods to the Pipe Explorer system can be used to transport detectors into pipes, such as pipe crawlers and push rods. However, these methods lead to ambiguous results if there is removable contamination present. With nothing to prevent contamination from getting on the detector there is no way to differentiate between contamination on the pipe wall and contamination on the detector. There are additional limitations associated with these alternative methods. For example, pipe crawlers are typically limited to larger diameter pipes (> 4 in). They are also cumbersome to operate around elbows and have a difficult time in pipes with slippery surfaces. Push rod methods are limited in length and are often unreliable when trying to get a detector around elbows.

SOLUTION

As a solution to this problem, SEA adapted its inverting membrane technology to transport radiation detectors and other characterization tools into pipes. The system uses an air-tight membrane configured such

that when it is pressurized it inverts into a pipe. As it inverts the pressure force on the end of the membrane is adequate to tow a detector around multiple elbows and through several hundred feet of piping. This technology not only provides an effective transportation method for detectors, but it also provides a clean conduit through which the detector can travel.

TECHNOLOGY DESCRIPTION

The primary components of the Pipe Explorer technology are illustrated in Fig. 1. The heart of the system is an air-tight membrane which is initially spooled inside of a canister. The end of the membrane protruding out of the canister is folded over and attached to a basepipe. When the canister becomes pressurized in this configuration, the air pressure on the membrane causes the membrane to be pulled from the spool. This continues until the membrane is completely off the spool. A characterization tool, such as a radiation detector, is attached to the end of the membrane and towed into the pipe as the membrane continues to invert. The detector cabling is also towed into the pipe from the spool. To retrieve the system from a pipe, the process is simply reversed, where the cabling, detector, and membrane are wound back onto the spool. The system can thus be used to move a detector freely back and forth through a pipe while the detector output and position are continuously recorded. As a result, the Pipe Explorer system provides high resolution analysis of the location of radioactive contamination in pipes.

The membrane also provides a clean conduit through which the detector travels. This protects both the detector and the workers handling it. Furthermore, measurements are inherently more reliable. A detector transported in any other fashion runs the risk of removable contamination adhering to the sensor, which can cause erroneously high or false positive readings.

The general operating procedure is to first deploy the membrane halfway into the pipe. This is the point where the detector begins to enter the pipe from the deployment canister. At this time data acquisition is initiated. In most cases the detector is deployed out relatively quickly (up to 30-ft/min). More detailed radiological measurements are taken as the detector is retrieved from the pipe at a slower rate.

As the detector is being retrieved, the tether is wound back into the deployment canister. The membrane prevents contamination from contacting the tether. However, as a precautionary measure, two sampling smears are used to swipe the entire surface of the tether and the detector. When the tether is completely retrieved the smears are surveyed with a pancake GM probe to ascertain if any contamination has potentially been transferred into the canister. To date, no contamination of the canister or tether has been noted. Once the detector has been retrieved and the survey completed (the detector can be re-deployed for additional data if needed), the detector is removed from the end of the membrane. The membrane is then fed through a diaphragm to an external reel assembly or manually fed into a disposal drum. The membrane being handled has been inverted. Therefore, the side of the membrane that has been in contact with the contaminated pipe is contained within itself (this is analogous to the way a Hazmat worker removes rubber gloves). The inexpensive membrane (about $0.03/ft) is then disposed. This secondary waste generation is minimal. Several hundred feet of membrane is easily compacted into less than a cubic foot.

Fig. 1

Capabilities Summary

The absolute maximum deployable distance of the system is currently limited by the length of cabling and canister size. The current configuration allows for 250-ft deployments. Longer distances may be achievable but no applications to date have required any longer attempts. Practical deployment lengths are limited by elbows in the lines and the diameter of the pipe. Table I lists typical results that have been achieved, in laboratory tests, and are used as general guidelines.

Table I

The Pipe Explorer system has been used to transport several different types of radiological measurement instruments. Table II lists these instruments and their descriptions.

Table II

SEA currently has two deployment systems available. The first is a fully automated system. With its motorized operation and built in deployment sensors, it allows for continual unattended pipe surveys. The second system is a smaller, manually operated system.

Additional uses of the Pipe Explorer have been identified and have either been nominally demonstrated or are being integrated with the system. These include;

- Transport of pipe locating beacons
- Transport of video cameras
- Alpha detection methodologies

RESULTS

An extensive demonstration of the Pipe Explorer was conducted for the DOE Formerly Utilized Sites Remedial Action Program (FUSRAP) at a site in Adrian Michigan. During the 1950's the Bridgeport Brass Company operated a Special Metals Extrusion Plant at the site. This was done under contract with the DOE, then the Atomic Energy Commission. The product of this operation was material for uranium fuel elements for reactors in Hanford, Washington, and the Savannah River Plant in South Carolina. Uranium handled in this operation included depleted, natural, and up to 2.1 percent enriched in U-235. The site is still an active factory where plastic automobile parts, such as door panels and dash-boards, are extruded and finished.

During production of the uranium fuel elements, waste material from the extrusion process mixed with oil from the machinery. This mixture subsequently flowed into the oil drainage system contaminating over 1000 ft of buried drain-lines with varying amounts of uranium tainted oil. In order to quantify the extent and degree of this contamination and to conduct post-remediation measurements, the DOE FUSRAP hosted a demonstration of the Pipe Explorer system.

SEA conducted surveys at the site on two separate occasions. The first occurred in April 1995 and the second in May 1995. Thirteen surveys were carried out in eight drain-lines. Several lines were surveyed more than once to confirm success of remedial actions. Two Pipe Explorer deployment systems were used with 3 different radiological sensors. The first system used during the April demonstration was a manually operated system. Deployment with this system is controlled by a hand crank. Figure 2 shows the system in operation at the site.

Fig. 2

With this system, the detector is deployed to a specified location where the position of the detector and its output are recorded by the operator. Figure 3(a) shows data from one of the surveys conducted with the

manually operated Pipe Explorer system in conjunction with a beta detector. The data was taken prior to any remedial actions. Thus, the drain-line had a substantial amount of thick oily sludge in it (about the consistency of peanut butter). The detector and its tether were successfully deployed and retrieved with none of the oily contamination coming into contact with the detector, tether, or workers. The data in Fig. 3(a) was obtained with a detector designed and calibrated by the DOE Grand Junction Projects Office Radon Laboratory (1).

For the second stage of the demonstration carried out in May 1995, the automated Pipe Explorer system was used with a higher sensitivity beta detector. The system canister includes a motorized reel and a deployment distance measurement sensor. Additional sensors in the canister such as a slack indicator, a tension meter, and pressure transducers enable the system to run with minimal operator interaction. All outputs from the sensors are displayed on a control panel. In addition, they are recorded and displayed on a laptop computer acting as a virtual instrument through a LabView program. The radiological data is also recorded on the laptop so that surface activity as a function of distance into the pipe can be monitored in real time. Figure 4 shows the automated system in use at a DOE site.

A sample of the data obtained with this system is shown in Figure 3(b). The actual drain-line begins at a distance of 27 ft. Since access to the drain-line was obtained through a deep manhole it was necessary to construct a conduit of this length to guide the membrane to the drain line entrance. The structure of this data shows the utility of a continuous survey. The data shows a small amount of contamination up to the 40-ft mark in the drain-line. At this point the line intersects another drain line which had been thoroughly cleaned. After the intersection, however, substantial contamination was encountered. The only exception was a relatively clean section between 90 and 100 ft.

Fig. 3

Fig. 4

Confirmation of the Data

Data obtained with the Pipe Explorer system at the FUSRAP site was verified with several methods. The first was purely qualitative, where the membrane was visually inspected as it was retrieved from the drain-line. This was useful in such instances as shown in Fig. 3b where the data showed significant structure. For example, a large amount of the oily sludge was noted on the portion of the membrane that had traveled 100 to 120 ft into the drain-line. The portion of the membrane around 98 ft had virtually no oil on it, but below 90 ft substantial amounts of the oily sludge were again seen on the membrane.

Another validation method used was to measure the activity of contamination adhering to the membrane as it was being retrieved. Measurements were taken with a conventional pancake GM probe. This data is shown as triangles in Fig. 3b. The distance accuracy for these measurements is substantially less than the accuracy of the Pipe Explorer data (pancake meter data accurate to approximately 2 ft, Pipe Explorer accurate to 1 in). Surface activity measured with the Pipe Explorer is consistently higher than that measured with the pancake GM probe because the Pipe Explorer system measures the contamination in the pipe and the pancake GM probe measures only the contamination that adheres to the external surface of the membrane. Furthermore measurements with the

pancake probe are not calibrated for attenuation effects of the membrane, whereas the data obtained by the Pipe Explorer system is.

Confirmation of the data was also attempted by pushing a small GM detector into the drain-line. However, contamination adhering to the GM probe assembly tended to obscure the measurement of contamination on the pipe wall.

Detector Calibration

The ideal way to confirm the Pipe Explorer system data would have been to excavate a portion of a drain line and have it analyzed. However, the motivation for using the system at the FUSRAP site was to avoid excavating drain-lines. Therefore, confidence in the data was obtained through rigorous calibration of the detector.

Detectors used with the Pipe Explorer system are specifically calibrated for each use. They are calibrated with an isotope of similar energy of the contaminants that are suspected in a pipe and calibrated in the same measurement geometry. For example, U-238 was suspected at the FUSRAP site. Therefore, Sr-90 was used as a calibration source (U-238 is not available in sufficiently high activities for calibrations). The daughter product of Sr-90 (Y-90) emits a beta particle with similar energy as the dominant U-238 daughter product, Pa-234m. The Sr-90 calibration source has a known activity traceable to the National Institute of Standards and Technologies. Using this calibration source results in slightly elevated detection efficiencies because of a lower energy beta emitted by Sr-90 (546 keV max.). This emission is more heavily attenuated by air and the membrane material than the higher energy beta from Y-90, but no effort was made to determine this difference. The significant added cost of assessing this effect on the calibrations was not deemed necessary, since the error was not considered significant (on the order of 20 percent) and results in conservative measurements.

The calibrations were carried out to best simulate the measurement conditions that would be encountered at the FUSRAP site, where the detector rests on the bottom of a 4-in pipe inside of a 4-mil polyethylene membrane. Therefore, all of the calibration measurements were made through a sample of the membrane material in 4-in pipe. The fundamental procedure used in the calibrations was to move the calibration source to various grid locations surrounding the detector and determine the probe response at each location. The response of the detector to the Sr-90/Y-90 source was integrated over all angular and axial positions to determine detector response to distributed contamination inside of 4-in pipes. The response of the detector to a check source in a fixed geometry was recorded immediately before and after the detector calibrations. The check source measurement was repeated prior to and after each drain-line survey at the FUSRAP site to verify the detector performance had not changed since the calibration.

BENEFITS

The use of the Pipe Explorer offers many technical benefits. These include;

- Video surveys
- 100% gamma and beta surveys of pipe interiors, even in buried pipes.
- 100% alpha surveys of pipe interiors (available soon)
- Detector does not become contaminated
- Removable contamination is not spread along pipe.
- Personnel exposure significantly reduced.
- Immediate results.

Technical benefits such as the ones listed above for the Pipe Explorer are usually heralded as the pay-off for a DOE investment in a new technology. However, the primary reason the DOE provides funding for development of environmental technologies is so that economic benefits will result through more expedient and cost effective methods. Substantial cost savings have already been realized from use of the Pipe Explorer system at the FUSRAP site demonstration. These cost savings to the DOE are nearly three times the amount invested in the development of the Pipe Explorer system.

The DOE FUSRAP recognized that the cost of excavating buried drain-lines at the site in Adrian, MI would be substantial. Therefore, they developed a methodology to avoid these excavation costs. The plan was to verify that activity levels of contamination in the pipes were below a criteria level of 7x105 dpm/100cm2 (averaged over the length of the drain-lines). It was determined through a hazard assessment that such levels of contamination posed no threat to the general population. Thus,the drain-lines could be left in place after filling them with grout to seal the contamination. If surface activities were found in excess of the criteria level then the drain-lines were to be flushed and cleaned prior to grouting.

The initial method used to characterize the drain lines was to insert a small geiger-mueller (GM) detector directly into the drain lines. This was soon found to be an ineffective method because of the abundance of removable contamination. The contaminated oil would adhere to the detector, making it difficult to differentiate between measurements of contamination on the pipe walls and contamination directly on the detector. In addition, only limited lengths of the drain-lines could be accessed since in many cases the detector could not be shoved around elbows.

The benefits of using the Pipe Explorer over direct insertion of a detector were readily seen in surveys of one of the drain-lines. Data initially obtained with the manually operated Pipe Explorer system from this drain line showed surface activity levels in the pipe in excess of the 7x105 dpm/100cm2 criteria level. The drain-line was then cleaned and a subsequent survey was conducted. Activity levels were found substantially reduced with the exception of a hot spot near the beginning of the drain-line. A detector manually inserted into the pipe would have come into contact with this hot spot and measurements through the rest of the drain-line would have been inaccurately high. Therefore, the Pipe Explorer system provided accurate results showing that the drain-line was within the criteria level. Similar results were obtained in the other seven drain-lines surveyed.

Had data not been available from the Pipe ExplorerTM system there would not have been an accurate way to assess activity levels in the drain-lines. Therefore, it would have been necessary to excavate them. It is estimated that the costs to excavate the drain-lines would have been on the order of $1.2 million (2). However, this estimate neglects the fact that the site is an active automotive parts factory. Therefore, costs associated with plant impacts and relocating factory operations should also be included. Factory personnel have current estimates of these costs from prior experiences of modifications to the plant. Their estimate of these costs are about $0.8 million. The cost savings were diminished somewhat by the expense of cleaning the drain-lines and disposing of the

waste generated from the cleaning. This cost is estimated at $0.5 million. Therefore, the net savings is estimated to be;
$1.2 +$0.8 -$0.5 = $1.5 million

FUTURE ACTIVITIES

The development of the basic Pipe Explorer system which includes gamma and beta detection capability is nearing completion. The final aspect of this phase of development was to demonstrate the system at another DOE site during November 1995. Video inspection capability of the system was demonstrated along with radiological surveys. Significant cost savings ($0.5 M) to the DOE were also realized from this demonstration of the system.

The Pipe ExplorerTM system is now available for service work as an inspection tool. A great deal of interest has already been expressed in using the system at;

- Rocky Flats
- Los Alamos National Laboratory
- Sandia National Laboratory
- Other FUSRAP Sites
- and Argonne National Laboratory

In July of 1995 the development of an enhancement to the system was funded by the DOE METC. This will enable the system to be used for detecting low levels of alpha emitting contaminants such as Pu-239. This will be accomplished by making the inverting membrane component of the system an alpha sensitive scintillator. A photo-detector, towed through the membrane, much the same way as gamma and beta detectors, will quantify activity levels as a function of length over 100% of the internal surface area. After this enhancement is added to the system, complete alpha/beta/gamma surveys will be possible with the Pipe Explorer system.

ACKNOWLEDGMENTS

SEA wishes to thank Mr. C. Edward Christy of METC for his diligent and enthusiastic support as COR of the Pipe Explorer system development. SEA also wishes to thank Mr. Al Tardiff of EM-50 who has helped in finding demonstration sites for the system. We also thank Mr. James D. Kopotic of the FUSRAP for agreeing to host the demonstration of the Pipe Explorer system. Finally we wish to thank Mr. Marty Keller and Mr. Gil Drexel of Bechtel along with the site crew at the FUSRAP site who were all extremely helpful in the successful demonstration of the system.

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RADIOLOGICAL CHARACTERIZATION OF WASTE UNDER THE DOE WASTE MORATORIUM

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ABSTRACT

In response to the release of radioactively contaminated wastes from DOE facilities to non-licensed commercial Treatment, Storage, and Disposal (TSD) facilities, the Office of Waste Operations (EM-30) issued a

moratorium on DOE shipments of hazardous and toxic waste that had the potential for radioactive contamination. To ship waste under the moratorium each facility is required to defend their characterization and certification programs against the requirements of the EM-30 Performance Objective for the Waste Moratorium (PO) (DOE 1994b). The PO defines Radioactive Material Management Areas (RMMAs), and requires waste characterization for radiological contamination both at the surface and in volume. These actions have resulted in an increase in the volume of waste managed as radioactively contaminated.

This report presents a methodology for certifying radioactive waste characterizations in compliance with the requirements of the PO. The methodology addresses the unique definitions of radioactive and mixed waste presented in the PO as well as the logistics of establishing and managing RMMAs. Recommendations are made for using existing waste management and radiation control documentation to develop a documentation package capable of defending characterizations of "no radioactive waste present". Proper implementation of the methodology presented can significantly reduce the volume of radioactive and mixed waste generated at DOE sites.

HISTORY

In order to implement a successful program for certifying the radioactive status of wastes, it is important to understand the history behind the Department of Energy (DOE) waste moratorium and the affect it has had on the DOE complex.

What is the DOE Waste Moratorium?

The DOE Waste Moratorium (moratorium) is a one page memorandum issued on May 17, 1991, by the Office of Environmental Restoration and Waste Management, Office of Waste Operations (EM-30). This memorandum called for a freeze on shipments of hazardous waste as defined by the Resource Conservation and Recovery Act (RCRA), and toxic wastes as defined by the Toxic Substances Control Act (TSCA) which had the potential for being radiologically contaminated. The memorandum required the freeze to remain in effect until a DOE-HQ Moratorium Review Team could review and approve the site's, programs, and procedures for certifying "No Radioactive Waste Present." This approval is based on the facility being able to meet all of the criteria established in the PO. The PO is a series of requirements defining what is required to defensibly characterize and certify the radiological nature of wastes.

Why was the moratorium established?The moratorium was established as a result of the lack of "de minimis" standards (amount of radioactivity below which there is no concern for disposal) for radioactivity in waste. Because there is no de minimis standard each facility within the DOE complex was left to establish their own waste release criteria.

From 1980 to 1991, the DOE shipped hazardous waste containing low levels of radioactive materials to commercial treatment, storage, and disposal (TSD) facilities which were not licensed to receive radionuclides. This was discovered when ash from the Rollins Incinerator facility in Baton Rouge was monitored and found to be radioactive. The ash was generated from shipments of uranium-contaminated waste from the Oak Ridge and Savannah River sites. These shipments occurred due to the use by these sites of self-generated standards which defined waste containing low levels of radionuclides as non-radioactive. The self generated standards had not been formally approved by DOE. DOE acted to end this practice by imposing the moratorium on the off-site shipment of all DOE hazardous and

toxic wastes which had the potential to contain radioactive material until the programs and procedures of each field installation could be reviewed and approved of by DOE-HQ.

What have been the effects at DOE Sites?

The moratorium established a new area designation called a radioactive materials management area (RMMA). A RMMA must be established around any location which has the potential for generating radioactively contaminated waste. Most DOE sites have taken a very conservative approach in designating RMMA boundaries. All wastes from RMMAs are defined as potentially radioactively contaminated or suspect radioactive. That results in managing nonhazardous waste (e.g., paper, plastic, glass) as radioactive waste, and hazardous waste (e.g, waste lead, waste solvent, corrosives) as mixed waste. For most DOE sites the amount of mixed and low-level radioactive waste generated increased greatly after imposition of the moratorium.

The impact from the increased mixed waste volumes has been severe due to the limited capacity for mixed waste treatment and disposal. Without the ability for disposal, mixed waste volumes in storage have increased greatly. Mixed wastes are regulated by the RCRA as a Land Disposal Restricted (LDR) waste. LDR wastes can be stored for one year only. Many regulating agencies have granted extensions to DOE sites for continued storage of mixed waste beyond one year, however, these extensions are not indefinite. Regulators can elect to cancel these extensions and cite facilities with storage violations. These violations can result in fines of up to $10,000 per drum, per day.

This report provides a methodology for operating in compliance with the PO, while reducing the volume of waste managed as suspect radioactive.

PERFORMANCE OBJECTIVE REQUIREMENTS

In order to have the moratorium lifted, a site must show compliance with the requirements established in the PO. This report does not discuss all of the PO requirements, but, rather focuses on those requirements directly impacting radiological waste characterizations.

Performance Objective Definitions

The PO has redefined several terms commonly used within the waste management industry and added several new concepts for managing radioactivity in waste. The following definitions are now applicable at DOE sites and all procedures using these terms must utilize the PO definitions.

The PO definition of radioactive waste defines two aspects of contamination that must be addressed for all radioactive waste determinations. A radioactive waste is any waste managed for its radioactive content, which is not otherwise regulated for that content (e.g., emissions regulated by the Clean Air Act, and effluents regulated by the Clean Water Act). If nonradioactive material is used in a process, the resulting waste is not radioactive as long as no measurable increase in radioactivity is found above background in volume or bulk (at a statistically defined confidence interval); and the waste contains no surface radioactivity above limits established in DOE Orders or guidance. It is important to note that the surface criteria has not changed, it is only the definition of contamination in volume that presents a new challenge.

A mixed waste under the PO is a radioactive waste as defined above which is also regulated by either RCRA or TSCA. Radioactive TSCA waste, although not a mixed waste by regulation, is defined as such by the PO.

Most DOE sites have taken this concept one step farther, by managing all wastes and materials in accordance with PO criteria.

A RMMA is an area in which the potential exists for contamination due to the presence of unencapsulated or unconfined radioactive material or exposure to high energy beams or particles (neutrons, protons, etc.) capable of causing activation.

Unrestricted release (i.e., free release) is a release of property (e.g., waste), based on a formal, documented decision reflecting risk-based standards, licensing considerations, and associated implementing procedures, that the property may be utilized, treated, or disposed of by any party without concern for radioactive content. The PO definition has specified treatment and disposal for unrestricted release due to the potential for concentrating radionuclides during these processes. This definition arose from the incident at the Rollins incinerator.

Basic Principle of the Performance Objective

The PO is a lengthy and complex document, but, it can be reduced to one basic principle. This principle is that mixed waste, as defined above, is not to be shipped off-site to a facility unless that facility is specifically licensed for the receipt of the radioactive component of the waste.

Three Cases of the Performance Objective

To ensure that the basic principle of the PO is met, three cases (i.e., waste management scenarios) have been defined for the management of wastes. In the first case the waste is determined to be a non-radioactive hazardous waste and is shipped off-site for treatment and disposal. If the waste is determined to be a mixed waste it can be managed, treated, and disposed of at DOE sites as a second case. If no capacity exists within the DOE for the treatment and disposal of mixed waste, the third case allows the waste to be shipped to a licensed off-site commercial facility for treatment; then the residual is disposed of in a licensed and/or permitted commercial disposal facility or returned to DOE for treatment/disposal. If the third case is selected the site securing the vendor must obtain an approved DOE Order 5820.2A, Radioactive Waste Management (DOE 1988), exemption for use of the facility.

Determining the Radioactive Status of Waste

In order to select a case to operate under the site must determine the radioactive status of their waste. This characterization can be made through process knowledge or survey, sampling, and analysis. Survey techniques are widely used for determining surface contamination, however, their application for characterizing volume contamination can be impractical, especially for heterogenous wastes. Process knowledge is often the only method available or practical for characterizing volume contamination. If process knowledge is selected as the characterization method it must be accomplished in accordance with PO requirements. These requirements have been established to ensure that the quality of process knowledge is adequate to certify that no radioactive waste is present.

Process Knowledge

A generator must show knowledge of three critical elements in order to use process knowledge for radioactive waste determinations under the PO. These critical elements are knowledge of waste origin, use, and potential for exposure.

Critical Elements

Knowledge of waste origin is defining the radioactive nature of the materials involved with waste generation. Were the materials used to

generate the waste radioactively contaminated? Knowledge of use goes one step farther to evaluate the waste generating process. If radioactive materials are present in an operation, does the process contaminate the wastes generated? Finally knowledge of potential for exposure involves defining the areas where the waste is generated and stored. Was the waste generated or stored in an area where a potential for contamination existed (i.e., in a RMMA)?

A REALISTIC METHODOLOGY FOR PROCESS KNOWLEDGE

Now that we have an understanding of the moratorium and the PO we will present a waste characterization program addressing PO criteria while supporting efficient operations. The program utilizes the waste management and radiological control practices routinely used at DOE sites to provide the process knowledge and documentation necessary to certify the radioactive status of waste.

Knowledge of Origin

The DOE Radiological Control Manual (RadCon manual) (DOE 1994a) establishes requirements for the identification and labeling of radioactive materials. These requirements are intended to maintain radiation exposure to workers and the public as-low-as-reasonably-achievable (ALARA). This control of radiation and radioactive materials forms the basis of the generator's knowledge of origin. If no radioactive materials were used in a waste generating process and the waste was not subjected to radiation of an energy capable of causing activation, the waste is not radioactive. For most routine operations knowledge of origin is already established through RadCon manual requirements for identification and labeling of radioactive materials.

Sometimes wastes are generated from materials whose origin is uncertain. Decommissioned facilities, equipment, and weapon components usually have insufficient documentation to certify the waste origin as non-radioactive. If this is the case survey and assay techniques for surface contamination can be used to show impermeable materials are non-radioactive. However, if the waste is a permeable substance or has a potential for volume contamination, a sampling and analysis approach approved by EM-30 or an approved area office is required for characterizing the waste. Although the potentially volume contaminated material requires additional analysis, survey results showing surfaces below release levels can be used to release secondary wastes (e.g., gloves, laboratory wipes) generated during management of the material.

Knowledge of Potential

Knowledge of Potential can be established simply by identifying the RMMA status of waste generation and storage locations. Remember, a RMMA is defined as an area where the potential for radiological contamination exists. If a waste can be generated and managed outside of RMMAs it has never had the potential for being contaminated and can be declared a non-radioactive waste. It is therefore, beneficial to reduce the size of RMMAs as far as practical. Many DOE sites have established extremely conservative RMMA boundaries, some extending around entire facilities. The large size of many RMMA boundaries is often due to complicated and onerous programs developed for establishing and maintaining these areas.

Knowledge of Use

Information from radiological control and characterization programs and the RMMA status of waste management locations can be used in conjunction with procedural evaluations to establish knowledge of use. To properly apply knowledge of use, generators must understand the behavior of the

radionuclides involved in their processes. For example, tritium, a radioactive form of hydrogen, is capable of contaminating hydrogenous materials, via molecular exchange, without noticeable physical contact. To ensure generator understanding, an approval process for waste generators should be established providing instruction on the PO requirements. Once trained, approved generators, should be able to defend their knowledge of use.

A REALISTIC APPROACH TO RMMA MANAGEMENT

Establishing RMMAs

A formal program should be developed for establishing RMMAs. The radiation control group for a site is usually best prepared to establish RMMA boundaries. The process used for establishing RMMAs should be simple and support RMMA boundary changes as needed.

In order to determine whether a RMMA will be necessary for a given location, two questions must be answered. The first question is, does this location have the potential for activating waste? This question can be answered in a two step approach by first asking the generator if the location generates photons in energy above 7 millielectron-volts (MeV), has a neutron source, or has a particle accelerator present. If the generator responds positively to these questions, then the location and operations should be reviewed by a health physicist to evaluate the source and potential targets for activation potential. The second question is, are there any unencapsulated or unconfined radioactive materials present at the location? This question can be easily answered by a trained generator. If an activation potential exists or there is unconfined radioactive materials present, a RMMA is required. If the area was previously a RMMA, but no longer meets the RMMA criteria, area surveys can be used to release the location. If surveys show that the area is free of contamination above DOE Order 5400.5, Radiation Protection of the Public and the Environment (DOE 1993), release limits then it can be determined that no potential exists to contaminate waste. By making the RMMA determination criteria easy for generators to understand, RMMA boundaries can be established appropriate for the operation. Area surveys in conjunction with documented answers to the questions presented above are adequate to release a location from RMMA status. This approach is more efficient than complex management plans that require extensive paperwork on the part of the generator. With smaller and more realistic RMMA boundaries the volume of radioactive waste generated can be greatly reduced.

Posting RMMAs

The posting used for a RMMA identifies the area as one where the potential exists for radiological contamination of waste according to the PO definition. The posting should not be interpreted as a warning for personnel exposure. As stated earlier, there is no de-minimis quantity for radioactivity in waste. The RMMA is established for levels of potential contamination well below the level where personnel risk is an issue. Radiologically controlled areas defined by the RadCon manual are established to control exposures to personnel. RMMAs are established to minimize the amount of waste managed as radioactive. For this reason postings established for RMMAs should contrast visually from RadCon manual control area postings.

Operating in RMMAs

In order to perform work in a RMMA, a generator should be trained on the requirements of the PO and waste minimization planning. Because the RMMA

has the potential for contaminating waste, it is important to minimize the amount of expendable material taken into the area. Anything that has entered into a RMMA must be managed as radiologically contaminated unless a justification can be made that the item is clean. Developing defendable justifications takes time and resources therefore, it is more advantages to reduce the volume of waste generated in a RMMA.

Release of Materials/Wastes from RMMAs

In order to release material from a RMMA as "Not Radioactively Contaminated" a justification must be documented and readily retrievable. The justification for process knowledge determinations must demonstrate knowledge of origin, use, and potential for exposure.

Release of material from one RMMA to another is not considered an unrestricted release and therefore surveys are not required. Likewise when moving a material from a Non-RMMA to a RMMA no survey is required. However, when bringing material into a RMMA previous determinations that the material was not contaminated are placed in doubt.

Surveys can be used to release non-permeable items that have no potential for volume contamination. If items are moved across RMMA boundaries routinely, survey data can be accumulated and used to release the process. For example, a tool is used for the same operation and is moved across the RMMA boundary, if survey data is sufficient to statistically show that the item does not become contaminated in the process, the tool can be moved across the boundary without survey. Verification surveys (i.e.,surveys conducted on a scheduled basis) should be used to validate previous process characterizations. Verification survey frequency can be chronological (e.g., weekly, monthly) or process based (e.g., one survey for every 20 processes).

Waste Disposal

The moratorium was established to ensure that no mixed waste is sent to an off-site TSD facility unless that facility is specifically licensed to manage the radionuclides present in the waste. At DOE sites a few general rules of thumb will ensure that this criteria is met. All wastes generated inside of a RMMA should be disposed of in containers located within the RMMA and wastes generated outside of a RMMA should be disposed of in containers located outside the RMMA. Waste containers should be labeled to identify the RMMA status of the location and wastes should be removed from a location before RMMA boundaries are modified.

DOCUMENTATION

Proper documentation of process knowledge is critical for securing certification that wastes are not radioactive. The following forms are routinely used at DOE sites and can be modified to meet the criteria of the PO.

Origination Label

An origination label should be used to identify whether the material has been in a RMMA. This label identifies the materials point of origin as either a RMMA or a Non-RMMA. The label should be attached to all items coming from RMMAs unless another form of container documentation provides this information. For Non-RMMAs the label can be used to certify materials as not radioactively contaminated. The origination label can be issued by anyone with authority over the waste.

Radiation Survey Form

In order to free release a waste it must be shown to have no radioactive contamination on the surface and in volume. The surface can be defended as clean through surveys. These surveys must be recorded on a form to

document the determination. Proof that wastes are not contaminated in volume must also be documented. The generator should record and certify process knowledge on the survey form, (e.g., a container of hazardous solvent was not opened in the RMMA). Now the information necessary to make the determination that the material is "Not Radioactively Contaminated" is present in one document. The waste container number should be included on the radiation survey form to facilitate retrieval.

Free Release Label

A free release label is a certification that accompanies waste determined to be non-radioactive. The label can be issued when survey results show that surface contamination is below DOE Order 5400.5 release levels and justification is given that the volume is free of contamination. The label is an essential piece of documentation and should be protected. A free release label should only be issued by the group responsible for offsite certification. An origination label certifying that the waste was generated outside of a RMMA can be used in stead of a free release label to certify that the waste is not radioactive.

Surface Release Label

A surface release label is used to identify wastes which meet DOE Order 5400.5 surface release criteria, but have the potential for volume contamination. The surface release label allows a waste to be moved onsite for controlled release. The surface release label cannot be used to release a material offsite or for intrusive processing. Only the radiation control group should issue surface release labels.

Radioactive Material Label

The radioactive material label is used to identify equipment, components, and items that are radioactive, potentially radioactive or have been exposed to radioactive contamination or activation sources. They are also required for sealed and unsealed radioactive sources or associated storage containers. The radioactive material label must meet the RADCON labeling requirements to ensure that items are not accidentally released or processed as non-radioactive. The radioactive material label should be issued by anyone with authority over the item.

Container Inventory Form

Many DOE sites use a container inventory form to document the contents of a waste package. The inventory, in addition to providing adequate information on the make up of the waste, should identify the RMMA status of the waste generation location. In fact the RMMA status for the location of the container should be completed before any waste is placed in the container. The RMMA status on the inventory ensures that radiological characterizations are adequately documented for each waste container.

Waste Disposal Form

Many DOE sites also use waste disposal forms to provide communication between waste generators and waste management personnel. The waste disposal form usually identifies the labeling requirements for the waste package and provides a place for the generator to certify that the waste placed in the drum was approved for that container. This form like the inventory sheet should identify the RMMA status of the generation location. A disposal form is usually completed for each waste container shipped offsite for disposal. Many times wastes are compiled into one drum from several other containers. That is why it is necessary to identify the RMMA status on the container inventory sheet. From the

container inventory sheets, personnel responsible for drum disposal can accurately determine the RMMA status of the disposal drum.

Waste Shipment Checklist

A waste shipment checklist should be used to identify the containers sent under a given manifest, ensure that certification criteria are met, and verify that certification documentation has been compiled. The checklist provides a vital link between the information sent with a waste shipment and the supporting documentation maintained at the site. If the checklist is constructed properly, certification documentation can be maintained by different groups at different locations within a site and still be easily retrieved. The checklist should confirm that all non-radioactive waste packages have documentation showing the waste was generated outside of a RMMA or justifying the non-radioactive determination based on knowledge of waste origin, use, and potential for exposure. If the waste is radioactive the checklist should reference the radiological characterization data showing the waste is within offsite TSD facility license restrictions.

Traceability of Rationales

The PO requires traceability of each waste container certification to the documented rationale for the decision that no radioactive waste is present. Traceability can be accomplished through the compilation of a shipment file. The shipment file should include all paperwork specific for a waste shipment, such as a copy of the manifest, waste shipment checklist, waste disposal forms, waste container inventories, origination labels, and free release labels. The waste shipment checklist provides a listing of all containers shipped under a specific manifest. For each container, a file of all container inventories, disposal forms, and labels should be compiled. The container inventories will identify the RMMA status of the location where the waste was generated. All free release labels associated with the container provide certification of "No Radioactive Contamination" and radiation survey forms provide the rationale used to free release containers generated in RMMAs.

In a scenario where a single drum is in question, such as a drum found off-site, or a TSD facility radiation alarm is set off, certification personnel can retrieve survey data, process knowledge rationales, and certifications generated throughout the waste management process defending the radiological characterization.

CONCLUSION

The basic methodologies presented here, when properly implemented, can greatly reduce the volume of suspect radioactive waste generated and can provide a mechanism for better controlling the radiological status of materials. The approach utilizes standard conventions used throughout the DOE to provide the documentation necessary to defend radiological characterizations. This approach cannot be used for all wastes and some survey sampling and analysis will be required for non-routine waste, however, the majority of process waste can be effectively managed under this methodology.

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U. S. DEPARTMENT OF ENERGY NEVADA OPERATIONS OFFICE (DOE/NV) LOW-LEVEL WASTE (LLW) DISPOSAL COST ANALYSIS

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ABSTRACT

The Nevada Test Site (NTS) is a major waste disposal site for low-level radioactive waste generated by DOE installations. Approximately 22,300 cubic meters (796,625 ft3) per year of LLW have been disposed at the NTS in the last four years, placing it among the waste volume leaders for radioactive waste disposed in the United States. The waste disposal costs at the NTS are significantly lower than the other DOE and commercial disposal sites around the United States.

A waste generator fee for disposal of LLW at the NTS has been used since 1978. However, the generator fee does not provide a full cost recovery for all functions of LLW disposal at the NTS. Additional funding is obtained from DOE/Headquarters (HQs). This paper addresses the costs associated with LLW disposal at the NTS and the factors that keep the costs minimal. Some of the factors that will be discussed are operations efficiency and productivity savings. The factors do not hinder the state regulator views or the safe operation of the disposal site. The DOE/NV costs to dispose of waste at the NTS include all activities (i.e., performance assessment, site monitoring, site characterization, waste acceptance program, etc.) associated with complying with DOE Order 5820.2A, "Radioactive Waste Management," in addition to the actual waste disposal operations.

INTRODUCTION

The DOE/NV has two LLW disposal sites at the NTS: the Area 5 Radioactive Waste Management Site (RWMS) and the Area 3 RWMS. The Area 5 RWMS is located approximately 22 km (13 mi) north of the NTS entrance to Mercury in Frenchman Flat, an enclosed geological basin. The Area 5 RWMS consists of 296 hectares (732 acres), 37 hectares (92 acres) of which are being utilized for disposal of LLW. At the current rate of land utilization and a disposal rate of 11,552 cubic meters (407,901 ft3) of LLW per year, the facility can be expected to provide disposal capacity for 10,884,440 cubic meters (388,733,000 ft3) and will have a life expectancy of more than 900 years. Pits and trenches are used for shallow land burial at the Area 5 RWMS.

Fig. 1

The Area 3 RWMS occupies an area of approximately 50 hectares (125 acres) and is approximately 38 km (23 mi) north of Mercury and 17 km (10 mi) north of the Area 5 RWMS. Adjacent subsidence created from underground nuclear weapons tests conducted at depths well above the groundwater table are used as waste disposal cells for shallow land burial. The active disposal cell, U3ah/at, has an available volume capacity of

approximately 169,901 cubic meters(6,000,000 ft3). The remaining disposal capacity for the Area 3 RWMS, which includes three additional subsidence craters, is 1,745,767 cubic meters (62,348,832 ft3). The average disposal volume for the Area 3 RWMS is 11,694 cubic meters (412,906 ft3) and a life expectancy of 151 years.

Fig. 2

Fig. 3

SITE CHARACTERIZATION

The RWMSs are located in closed basins which receive the lowest precipitation of any LLW disposal site in the country. The disposal sites have the thickest unsaturated zones (deepest water tables) of all sites and are located in the most remote sites with respect to nearby population. The water table at the Area 5 RWMS is approximately 240 m (770 ft) below the surface. The Area 3 RWMS is located on an alluvial plain, and the water table is approximately 500 m (1600 ft) below the land surface.

COST ANALYSIS

Costs Associated With LLW Disposal

In order to determine the incurred and estimated costs for disposal, costs were categorized. The categories consist of: Programmatic costs (DOE regulatory activities, permits, other outside agencies, and required monitoring); Operational costs (direct activities of putting waste in the ground such as pit design and construction, maintenance of facilities, and security); and Program Support costs (program management, infrastructure).

Programmatic Costs

Costs to dispose of LLW at the NTS include all activities associated with DOE Order 5820.2A, "Radioactive Waste Management," which "establishes polices, guidelines, and minimum requirements by which DOE manages its radioactive and mixed waste contaminated facilities." Waste management activities associated with this order include the following:

Provide characterization and engineering data for the systematic analysis of the potential risks posed by the waste management site to the public and the environment (i.e., Performance Assessments);

Establish waste acceptance criteria and audit any waste generating organization that ships waste to the NTS;

Assure that quality assurance activities are established and implemented for all waste management activities; and

Establish and maintain an integrated database program for all types of waste under DOE/NV purview.

Other activities which are part of programmatic costs include the LLW portion of the DOE/NV Site-Wide Environmental Impact Statement and site monitoring. State regulator involvement in LLW disposal is very proactive. The Nevada Division of Environmental Protection (NDEP) is actively involved in assuring that no mixed waste enters the state for disposal. The NDEP is confident that the LLW disposed at the NTS is free of hazardous constituents due to the strict and rigorous DOE/NV Radioactive Waste Acceptance Program.

Operational Costs

Activities related to the actual cost of burying waste consist of the following:

Engineering, design, and construction of new pits and trenches in the Area 5 RWMS and modifications to the subsidence craters in the Area 3 RWMS;

Waste handling capabilities for the Area 3 and Area 5 RWMS;
Health physics support for the facilities;
Purchasing equipment and supplies for RWMS operations;
Maintenance of existing facilities and roads;
Engineering, design, and construction of support facilities; and
Provision for specific user fees which include feeding, housing, janitorial, fire protection, and electrical power.

Program Support Costs

Activities which provide support for the actual burial of the waste include the following:

Overseeing budget responsibilities for disposing of waste and establishing a fee to recover the costs of disposal;
Generating budget requirements and reports for DOE/HQs; and
Preparing updates of waste management plans and procedures.

Generator Fees

A waste generator fee was established in order to recover incremental costs (disposal-dependent) associated with waste disposal at the NTS. The waste generators are required to provide to DOE/NV a "Three-Year Waste Shipment Forecast"; this forecast is updated on a quarterly basis. The generator issues a purchase order to the disposal contractor based on an estimate of the amount of waste expected to be shipped. In previous years, the generator fee only paid for the direct disposal of the LLW. In Fiscal Year 1995, DOE/NV received permission from DOE/HQs to add to the fee structure, a broader range of activities associated with complying with DOE Order 5820.2A.

COST COMPARISON

Attempts have been made in the past (Salomon, 1993) to compare the costs of LLW disposal facilities. This is not an easy task since all disposal facilities are designed differently, serve different functions, and are located in different geologic environments. Salomon attempted to compare proposed facilities located in two distinctively different environments: one facility was proposed as a earth mounded above grade vault in a humid site with a projected high volume capacity; the other facility was a below grade concrete canister technology in an arid site with much lower volume capacity. Life cycle costs varied by a factor of three, and unit costs were nearly 20 percent different. He found similar results for foreign LLW facilities. His conclusions were that significant differences in costs exist because of the variable environments in which the sites are located and the different designs which were planned.

Comparison With DOE Disposal Facilities

Similar differences exist when comparing costs of the NTS sites to other DOE and commercial facilities. A cost comparison study was initiated by DOE/HQ in FY 1995; this study compared LLW disposal costs at major DOE disposal facilities. The comparison study consisted of collecting FY 1994 and projected FY 1995 standard operational costs at six DOE disposal sites. The standard costs compared were: putting waste in the ground, routine maintenance, required monitoring, regulatory permits and other outside regulatory activities, DOE regulatory activities, Field Office program control and oversight, waste acceptance criteria activities, and overhead "taxes." The standard costs were compared to annual LLW disposal volumes and disposal fees. Out of the six disposal sites, the NTS ranked the highest in volume disposed by 313 percent and had the lowest cost per volume by 74 percent.

Table I shows total costs for LLW disposal operations at the major DOE disposal facilities.

Table I

One of the main differences noted in the study was the use of different disposal systems at the DOE sites. Savannah River disposes of LLW in an aboveground concrete vault, and Oakridge disposes of LLW in a tumulus. These engineered structures are more expensive than excavating a pit in the alluvium of the NTS or emplacing waste in a subsidence crater developed from an underground nuclear test. Idaho National Engineering Laboratory, Los Alamos National Laboratory, and the Hanford-Richland sites utilize unlined shallow land disposal for LLW. The primary difference in unit costs compared with the sites that use shallow land burial is the volume of waste disposed. The total cost per cubic meter decreases as the volume increases.

Comparison With Commercial Disposal Facilities

The cost of NTS disposal facilities compare well with the Barnwell facility, a commercial site located in South Carolina, which annually disposes approximately the same volume as the NTS. The comparison confirms that larger waste volumes result in a decreasing cost per unit volume since costs for labor can be spread over these larger volumes. The comparison becomes difficult when the cost per cubic meter is compared to the NTS because Barnwell costs depend on surcharges made on certain shipments of LLW.

OTHER FACTORS

Some of the other factors that keep the NTS costs favorable are controlling the manpower base and performing non-routine tasks with subcontractors. History has proven that as many as 38 shipments (truckloads) of waste can be inspected and off-loaded in one day. An analysis performed at the NTS indicated that an additional 50 percent of LLW can be disposed before it would be necessary to increase the labor force at the RWMS facilities.

CONCLUSION

The LLW disposal costs at the NTS are significantly lower when compared with other DOE sites and commercial facilities. The largest cost inhibitors are the geologic location of the NTS RWMS facilities, the type of LLW disposal utilized at the NTS, and the annual volume disposed. Another cost inhibitor is controlling the manpower base for LLW disposal operations.

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38-38

EXPERIENCES FROM "POST-DRYING" OF SUPERCOMPACTED WASTE FROM THE INTERIM STORAGE BUILDING OF THE NUCLEAR POWER PLANT BRUNSBTTEL, GERMANY

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ABSTRACT
In the drum stores of different NPPs, several drums turned out to be bulged at their top and bottom regions due to the occurrence of drum-internal gas formation. Investigations have shown that, within a storage period of several years, the internal drum pressure can rise up to a value of 4 bar. Apart from nitrogen, mainly hydrogen is contained in the gas. The drums principally affected by these occurrences contain low level radioactive waste (LAW) in a super-compacted form (1).
The repository specifications in Germany stipulate that suitable waste conditioning measures be taken in order to prevent the waste in an effective way from producing gas (2).
In the following, the causes of gas formation in super-compacted mixed waste packages are going to be dealt with. It will clearly be shown that further gas formation can safely be avoided by drying such waste. Following an introduction on the drying process, the experience gained in the drying campaign presently being carried out at the Nuclear Power Plant of Brunsbttel (KKB) will be presented.

INTRODUCTION
Low level radioactive mixed waste from nuclear power plants contains a wide variety of substances, such things as cleaning rags, safety gloves, plastic sheets, insulating material, rubble. The waste often is interspersed with cleaners (water, cleaning alcohol, chlorinated hydrocarbons, etc.). Due to this, a small portion of residual moisture is contained in it.
In order to achieve a waste package suitably conditioned for the repository, this type of waste usually is super-compacted. As the initial waste volume can be reduced up to a ratio of 10 to 1 by that, a considerable reduction of the transport and storage costs is obtained at the same time. The waste, partly preclassified in advance, is filled into steel cartridges, is capped, and subsequently super-compacted. The pellets resulting from this process are then introduced in overpacks (drums of 200 liters or containers) and sealed for transport and storage or repository purposes.
While storing the waste packages that had previously been conditioned this way, gas formation often occurs in the enclosed waste. The affected waste drums show then clearly recognizable bulges at their lids and bottoms. Table I provides information on the drum-internal pressure values and the principal elements which the drum atmosphere consists of.
Table I
It is obvious that the oxygen content of the air enclosed in the drums is being consumed while, at the same time, a large amount of hydrogen is developing.
Inspecting the waste composition inside the drums, it was revealed that the containing matter was damp and there was a penetrating smell of organic solvents.

SOLUTION OF THE PROBLEM

It is the aim to avoid gas formation in the drums and containers filled with super-compacted waste. For this purpose, further investigations have been carried out which, as a first step, were intended to work out the causes of the gas formation phenomenon and, as a second step, were aimed at finding out proper measures to prevent gas formation. The following results have been achieved:

After the drums filled with damp mixed waste had been sealed, first a pressure decrease took place in the drums due to oxygen consumption (oxidative corrosion of metal). Subsequently, a pressure increase could be observed which was due to the formation of hydrogen gas.

Hydrogen gas formation occurs when metals exist in an environment that contains residual moisture. A considerable gas formation rate results when different metals exist, e.g. iron and aluminum, together with small amounts of water and chlorinated hydrocarbons. The mere presence of residual moisture vapors provokes a perceptible formation of gas. The cause of hydrogen formation is the occurrence of electrochemical metal corrosion. The chlorinated hydrocarbons have an accelerating effect on this process.

Oxidative as well as electrochemical corrosion can to a great extent be prevented by drying the waste (i.e. separating the electrolyte). This statement was proved to be correct by the results of the above mentioned tests.

Hydrogen gas formation in the drums is to a satisfactory extent prevented by drying the waste appropriately.

Based on this provable fact, inter alia, the super-compacted waste stored on the grounds of the Nuclear Power Plant of Brunsbttel (KKB) has been subjected to drying procedures since January 1995, which in the following will be reported about.

OPERATING EXPERIENCE

In the interim storage building of the Nuclear Power Plant of Brunsbttel (KKB), a number of drums of 200 liters loaded with super-compacted waste of 10 to 15 years of age are kept in storage for being subjected to a drying procedure. This is due to the fact that some drums are showing bulges caused by internal pressure increase.

The drying procedure is being carried out in batch quantities in a multi-drum dryer. In 1995, thirty waste batches of 30 drums each (a total of 900 drums) have been dried in accordance with the drying procedure described in the following, so that the repository requirements are satisfied. The principals of this procedure are shown in Fig. 1. A summary of plant data as well as some information concerning consumption of energy and service media are given in Table II.

After the lids have been lifted, the drums are brought to the drying chamber. The dryer loading device as presented in Fig. 2 has shown good results in operation. Loading is effected row by row (2x3 drums per row stacked in two tiers). Spacers are placed between the stacked drums. The drying chamber can this way be loaded to a total of 30 drums (Fig. 3).

Table II

Fig. 1

Fig. 2

Fig. 3

Once closed the loaded drying chamber, the drying procedure is started. The recirculation fans force the air enclosed in the drying chamber to undergo a powerful recirculation. Upon starting up the thermal oil system, the air is heated up to a temperature between 100 and 130C.

With the evaporation of moisture-causing substances, the drying process already starts during the heating-up phase. Heat transfer to the waste is mainly effected through convection (direct contact with the carrier gas) and thermal conduction via the drum's shell which is exposed to the flow of heated air. As a result of this, quite short drying periods of 8 to 12 days per batch are achieved. For transferring the evaporated waste components out of the drying process, a part flow of the recirculating air is continuously extracted from the drying chamber and lead via a vapor condenser. This causes the temperature of the previously evaporated waste elements to fall below the dew point. The resulting condensate, after having been subjected to a volumetrical measurement, is lead via a drip collector to a condensate collecting tank. From there it is turned over to the NPP. The dried air is then fed back into the drying chamber. The drying process is controlled by a continuously operating, fully automated monitoring system. For process documentation purposes, all the operating parameters with process relevance are documented by means of a multi-channel recorder.

As far as safety aspects are concerned, the concentration levels of inflammable organic substances as well as of carbon monoxide (CO) in the recirculating air are measured and monitored. It is thus ensured that the drying process is run with a high safety standard.

If the limiting value for inflammable organic substances is exceeded, the plant changes its mode into cooling operation. The electrical heat supply to the thermal oil system is then immediately cut off, while the cooling process for the recirculating air remains active.

Monitoring the CO concentration level is important with respect to detecting a fire caused by spontaneous combustion. If a steady increase in CO concentration is registered, first the inlet temperature of the thermal oil heating circuit will gradually be reduced. If this trend continues to exist, the plant changes its mode into cooling operation. In case that a further increase in CO concentration occurs, an inert atmosphere will be provided for by injecting CO2 as an additional measure.

In Fig. 4, a typical drying history is presented. Shortly after the heating-up phase, an increase in CO concentration takes place in the loaded drying chamber due to the presence of organic mixed waste. As a consequence, the temperature of the thermal oil is reduced. Evaporation of water starts with a certain delay relative to the formation of CO. This can be recognized through the time history of condensate formation. The evaporation of water causes the waste surface to become inert (decrease in oxygen concentration). This leads simultaneously to a decrease in CO concentration. The thermal oil temperature is once again being increased during this drying phase. This accelerates the drying effect. The drying procedure is finally indicated to be concluded when the cutoff criterion has been reached (under 0.5 liters of condensate formation within four hours). This can provoke the formation of some CO at the same time (reduced inert effect). The evaporation rate can be determined from the time course of condensate formation, which in the main drying phase amounts to a value between 1 and 2 liters per hour. The drying procedure is concluded when less than 0.5 liters of condensate arise within a period of four hours.

Fig. 4

SUMMARY

In the past, gas formation phenomena have been observed in connection with the intermediate storage of super-compacted, low-level radioactive waste packed in gas-tight drums. This is due to corrosion processes which are triggered by the complexity of the waste composition associated with the presence of residual moisture.
Investigations revealed that this gas formation can be prevented by eliminating the residual moisture in a drying procedure. Based on this discovery, the super-compacted mixed waste stored In the interim storage building of the Nuclear Power Plant of Brunsbttel (KKB) is being dried in accordance with the repository requirements, employing a compartment drier which is operated with recirculating air.
In 1995, a total of thirty batches, each batch consisting of 30 drums (=900 drums), was conditioned this way in accordance with the repository requirements. Drying periods between 8 and 12 days were reached. The average condensate volume per drum ranges between 4 and 10 liters. When the drying chamber has been loaded, the drying process is being controlled by a fully automated system and permanently monitored with respect to plant safety. The batch record, which is drawn up by means of a multi-channel recorder in the on-line mode, contains the operating parameters with process relevance.
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38-40
DURABILITY ASPECTS OF HIGH-PERFORMANCE CONCRETES FOR A WASTE REPOSITORY
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ABSTRACT
A waste repository for the disposal of low-level radioactive waste is planned at the Chalk River Nuclear Laboratories, Chalk River, Ontario, Canada. It relies greatly on the durability of concrete for the required 500 years of service life. A research program based on laboratory testing to design a durable concrete and assess its long-term behavior was initiated in 1988, and is still in progress. This paper discusses the methodology to assess the long-term behavior of concrete, and some initial observations. Longevity predictions for concrete formulations based on diffusion testing are also presented.
INTRODUCTION
The disposal concept of the low-level radioactive waste repository, planned at Chalk River, relies on the durability of concrete to isolate the waste from the human environment for a minimum of 500 years. The durability of concrete and the corrosion of reinforcements have been studied for some time. Most of the principal factors that reduce the long-term integrity of reinforced concrete elements are well understood (1). Unfortunately, few studies can help predict the longevity of reinforced concrete structures over hundreds of years. Very little information in the literature is helpful in relating in a quantitative way the rate of degradation of concretes subjected to the principal

characteristics of the environment. For these reasons, a research program on concrete durability was initiated by AECL Research in 1987 as part of the licensing support for the repository. The objective of the research program was to assess the durability of a wide variety of concrete types and qualities subjected to different environmental exposure conditions and to develop a high-performance concrete for the waste repository. The program is jointly conducted by AECL Research and the Institute for Research in Construction of the National Research Council (NRC), Ottawa.

METHODOLOGY

Traditionally, concrete durability has been assessed by measuring either the strength or length change of specimens subjected to a corrosive agent. Usually, the corrosive agent is applied at the external boundary of the specimen, but sometimes it may be included within the mix as an internal agent. Results from these types of tests are not applicable to lifetime predictions and are not sensitive to the design geometry of structural members. Tests for sulphate resistance for cements,for example, which involve measuring the expansion of specimens in which sulphates have been integrally included, yield information on whether the cement is suitable for use. These tests give no information on the rate of deterioration. The same criticism can be made of tests of standard laboratory-size specimens exposed externally to a corrosive agent, where failure is indicated by a given loss of strength or a given expansion. The measure of durability in these cases would be strictly relative and insensitive to geometry (2).

The service life of concrete is dependent on a slow rate of deterioration and is influenced by the quality of concrete and the service environment. Factors such as cement type, cement content and water-to-cement ratio can affect the diffusion rate of ionic species into concretes. In addition, service life will depend on the size of specimens and failure criteria adopted. After examining the major failure mechanisms for the repository concrete, corrosion of reinforcement was selected as the mechanism for the failure of the waste repository structure. Chloride ions in the presence of oxygen can initiate corrosion of reinforcement and failure of the reinforced concrete components (3). The failure criteria chosen for the concrete was the time taken for the aggressive ions to reach the reinforcing steel by diffusion through the concrete cover (75mm thick). Based on these criteria, the rates of deterioration and hence an assessment of the longevity of concrete can be made (4 to 7).

The rate of penetration of aggressive ions into the concrete was evaluated by determining the reaction zone front with time of exposure in the solution baths. Prediction of long-term concrete behavior involves the extrapolation of current data, based on the assumption that long-term processes (not currently identified) will not invalidate the extrapolation. The durability prediction chosen for the study was based on the time-dependent depth of penetration of chloride and sulphate ions into the test concretes. Concretes were selected so that the effect of cement blends containing silica fume or blast furnace slag on the diffusion rate of chloride or sulphates in concrete could be investigated and compared with the diffusion in Type 10 cement concrete.

During a post-closure period of hundreds of years, the repository structure will be subjected to various aggressive elements in the environment. For example, the environment inside the repository will be influenced by the chemicals leaching out of the waste, or generated by the waste, whereas the external environment will be influenced by the

changes in the anion and cation content of the precipitation, due to the changes in acid rain and the addition of road salt. On the basis of an analysis of the repository service environment, the following major degradation parameters were selected for laboratory testing of concrete specimens:

- sulphate ions,
- chloride ions,
- leaching of calcium hydroxide by water,
- carbon dioxide reactions, and
- several agents in combination.

EXPERIMENTAL

1. Binders: Portland cement Type 10 and 50, silica fume and blast furnace slag were used for the three concrete systems described in this paper. The nomenclature and composition of cement systems are listed in Table I.

2. Aggregates: An unblended sand consisting mainly of quartz and feldspar was used. The following tests were carried out: an accelerated mortar bar test for alkali aggregate reactivity where samples are placed in 1M NaOH solution at 80 C, a magnesium sulphate soundness test, freeze-thaw cycling and petrographic evaluation. All testing indicated that the sand was satisfactory.

A limestone coarse aggregate of somewhat variable composition was used. Tests conducted were similar to those for the fine aggregate and the results were satisfactory.

3. Concrete: Three concrete systems (S1, S2, S5) were each prepared at Four different water-cement ratios: 0.35, 0.42, 0.5 and 0.60, denoted as mix 1, 2, 3 and 4 (M1, M2, M3, M4).

The cement contents for S1, S2 and S5 are as follows: S1 (M (1-4)): 485, 370, 335 and 280kg/m3. S2 (M (1-4)): 383, 338, 338 and 275kg/m3; S5(M(1-4)): 437, 359, 325, and 259kg/m3. A target slump of 125-150 mm was maintained for all mixes.

Table I

Concrete Specimens

Two concrete prisms, 75 x 75 x 280 mm, were cast for each mix and each exposure condition. S1, S2 and S5 were moist cured for 7, 14 and 28 days, respectively. Prior to immersion in the test solutions, the prisms were coated with wax on all sides but one, to allow a unidirectional ingress of chloride or sulphate ions.

Solution Baths:

Of the 25 baths used in this study, the worst-case scenario for the repository concrete was simulated in the laboratory in baths containing the following aggressive ions and ionic combinations:

NaCl (g/L):	5
Na2SO4 (g/L):	22
CO2:	1 Atmospheric
Temperature:	22oC and 45oC

Methods

1. Sample preparation: Specimens for Energy Dispersive X-ray Analysis (EDXA) were prepared according to the following procedure. A 12.5 mm slice was cut from the concrete prism taken from the test solution. This slice was further cut about 25mm from the exposed surface and then immediately surface dried.

2. Depth-of-Penetration, Profiles determination - C1- and SO4:

a. A Cambridge Stereoscan S-250 was used for the examination and a Tracor Northern TN 5500 Energy Dispersive X-ray Analyser was used for the

quantitative analysis. Calibrations were performed with an obsidian standard containing 0.36% Cl. EDXA analysis on several pieces gave 0.38%, a difference of 5.5%. Analyses were performed in 0.5mm steps, and three profiles for each concrete sample were determined. A value of 0.3% Cl was used to determine the position of the reaction front.

b. Profiles were also determined by slicing concrete specimens approximately 1-3 mm thick and in 1.5 x 1.5 cm cross section. Kerosene was used as the lubricant. The concrete was powdered, the anion was extracted by nitric acid and quantitative evaluation was made by titration.

3. Characterization of concrete:

a. Pore-size distribution: Pore-size data were obtained on mortars extracted from the concretes. An Autoscan 33 Quantachrome mercury intrusion porosimeter was used at pressures up to 227 MPa.

b. Non-evaporable water and Ca(OH)2 content: These determinations were made using the mortar phase. Non-evaporable water was taken as the weight loss on heating between 100-55C, assuming the weight loss between 550-1000C to be negligible. Calcium hydroxide content was determined by estimating the weight loss associated with its decomposition at around 450C.

4. Diffusivity measurements: The two-compartment diffusion cell used for these measurements is similar to that used in other research works (11, 12, 13). The data was recorded over 200 days in order to determine the effective diffusivity of chloride ions at 21C. Steady-state conditions were achieved after that time.

RESULTS AND OBSERVATIONS

Durability Aspects

Diffusivity:

Even though properties such as compressive strength, Young's modulus of elasticity and the porosity of hardened cement paste, mortar and concrete are indirectly related to durability, the transport properties of diffusivity and permeability are probably the most important factors relating to durability (8).

Table II lists the values obtained for the diffusivity of the mortars of Systems 1, 2 and 5. Data to calculate diffusivity (D) were obtained using diffusion cell techniques after exposure to NaCl solution for 75 to 175 days. Generally, diffusivity increases with an increase in total porosity and with cement systems S5<S2<S1. Thus the cement systems rank similarly for the properties such as median pore diameter, resistivity and diffusivity. This is similar to the ranking for Ca(OH)2 content; the higher the Ca(OH)2 content, the lower the resistivity. System 5 concrete has the lowest diffusivity coefficient, as it has the lowest total porosity percentage.

Table II

These measurements indicate that median pore diameter and Ca(OH)2 content are ranked in the same order for the three cement systems, and that they are similar to the electrical conductivity and diffusivity. The data obtained from the SEM indicate that concrete System 5 ranks the lowest with respect to permeability, and provides maximum resistance to chloride attack. On the basis of the physical test results and the diffusion test data, System 5 mix 2 was selected as the candidate high-performance concrete for the repository construction.

Ionic ingress:

Results of the influence of concrete quality on ionic ingress are shown in Fig. 1. Depth of chloride ion penetration is plotted as a function of the square root of time of exposure. The figure shows that supplementary materials provide enhanced performance with respect to ionic ingress. The reference OPC concrete, System 1, has the least resistance to chloride ingress. System 2 with Type 50 cement and 10% silica fume provides better resistance. This is expected, since the addition of silica fume makes the concrete less permeable for ionic ingress, due to the pozzolanic reaction between Ca(OH)2 and the silica fume. System 5 concrete, with 75% slag and 3% silica fume replacing Type 50 cement, provides the most resistance to the chemical ingress. This illustrates the superior quality of the blended cement with regard to permeability, and hence resistance to deterioration (6, 7). This trend is also true for all water-to-cement ratios from M1 to M4 (from 0.35 to 0.6), and for all salt concentrations from 0.99 to 49.95 g/L of NaCl.

Fig. 1

Figure 2 shows the depth-of-penetration of chlorides into System 1, 2 and 5 concretes with different water-to-cement ratios. Water-to-cement ratios of 0.35 and 0.42 are clearly superior ratios of 0.5 and 0.6, in terms of resistance to the penetration of chloride ions.

Fig. 2

Figure 3 shows the synergistic effects due to chloride, sulphates and CO2 bubbling through the solution, for an exposure period of 34 months. The chloride ions penetrate the concrete more when it acts alone than when combined with sulphate or sulphate and CO2. This phenomena can be due to the precipitation of solid reaction products such as calcium sulphates and sulpho-aluminates into the pores, making the concrete denser, and thereby less permeable to the chloride ions.

Fig. 3

Service life predictions:

The service life of reinforced concrete structures exposed to sufficient chloride ions to initiate the corrosion of embedded reinforcement is largely controlled by the rate at which the chloride ions penetrate the concrete. Figure 4 shows the depth-of-penetration of chloride ions into Systems 1, 2 and 5 concrete specimens as a function of square root of time of exposure in Bath 13. Even with the scatter in the data, the System 5 concrete shows a remarkably lower rate of chloride ion penetration. Regression lines are also drawn for the concrete systems. Table III lists the time in years required for the chloride ions to penetrate a depth of 75mm into the concrete Systems 1, 2 and 5, based on the assumption of constant diffusivity during that period of time. The correlation coefficient of 0.8 or above in the Table provides higher confidence in the analysis results. Unlike the laboratory test specimens, the repository concrete will undergo microcracking or cracking due to imposed mechanical loads and other effects. The influence of cracks in concrete on the rate of ionic ingress has to be taken into consideration for the final assessment (3).

Fig. 4

Table III

DISCUSSION

The median pore diameter and Ca(OH)2 content are ranked in the same order for the three cement systems, and are similar to the electrical conductivity (inverse of resistivity) and diffusivity. Generally, the diffusivity increases with cement systems S5<S2<S1. Comparison of these

diffusivity coefficients with those obtained by previous workers (13, 14, 15) can only be general, because the cements and curing times are not exactly the same and the values of the previous workers are for pastes. Values for Type 10 cement at water/cement ratios of 0.4, 0.5, and 0.6 were found to be 260, 447, and 1235x10-10cm2s-1, respectively, for diffusivity (D), compared to 247, 418, and 581x10-10 cm2s-1, respectively, obtained in this work. Curing times were longer for the latter set. These results confirm that hydrated blended cement bodies have significantly lower diffusivities than normal hydrated Portland cement bodies, and that the lowest diffusivities were obtained with the 75% blast furnace slag, 3% silica fume blend.

Ionic profiles and depth-of-penetration measurements (determined by EDXA) in concrete show that reasonably accurate results can be obtained and predictions of ionic ingress made. There is some scatter in the experimental results, because of the difficulty of locating the reaction front in concrete test specimens, due to the tortuous path of ionic ingress through dense concrete. In addition, the rate of movement of the front can be speeded up by the rapid diffusion of ions in interfacial regions and in cracks. However, there is enough consistency and redundancy in the system to obtain fairly accurate results. The procedure following the diffusion path around the fine and coarse aggregate particles, using the scanning electron microscope and electron microprobe for analysis, has been successful.

CONCLUSIONS

The following can be concluded from the experimental test data:

1. Hydrated blended cements mortars have diffusivities up to 25 times lower than equivalent Type 10 hydrated Portland cement mortars. A 75% slag system generally yields the lowest values of diffusivity among the blends.
2. Median pore diameter and Ca(OH)2 content are ranked in the same order for the three cement systems (S5<S2<S1), and are similar to the ranking for electrical conductivity and diffusivity.
3. Lower water-to-cement ratios in concrete systems decrease the diffusion rate of ions, and sulphate ions inhibit the rate-of-penetration of chloride ions.
4. On the basis of experimental studies to-date, the System 5 concretes rank the lowest with respect to permeability, and provide maximum resistance to chemical attack.

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38-41

APPLICATION OF A NEW TECHNIQUE FOR INVESTIGATING THE CONFINEMENT OF REACTIVE CONTAMINANTS UNDER FIELD CONDITIONS

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ABSTRACT

The effectiveness of engineered barriers on long term performance of disposal facilities is linked to effects of variations in infiltration and of rock-water interactions. This work investigates the effects of nonsteady flow on reactive contaminant migration in a radioactive waste disposal facility, using a new computer code, DYNATRAN. This code differs from conventional models in two aspects: a) use of DYNATRAN can eliminate the need to use an overly conservative infiltration rate; and b) this code can take into account rock-water interactions which can significantly enhance waste confinement in a disposal facility. Use of DYNATRAN therefore, has the potential to demonstrate environmental compliance with less effort and higher reliability than conventional approaches. This study considers a hypothetical near-surface disposal facility that involves engineered low-permeability earthen covers, concrete canisters, and backfill. A numerical investigation is conducted in a phased manner to evaluate the combined impacts of infiltration and

geochemical reactions, with a focus on the performance of engineered features in the confinement of waste materials.

INTRODUCTION

In analyzing the confinement of contaminants by engineered barriers in waste disposal facilities, a critical concern is with the infiltration that may come into contact with waste canisters since the process will gradually dissolve the contaminants as they are exposed. In fact, an accurate and reasonably conservative estimate of infiltration rate is perhaps more important than any other factor in predicting radiation dose via groundwater pathways in a radioactive waste disposal facility. An important question is then raised about the impacts of variations in the infiltration rate on the long-term performance of engineered features in the facility, and specifically, on their ability to confine radionuclides. In addition, the performance analysis is complicated by various soil/rock-water interactions such as chemical precipitation and sorption. The purpose of this work is to develop a new technique for investigating the effects of variable infiltration on containment of radionuclides and other types of reactive chemicals in such environments. For this purpose, DYNATRAN, a general-purpose numerical code for groundwater flow and reactive contaminant transport, has been developed (1, 2). The code has the potential to demonstrate enhanced waste confinement due to two advantages over conventional models: a) DYNATRAN simulates variable infiltration under realistic field conditions, thus avoiding the use of an overly conservative infiltration rate; and b) the code considers rock-water interactions such as precipitation and sorption which have been found to significantly retard the movement of contaminants in cases studied (1, 2).

As an application to radioactive waste management, DYNATRAN is used in the present work to simulate infiltration in a hypothetical LLRW disposal facility. This work focuses on the performance of engineered features in confining radionuclides of concern. Numerical simulations are carried out in a phased manner. First, unsaturated flow within the facility resulting from steady infiltration is studied. Next, the impacts of variations in infiltration on flow will be examined. Following that, the release of radionuclides from the facility will be simulated by considering the full problem of infiltration, chemical reactions, and transport. Comparison will be made between the steady and nonsteady infiltration conditions. Progress so far and plans for future activities are presented in this paper.

NUMERICAL MODEL DYNATRAN

For application to this and similar problems, a generic numerical model, DYNATRAN, was recently developed to simultaneously handle transient saturated-unsaturated flow, multicomponent transport, and solid-liquid interactions in the subsurface (1, 2). DYNATRAN can be run in three modes: a) subsurface flow, b) geochemical equilibration, and c) subsurface transport under steady or transient flow conditions. In equilibration and transport simulations, DYNATRAN can handle liquid or liquid-solid systems. In transport simulations, it can handle either single or multiple species that are subject to reactions such as ion pairing, precipitation-dissolution, sorption-desorption, and reduction-oxidation. The code is capable of handling heterogeneous and anisotropic media in up to three dimensions with irregular geometry and time-dependent boundaries.

DYNATRAN has been verified extensively against analytical solutions and numerical models for several cases that involve flow, geochemical equilibration, and transport under steady flow (1, 2). One of the verification cases considered simultaneous transport of 14 chemical components, 78 additional aqueous complexes, and 9 minerals with highly contrasting alkalinity and redox conditions. The code has also been used in two studies to model migration of single- and multiple chemical components subject to various types of rock-water interactions in a dynamic flow field. The first application involved the impacts of seasonal infiltration/evaporation on contaminant leaching in a landfill. The second case involved a detailed geochemical evolution of an irrigated soil column, that included 16 components, 93 additional aqueous species, and 10 minerals.

DYNATRAN differs from conventional flow and transport models in the following two aspects:

1. The incorporation of transient flow allows for simulation of infiltration fluctuations, thus eliminating the need to use overly conservative estimates of the infiltration rate. The reduction in infiltration rate will strongly reduce contaminant concentration and environmental risk resulting from leaching.
2. The incorporation of detailed chemical reactions allows for simulation of precipitation, sorption, and reduction/oxidation reactions. In a study of acidic mine drainage at Pinal Creek, Arizona, the migration rate characterized by the fronts of pH and heavy metal concentrations, was reduced by a factor of 5 when precipitation was taken into account. Similar retardation due to rock-water interactions was also observed in other cases investigated.

Therefore, use of DYNATRAN in evaluations of waste confinement provides realistic simulations of field conditions that can potentially result in significant reduction in the licensing, design, and operation costs of waste containment facilities. Due to its generic formulation, DYNATRAN can also be applied to other cases that involve subsurface processes such as the remediation of contaminated groundwater and soil.

STUDY OF CONFINEMENT WITHIN A DISPOSAL FACILITY

Conceptual Model

This study considers a hypothetical disposal facility for low- and intermediate-level radioactive waste (LLRW). The facility is based on the concept of below-ground concrete vault and involves engineered low-permeability earthen covers, concrete canisters containing waste packages, and backfill (Fig. 1). To minimize infiltration into the vault, multiple layers of low-permeability natural clay and engineered high-density polyethylene liners will cover the vault. Void space between waste canisters will be filled with sandy materials to maintain structural integrity and divert infiltration away from the waste. Material and site hydrogeochemical data representative of the facility is collected from literature reviews of site characterization studies at several facilities in the U.S. and abroad, on which a conceptual hydrogeologic model has been developed.

Fig. 1

The model is designed to represent a general near-surface disposal facility, and can be easily modified to accommodate site-specific data. The design includes waste canisters placed within several disposal units. In each disposal unit, fifteen 80"80"80" waste canisters will be uniformly placed in three levels of loading (Fig. 2). Each waste canister

is made of an outer fiber-reinforced concrete shell with an inner polyethylene container. The horizontal space between two neighboring canisters is 2", which will be filled with a cohesionless material such as pea gravel. The fill will be selected according to NRC guidance (3). The top and two sides of each disposal unit are filled with 8" thick gravel, and the bottom with 4" thick gravel.

Fig. 2

This model considers the distribution of infiltration within the disposal unit after it reaches the top pervious layer overlying waste canisters. The loading configuration of waste canisters in one disposal unit is illustrated in a typical cross section (Fig. 2), which is considered in this study. Lateral flow in the third dimension was not simulated because this flow component was not anticipated to be significant in comparison with infiltration. By symmetry, only half of the cross section is modeled. The cross section was discretized into a computational grid of 612 nodes. To ensure reasonable accuracy in the simulated results of the partially saturated flow case, the grid was made finer at the top of the disposal unit where infiltration occurs and around the fills where higher flow fluxes are expected.

Flow under Steady Infiltration

Numerical simulations are carried out in a phased manner. First, a preliminary analysis was conducted to investigate the subsurface flow under the influence of steady infiltration within the disposal facility. Specifically, we used an overall infiltration rate independently estimated with HELP3 (4), and representative material properties as input. Then we simulated the dynamic infiltration into a typical disposal unit under unsaturated conditions, and examined the relative amount of flow through the canisters and through the fill materials during a period of 300 years following significant degradation of the concrete canisters.

The flow analysis was based on the following assumptions:

1. Degraded waste canisters were used so that their properties would ensure conservatism in the calculations.
2. The initial conditions were assumed to be hydrostatic such that the water level was at the bottom of the infiltration collection system.
3. The infiltration, after percolating through the earthen cover, reaches the top of waste disposal unit at a constant and uniform rate. The infiltration will flow through either the waste containers or the space in between the waste containers that is filled with the fill materials.
4. The left and right sides of the model were treated as no flow boundaries. The bottom boundary was assumed to be such that the infiltration collection drain remains dry during the time period of interest.

The infiltration rate reaching the top of the disposal unit was independently assessed using the HELP3 model. To be conservative, the maximum rate obtained, 3 inches/year, was used in a steady infiltration case. Variations in the infiltration rate will be taken into account in a more realistic study of transient flow. Steady infiltration was treated as a constant flux in the simulations.

Both fully saturated and partially saturated conditions were considered. The flow problem is highly nonlinear due to the strong dependence of hydraulic conductivity on pressure head in the fill and the contrast in hydraulic conductivity between the concrete and the fill. Important factors in the infiltration study are the hydraulic characteristics of the concrete and fill materials. Backfills were considered to be

consistent with NRC recommendations. Based on these specifications, the hydraulic conductivity of the backfill was estimated using the Fair and Hatch formula (5) and data from Maidement (6). Material properties of degraded concrete were derived from measured fresh concrete and conservative assumptions about degradation (7). The hydraulic conductivity and saturation characteristics of the waste canisters and the fill were both developed using the van Genuchten model (8) (Fig. 3).

Fig. 3

Several cases were carried out to test the sensitivity of infiltration-induced flow to key model assumptions and input parameters (Table I). Case I was chosen as the base case. Additional cases were studied considering variations in hydraulic conductivity of degraded waste containers and backfilling materials. Relative amount of flow through the waste-containing concrete canisters and that through the backfills were calculated for the time period of interest, which provides a more realistic estimate of the amount of leachate generated than the often too-conservative approach traditionally adopted in performance assessment.

Table I

For the particular configuration and material properties assumed, it was estimated, based on a simple water balance, that when the disposal unit is fully saturated and a uniform hydraulic gradient exists over the top of the disposal unit, more than 99% of the total infiltration that reaches the top of the disposal unit will flow through the filling materials and will not come into contact with the waste.

Numerical simulations suggest that the ratio of flow bypassing the waste canisters would decrease when the fills are partially saturated (Table I). In addition, the results showed that the waste containers would remain at full saturation even when the fill materials are under very dry conditions, which is consistent with the observation of Walton and Seitz (9). Partially saturated flow, therefore, is significant only in the fills. It was noted that in the multi-layer system, the bypass ratio is dependent on the location of waste containers. Around waste containers closer to the infiltration collection drain, the fills tend to become more saturated and will withdraw more fluid away from the neighboring waste containers. Infiltration is also a time-dependent phenomenon in the disposal facility; the ratio is dependent on how long infiltration has taken place after the waste canisters are degraded. For the specific combinations of parameters (Table I), the bypass ratio was found to be at least 60% (Fig. 4). Therefore, under similar site conditions the source term methodologies that assume that all the infiltration will come into contact with waste can be considered conservative. The results also suggest that design features of the facility will continue to divert a significant amount of the water away from contact with the waste after degradation of canisters has occurred.

Fig. 4

Future Work

In the next phase of this study, we will consider transient flow within the facility caused by fluctuations in infiltration. Incorporation of variations in infiltration will allow simulations of realistic field conditions, thus avoiding the need to use a conservative infiltration rate throughout the time period of interest. Appropriate boundary conditions are being developed to handle variable infiltration. As in the steady infiltration case, relative flow through the waste canisters

(i.e., leaching), will be calculated as a percent of total infiltration that reaches the top of the disposal unit.

The final phase will utilize DYNATRAN's provision for integrating flow and transport to simulate the long-term performance of a disposal unit for confining radionuclides under realistic infiltration conditions. Amount of leachate and radionuclide concentrations in the leachate will be calculated as a function of time and location within the disposal facility. This will help to identify locations within the facility that are weak to resist infiltration, and thus may assist in refining the design regarding the placement of waste canisters. When geochemical characteristics of the infiltrating water and the facility are available, geochemical reactions among radionuclides and other chemicals will also be considered. It may be noted that the current practice of disposal assessment usually ignores or oversimplifies such chemical interactions. It will therefore be of particular interest to examine the system behavior when the coupled flow-transport-reaction processes are taken into account in the full problem.

SUMMARY

A new technique and the associated numerical code (DYNATRAN) have been developed for studying the combined effects of variable infiltration and rock-water interactions on the confinement of residual materials in disposal facilities. This approach has the potential to demonstrate environmental compliance through realistic confinement evaluation due to its inherent advantages over the conventional methods: a) use of DYNATRAN can avoid the need to use an overly conservative infiltration rate; and b) DYNATRAN takes into account of rock-water interactions such as precipitation and sorption, which can significantly enhance waste confinement. In an application to radioactive waste management, DYNATRAN is used to investigate the confinement of radionuclides and other reactive chemicals within a disposal facility. Preliminary results of the analysis of infiltration-induced flow show that the majority of infiltrating water will be diverted by the fill in between concrete canisters, even after the canisters are degraded. The amount of diverted flow will be impacted by hydraulic properties of the canisters and fill, as well as the configuration of the disposal unit. To provide further insights into the confinement assessment, detailed evaluation of nonsteady infiltration and contaminant transport is under way.

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Session 39 -- MARKET DRIVEN RECYCLING
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39-1
COMPREHENSIVE APPROACH TO RECYCLING RADIOACTIVE SCRAP METAL

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ABSTRACT
A market-driven trend toward integrating the four major disposal and recycling options for radioactive scrap metal (RSM) into a single, comprehensive approach is described. Each of the four currently-available disposal/reuse options and the respective roles of RSM generators and processors are described as background for discussing shortcomings of traditional approaches to RSM disposition. Poor characterization and "bundling" of RSM into heterogeneous "lots" often result in critical errors in the selection of a single option for RSM disposal or reuse. A specific example of disadvantages associated with this "specialized" processing approach is provided. As a direct result of these problems and other important market forces, RSM processors have "diversified" their individual disposal/reuse processes to minimize risks associated with RSM recycling. A comprehensive approach integrating individual disposal/reuse options into a single, flexible scheme is described. Unlike the traditional methods, this comprehensive approach emphasizes up-front segregation of metal components into categories best-suited for specific technologies. An example of the new approach is provided.

INTRODUCTION

Generators of radioactive scrap metal (RSM) typically must choose one of four basic options for the final disposal and/or reuse of their material:
1. Direct Burial whereby RSM is packaged in boxes or drums and deposited in a licensed radioactive material landfill; alternatively, some landfills do not require prepackaging of RSM and the material is deposited directly into the landfill "cells".
2. Volume Reduction whereby RSM is physically modified to reduce its "packaged" or spatial volume prior to burial. Volume reduction is achieved in one of two ways:
a) Mechanical Volume Reduction in which RSM is cut and/or compressed using standard mechanical techniques including saws, torch-cutting equipment, plasma-arc and metal compactors.
b) Metal Melt in which RSM is transformed into a volumetrically-contaminated metal block at close to the ideal mass density of the original material (e.g., close to 495 lbs/ft3 for stainless steels).
Volume reductions (VRs) typically must exceed 80% (i.e., for every 100 ft3 of RSM volume reduced, less than 20 ft3 of resulting material is sent to a burial site for final disposal) to be cost-effective. It is

important to note that volume reduction affects only the volume of RSM buried and not the mass. Therefore, volume reduction is best thought of as a density change -- packing more material into less space.

3) Metal Melting/Beneficial Reuse whereby radioactive contamination is volumetrically dispersed throughout an homogeneous molten metal mixture and, subsequently, formed into ideal density blocks, ingots or other forms of internally-contaminated metal. Subsequently, this material can be formed into products for restricted use in the nuclear industry. Typical applications include shielding, waste boxes and drums, casks and canisters. Secondary waste resulting from melt operations can vary from 5 - 25% by volume of incoming material (as received at 20 - 35 lb/ft3 bulk densities).

4) Surface Decontamination/Recycling whereby radioactive contamination is removed from the external surfaces of RSM for proper handling and burial; concomitantly, the decontaminated or treated bulk metal objects are surveyed and, where specific criteria are met, released for unrestricted use in private-sector scrap metal markets. Secondary waste resulting from decontamination activities is typically limited to 1 - 10% by volume of incoming material (as received at 20 - 35 lb/ft3 bulk densities).

Over the last several years, RSM generators and private sector "processors" have learned that no single option (discussed above) is best-suited for all types of RSM. Technical, economic and political factors continue to drive processors toward "comprehensive" services which integrate the four (4) basic options into a single, systematic approach for RSM handling, recycling and disposition. This, in turn, means that the RSM must be properly characterized and sorted in terms of important physical, chemical and radiological parameters and, subsequently, properly assigned to the best available technical option for that specific type of RSM -- direct burial, volume reduction, metal melt or surface decontamination.

APPLICABILITY OF OPTIONS -- A SIMPLISTIC VIEW

The task of identifying and choosing the best disposition/reuse option from the four (4) presented above is made relatively simple by assuming that all of the RSM under consideration has already been sorted and collected into homogeneous "lots". In this case, homogeneity is defined as a close similarity among the individual RSM objects in a single "lot" in terms of the characteristics listed in Table I. Since empirical evidence confirms that, in general, RSM is notoriously heterogeneous in nature, the preceding assumption dictates that these homogeneous "lots" of RSM vary significantly in character from one "lot" to the next. In other words, the RSM exhibits homogeneity within "lots" and heterogeneity between "lots". Under this scenario, the two overriding factors affecting selection of an option are:

1) Technical feasibility of the option or process for the specific type and form of RSM contained in the "lot" under consideration. The specific characteristics of individual RSM material is the key to matching each "lot" to one of the four (4) disposal/reuse options. As the information provided in Table I clearly indicates, the number of pertinent evaluation criteria increases dramatically from direct disposal to volume reduction to metal melting to surface decontamination. In other words, as the disposal/reuse option moves from less "green" to more "green" in terms of true recycling, the selection criteria for RSM material becomes significantly more restrictive.

The diversity and breadth of evaluation criteria provides some insight into the complex nature of sorting RSM for treatment/disposal. Further, certain evaluation criteria such as "percent inaccessible surface" indicate opportunities for further processing decisions. In this particular case, a processor may determine that, although the amount of surface accessible to decontamination and survey is initially low, the RSM under consideration may be easily modified (mechanically) to expose most or all of the surfaces at little cost to the processor. Once these surfaces are properly exposed, the material qualifies for cost-effective treatment by surface decontamination.

2) Economic feasibility of applying a given option to a specific RSM "lot". In the absence of government- or industry-imposed incentives (i.e., recycling "rebates" or "credits"), economic feasibility continues to be largely dependent upon direct burial rates. This dependence on burial rates is due to the following:

a) All technical options for RSM disposal/recycle produce waste(s) -- primary, secondary or a combination of both primary and secondary wastes.

b) Waste volumes generated per unit RSM treated tend to decrease with greater restrictions on the physical, chemical and radiological characteristics (listed in Table I) of the material under consideration. However, the economic feasibility of a disposal/recycle option for a given RSM "lot" may also be heavily dependent upon other factors. For example, processing costs may increase with tighter restrictions on input material. In this case, any potential waste volume savings may be offset by these increases in processing costs.

Table I

In summary, the task of selecting the best disposal/reuse option is relatively straight-forward for any single, homogeneous "lot" of RSM. This selection is dependent entirely upon the relative technical and economic feasibility of each option for the specific RSM "lot" under consideration.

DETERMINING THE BEST APPROACH -- A REAL-WORLD PERSPECTIVE

Unfortunately, the preceding assumption that RSM is classified and sorted into "lots" of similar material prior to selecting a treatment option is a significant oversimplification of actual market practices. In fact, generators have always and, in most cases, continue to "bundle" RSM for direct disposal or further treatment as an heterogeneous mix. Although interested in the overall costs of treatment and extent of reuse achieved by processors, RSM generators are more heavily influenced by other factors:

1) Schedule and Critical Path Issues: For commercial utilities, RSM is often generated during power station "outages" and, consequently, greater emphasis is placed on speed and critical path issues than on sorting material into various "lots" just to promote the reuse of a few thousand cubic feet of RSM.

2) Demolition Costs: The decontamination and decommissioning (D&D) of commercial and government facilities generally involves the removal of large equipment, structures and buildings. Obviously, demolition and random "bundling" of the resulting RSM (in containers) represents the least-expensive method for achieving the demolition objectives. Although not clearly defined to date, the decommissioning costs associated with systematically dismantling structures so as to optimize RSM treatment processes are, intuitively, higher than those associated with traditional demolition techniques. However, potential economic savings from

optimization of RSM treatment processes (e.g., avoided transportation and disposal costs) have also not yet been fully defined and made available in the marketplace.

3) Direct Disposal Rates: In the case of government facilities generating RSM, perceived disposal rates less than $100 per cubic foot and, in some instances, as low as $10 - $20 per cubic foot have virtually eliminated reuse/recycling options from consideration. In these cases, classification and sorting of material prior to placement in containers for direct disposal is of no technical or economic value and, therefore, ignored in the D&D process.

4) Partitioned D&D Responsibilities: In most D&D applications, specific responsibilities for demolition and waste disposition/reuse, respectively, are "decoupled" -- assigning two different managers with specific tasks that are not considered to be related except in terms of project scheduling and which, in many ways, conflict with one another. The manager in charge of demolition is concerned only with removing the equipment or structure at least-cost and on- schedule while the waste-minimization manager is concerned with optimizing the RSM disposal/reuse process. In effect, the two (or more) managers are "partitioned" from one another -- each with specific responsibilities which, in most cases, are never integrated into one overall plan that may minimize the cost of the overall D&D effort. Until recycling/reuse is adopted as an integral part of D&D planning, it will be difficult to "couple" these activities into a single least-cost option.

Given these and other influences, RSM generators continue to "bundle" material for treatment as a heterogeneous mix of metal types, configurations, densities, shapes, geometries, sizes and other characteristics. Despite this practice by the RSM generators, the processors have, until recently, been providing relatively "specialized" services. One or two processors have emerged as metal melting specialists while others have focused their efforts on volume reduction and surface decontamination. Even within the various option categories, processors have developed and maintained specific areas of specialization. Although most processors have always claimed to offer a broad range of services, all have practiced some level of specialization over the last 5 or 6 years.

At first glance, this "specialization" by processors is somewhat confusing. Given the way in which generators "bundle" RSM and the greater restrictions on acceptable RSM forms for the "greener" options such as metal melting and surface decontamination, it only makes sense that each processor would want to provide all four (4) of the disposal/reuse options. However, large capital investment requirements and regulatory hurdles have precluded most from this approach -- forcing each processor to carve out a unique niche based on his/her respective financial and technical resources. With the exception of volume reduction, the less "green" options require greater financial resources and investment capital (e.g., metal melt equipment is typically more expensive than surface decontamination equipment). On the other hand, the less "green" options (including volume reduction) can accept a greater range of RSM (see Table I). Therefore, it makes sense that processors with the financial wherewithal would choose options that inherently offer a broader range of RSM treatment capability.

Unfortunately, this "specialization" by processors has placed the burden of the disposal/reuse decision squarely on the shoulders of the

generators themselves. Although the "greener" options are available, the fact that RSM is still "bundled" as an heterogeneous mix often precludes generators from optimizing the disposal/reuse of their material. Further, once the metal is mixed and containerized for shipment, the arduous task of characterizing and sorting the material represents an additional cost which, despite avoided disposal costs achieved through improved reuse/recycle, may still be cost-prohibitive. Consequently, the generator must make some assessment of the character of his/her RSM and, based on claims and prices from the processors, make a decision regarding the ultimate fate of the material. In most cases, generators shift most of the risk associated with the decision to the processors by requiring firm fixed price contracts and guaranteed recycling or volume reduction efficiencies.

The down-side risk for the RSM generators and processors associated with the selection of a "greener" option can be enormous. Figure 1 provides a visual representation of the decision faced by RSM generators in a market characterized by specialized processors. The generator, based on his/her own internal information, must decide the treatment path. Although surface decontamination and metal melt may offer economic benefits by means of avoided burial costs, an error in judgement at the outset can result in significant unforeseen costs. For example, assuming the generator believes that his/her RSM is well-suited for surface decontamination, it is reasonable to expect avoided disposal costs which will more than offset the cost of processing. However, if the material received by the processor is improperly characterized by the generator, the specialized processor (with no other technical option available) has no choice but to "re-characterize" the material and sort out the RSM unacceptable for surface decontamination. This unacceptable material, set aside as primary waste in Fig. 1, must then be placed back into acceptable containers and either returned to the generator or shipped directly to a burial site. In any event, the additional costs associated with transportation, containers and material handling can lead to a final cost well above that for direct disposal of the entire volume initially considered. Therefore, without some economic incentive (e.g., high burial rates, recycling credits), most RSM generators are motivated to choose the broader, less "green" options -- especially direct disposal. In a market characterized by low or ambiguous burial rates, it is difficult to justify any amount of true recycling for RSM.

Fig. 1

A REAL-WORLD EXAMPLE: FERMCO STRUCTURAL STEEL

In December 1994, ALARON Corporation was engaged by the Fernald Environmental Restoration Management Corporation (FERMCO) to participate in a feasibility study evaluating surface decontamination as a cost-effective option for the reuse/recycle of structural steel. The decontamination project was designed by FERMCO as an integral part of the D&D of the Fernald Plant 7 building. Therefore, following the initial implosion of the six (6) story building, the structural steel was to have been removed by cutting and shearing to produce material acceptable for surface decontamination. The structural steel consisted of a variety of painted I-beams, deck-plate, channel iron, angle iron, piping and bridge-crane steel. In almost all cases, the steel had been painted a number of times with lead- and latex-based paints. The lead-based paint was used originally to prepare the surfaces prior to use of the building and to "fix" contamination over the years of building use, while the latex paint

was used by FERMCO to "fix" any remaining contamination just prior to building implosion. A relatively small amount of non-metal material was, by design, to have been delivered with the steel. This material included wooden pallets and super-absorbent sheets used for shipping purposes. Based on design drawings and information provided by FERMCO, ALARON's technical staff determined that grit blasting represented the best general method for decontamination of this particular metal. Although some amount of sorting, further characterization and specialty decontamination (i.e., chemical decontamination of some components) was anticipated, the original characterization and assessment of the material indicated that the vast majority of the material would be easily decontaminated by grit-blasting. ALARON's basic approach, based on experience with similar materials, assumed that free-release criteria could be met by simply removing the paint from the metal surface. Although paint as thick as 8 mils was expected on some surfaces, preliminary testing with grit-blast equipment confirmed that the coatings could be effectively removed.

Figure 2 provides a flow-diagram summarizing the anticipated material flow for the structural steel and other materials shipped to ALARON's facility in Wampum, Pennsylvania. Note that the general form of Fig. 2 agrees with the "Specialized Processing Services" scenario provided in Fig. 1. As the RSM generator, FERMCO was faced with the decision of how best to dispose/reuse the Plant 7 structural steel. After assessing their own material and proposals submitted by various "specialty" processors, FERMCO chose surface decontamination as proposed by ALARON Corporation. The non-bracketed numbers provided in Fig. 2 indicate the originally anticipated quantities (tons) of material for receipt and processing per ALARON's plan. Note that, of the total 725 tons expected, approximately 700 tons were earmarked for surface decontamination. Some minimal amount of sorting and surface preparation prior to surface decontamination was accounted for in the original plan. Based on assumptions regarding material characterization and processing efficiencies, ALARON believed that a total of 693 tons of structural steel could have been free-released to scrap dealers for recycling. This represented, by weight, a 95% recycling efficiency based on the total material received (including non-metals) and a 99% recycling efficiency based on the structural steel selected for surface decontamination. Total primary and secondary waste anticipated was approximately 82 tons, with material unacceptable for decontamination accounting for 30% (by weight) of the waste generated.

Fig. 2

Despite the planned approach, extensive preparation for recycling of the structural steel, and FERMCO's commendably high level of understanding of the criteria for cost-effective decontamination, unforeseen circumstances negatively impacted the recycling process. The initial implosion of Building 7 was not successful. Consequently, in order to meet schedule and budget requirements, further demolition efforts were required to prepare the building for dismantlement. Unfortunately, these unanticipated demolition activities and tight schedule for removal of the building resulted in a departure from the originally-planned dismantlement procedures. The overall approach shifted from systematic dismantlement to a "brute-force" demolition in which material was sheared, ripped, bent, and otherwise modified without regard for any subsequent impact on the planned decontamination process. A clear indication of the twisting, bending and poor treatment of the material

from a decontamination perspective was provided by the material bulk densities achieved prior to shipment to ALARON's facility. Under the original assumption of systematic dismantlement, bulk densities of 24-30 pounds per cubic foot were anticipated and, in fact, achieved for the bridge-crane and other material unaffected by the modified dismantlement approach. By comparison, the bulk densities achieved for structural steel resulting from the demolition were typically less than 20 pounds per cubic foot -- with some as low as 16 pounds per cubic foot.

In contrast to the originally anticipated material flow, the tons of material actually received and processed (bracketed numbers provided in Fig. 2) reflect the strong, negative influence of the demolition method on recycling efficiency. The additional 36 tons of material received (761 tons actual versus 725 tons originally anticipated) was, for the most part, due to reasonable error in estimating the structural steel weight prior to demolition. However, the division of material by sorting into metal suitable and unacceptable, respectively, for decontamination clearly illustrates the detrimental impact on recycling. Of the 761 tons of material actually received, only 448 tons (59% by weight) were characterized by ALARON as "acceptable" for surface decontamination. It is important to note that the material identified as "acceptable" by ALARON also possessed the general characteristics of the RSM specified in FERMCO's original Request for Proposal. The remaining 313 tons of material, mostly steel, were generally characterized by a high percentage of inaccessible surface area. This inaccessibility would have not only been detrimental to the grit-blast operation, but also would have prevented survey instrumentation from the access required to confirm that free-release criteria had been met.

The final results indicate that, overall, 445 tons or 58% (by weight) of the received 761 tons of material were actually decontaminated and free-released as "clean" scrap. However, it is important to note that, of the 448 tons characterized by ALARON as "acceptable" for decontamination, the originally-anticipated goal of 99% recycle (by weight) was achieved. Total primary and secondary waste was close to 350 tons, but material unacceptable for decontamination accounted for 90% (by weight) of the waste generated. These results serve to illustrate the previous argument regarding the level of risk taken by RSM generators in determining the best option for RSM disposal/reuse -- especially in a specialized-processor environment. In this particular case, the generator probably made the correct decision based on initial assumptions and original plans for dismantlement of Building 7. However, the shift in approach toward a demolition-based technique resulted in a significant difference in the final form and characteristics of the structural steel. In turn, this difference in the characteristics of the RSM precluded a large portion of the material from cost-effective decontamination.

A CHANGING LANDSCAPE -- MARKET FORCES AT WORK

Despite the preceding explanation and example of risks faced by RSM generators in recent years, the market for RSM disposition/reuse is evolving in such a way as to place significantly more risk and financial burden directly on the RSM processors. Market opportunities, direct disposal rates, and the level and nature of competition have all changed drastically over the last five or six years, forcing RSM processors to improve and broaden their capabilities and contract terms.

At the outset of the 1990's, the marketplace was relatively "simple" for RSM processors -- with commercial nuclear power station opportunities

dominating the market, unambiguous and firm burial rates and relatively limited competition. Most RSM was generated during power station "outages" and, as mentioned earlier in this paper, the generators themselves were more concerned with on-site critical path issues than the disposition/recycling of RSM -- especially when the worst case scenario was direct disposal at $35 per cubic foot. In order to compete, processors had to offer off-site (from the utility itself) processing, firm fixed-price contracts and guaranteed volume reductions. As shown previously, the RSM generators were required to decide which option represented the best approach for disposal/reuse of their metal wastes. Once that decision was made, the only other responsibility of the generators was to place the RSM in certified containers for shipment to the processing facility. Since the primary emphasis was usually on speed to prevent outage critical path conflicts, little or no thought was given to the condition, characterization or sorting of RSM prior to demolition, removal and shipment to the processor. Consequently, any required characterization, sorting and metal preparation was performed by the processor and, with burial rates so low and volume reduction the predominant option, these activities were relatively negligible in terms of cost to the generator.

Today the market can be characterized as significantly more "complex". Government opportunities have emerged and, with the promise of thousands of tons of RSM for disposition/recycle over the next 20 - 30 years, have caused an influx of competitors and technical alternatives. The number of competing companies has more than doubled since 1990 and new ideas and practices -- including RSM export to foreign countries, long-term storage and improved waste-minimization techniques -- have broadened the choices for generators of RSM. Although commercial burial rates have soared to more than $300 per cubic foot for non-southeast compact generators (forcing many utilities to invoke plans for long-term storage and other alternatives), government burial rates have yet to be clearly stated and supported by true life-cycle cost determinations. This ambiguity, together with the fiercely competitive nature of new companies struggling to "get in" the RSM recycle market, has served to erode RSM processors' margins and has forced processors to shoulder increasingly more financial risk. Processors have, over the last several years, invested millions of dollars in high-tech equipment and processes to meet the predicted demands of the Department of Energy and nuclear utilities. However, prices charged by processors today are not significantly higher than those charged at the outset of the 1990's. Further, even with all the private-sector capital at risk, processors continue to provide firm fixed-price contracts and guaranteed waste volume reductions without any real change in the way in which RSM is characterized, prepared and bundled by the RSM generators themselves.

A COMPREHENSIVE APPROACH

In order to effectively compete in this new and changing environment, many RSM processors have broadened their capabilities -- through capital investment and/or teaming with other processors -- to provide all four of the basic disposition/reuse options previously identified in this paper. These broadened capabilities provide the technical and economic diversification required to minimize their own risk in processing and disposing or recycling inherently-heterogeneous RSM. In other words, asystematic, comprehensive approach -- integrating all the available

options together to provide an optimal outcome -- is now available to meet the demands and restrictions of today's nuclear marketplace.

The comprehensive approach adopted by ALARON Corporation -- in conjunction and through a formal teaming relationship with Carolina Metals, Inc. (a subsidiary of Nuclear Metals, Inc.) -- is summarized in Fig. 3. The figure illustrates a flow scheme in which heterogeneous RSM is received, evaluated, characterized, sorted and prepared for disposition/recycle by each of the available options. In this way, RSM best-suited for metal melt is prepared specifically for and processed by the metal melt facility at Carolina Metals, Inc. in Barnwell, SC. Similarly, RSM best-suited for direct burial, mechanical volume reduction or surface decontamination is identified, prepared and processed by the corresponding technology.

Fig. 3

The development of comprehensive services is, in effect, a necessity for RSM processors in today's marketplace. Increased competition for RSM disposal and/or reuse has made it possible for generators to make more demands on the RSM processors. The generators still demand firm fixed price contracts and performance guarantees without any significant change in the way they "bundle" RSM as a heterogeneous mix. Consequently, the risk associated with the decision as to which option or set of options is best-suited for a specific "lot" of RSM is transferred completely to the generator. In order to minimize this risk, the processors must have at their fingertips the capability to handle any and all types of RSM. The challenge for processors is now to become highly efficient in characterizing and sorting RSM so as to optimize the overall process by ensuring that each piece of metal is matched to the best possible disposal/reuse option. The processors must "pay" for the additional handling and processing costs incurred by avoiding direct disposal charges.

COMPREHENSIVE APPROACH EXAMPLE: QUAD CITIES RWCU

In mid-1995, ALARON Corporation and Carolina Metals, Inc. (CMI) proposed a comprehensive approach to Commonwealth Edison Company (CECO) for disposal/reuse of retired reactor water cleanup unit (RWCU) components from the Quad Cities Nuclear Power Plant. The planned approach integrated direct disposal, volume reduction, metal melting and surface decontamination into a single-best overall option for the RWCU material. ALARON Corporation and CMI were selected by CECO for treatment of the RWCU material through a competitive bidding process.

The RWCU material originally described by CECO exhibited typical RSM characteristics. Metal varied significantly in terms of treatment selection criteria previously identified in Table I. However, based on the specific characteristics of the RSM, five (5) specific categories were established and evaluated for treatment -- "small" valves (< 2" o.d.), "large" valves (> 2" o.d.), piping, heat-exchanger tubing, and heat-exchanger "shell" material. Technical information provided by CECO indicated that all the RSM associated with this project was stainless steel. The heat-exchangers were regenerative and non-regenerative units typical of a boiling water reactor (BWR) system. Based on the initial characterizations provided by CECO, ALARON determined that the "small" valves and heat-exchanger tubing were best suited for metal melting. The "large" valves, piping and heat-exchanger shell materials were selected for surface decontamination -- due in large part to their relatively low

surface area to mass ratios and the accessibility of surfaces to treatment and survey.

Figure 4 provides a flow-diagram summarizing the anticipated flow for the RWCU steel accepted by ALARON Corporation. The non-bracketed numbers represent the tons of material expected to flow through each process based on initial RSM-characterizations provided by CECO. Of the roughly 32 tons expected for receipt, 10 tons (31% by weight) were slated for metal melt and 22 tons (69% by weight) for surface decontamination. Based on the initial characterization and plan, 9.8 tons of material were to be converted to ingots for beneficial reuse and 21.3 tons were to be recycled as clean scrap. The total waste volume was expected to be approximately 1.3 tons.

Fig. 4

The condition and characteristics of the RSM received by ALARON differed from the anticipated material in quantity, the presence of non-metal components, and activity level. Approximately 42 tons of RSM were actually received from Quad Cities compared to the 32 tons originally expected. Whereas the original plan was established for RSM exhibiting general field dose rates less than < 50 mR/hr (average), portions of the heat-exchangers and piping actually exhibited values exceeding 200 mR/hr (average) with a few hot spots as high as 1-3 R/hr. In addition, a small quantity of asbestos insulation was associated with the RSM received. This asbestos accounted for roughly 2.4% or 1 ton of the total 42 tons received. These differences further illustrate the difficulty in accurately characterizing RSM prior to dismantlement. In this case, restricted access prior to removal of each component prevented the generator from developing an improved picture of this material for disposal/reuse. However, the RSM characterizations provided by CECO were relatively accurate and quite useful in planning the processing activities.

The bracketed numbers provided in Fig. 4 indicate the tons of RSM actually received by ALARON. Of the 42 tons received, only 1 ton (asbestos) was classified as primary waste. The remaining 41 tons are presently being processed for reuse -- 14 tons by metal melt and 27 tons by surface decontamination. Based on preliminary results, 97-98% reuse/recycle efficiencies are anticipated for both metal melt and surface decontamination efforts. Of the total 42 tons actually received, only 2.7 tons of waste are expected -- with the asbestos (non-metal) material accounting for 37% of the total waste generated. Although the Quad Cities RWCU processing is still underway, available information strongly suggests that at least 95% of the RSM received will be processed for reuse. The resulting 26.1 tons of "clean" metal will be sold as scrap and the 13.7 tons of ingots from metal melt operations will be returned to Quad Cities for storage or on-site use as shielding material. Options for converting the ingots into other products are currently under evaluation.

Although the relative proportions of RSM slated for metal melt and surface decontamination are, in this particular case, very close to those originally anticipated, most estimates based on the characterization of RSM prior to dismantlement are not nearly as accurate. However, the availability of a comprehensive approach allows ALARON to bid and pursue market opportunities with a built-in degree of confidence and flexibility. Unlike the "specialized" processor scenario described in Fig. 1 and 2, ALARON is now able to "shift" material from one

disposal/reuse option to another in order to accommodate differences between initial and final characterization of the RSM under consideration. The preliminary results provided by the Quad Cities RWCU project (Fig. 4) indicate that, despite differences in key characteristics of the RSM originally anticipated versus that actually received, availability of comprehensive services allows ALARON to optimize the overall treatment scheme and, consequently, maximize recycling and minimize final waste volumes.

CASTING AN EYE TOWARD THE FUTURE

With the advent of comprehensive processing services, improved performance is likely to convince many generators that reuse and recycle represent cost-effective options. In turn, significant volumes of RSM are expected to be made available for processing over the next several years. Despite these changes in the RSM marketplace, potential for greater improvements in reuse/recycle may yet to be realized. As discussed earlier, present-day demolition practices continue to negatively impact RSM recycling options. This is due in large part to the fact that dismantling efforts do not incorporate recycling/reuse considerations and requirements into the up-front planning stage of the process. The two areas of the dismantlement process that offer the greatest opportunities for improved, cost-effective reuse/recycling are RSM characterization prior to demolition and, based on the results of characterization, sorting of material as an integral part of the demolition process. Based on the inherent heterogeneity of RSM, the four (4) disposal/reuse options are all critical components of an optimal solution. The potential savings in material handling, transportation and waste disposal costs made available by simply integrating recycle/reuse considerations into the dismantlement planning process are significant and, as such, should motivate RSM processors and generators to cooperate in the development of new dismantlement procedures.

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METALS RECYCLING IN THE YEAR 2000

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ABSTRACT

The future size of the radioactive scrap metals (RSM) recycling industry hinges on major policy decisions by the Department of Energy (DOE) and the implementation of a regulatory framework by the US Environmental Protection Agency (EPA) and Nuclear Regulatory Commission (NRC). If the industry meets its aggressive forecasts, over 50,000 tons from government and 20,000 tons from commercial, radioactively-contaminated scrap metal will be processed and recycled in the year 2000. Alternatively, without these key changes by the federal agencies listed above, the current industry will remain essentially flat consisting primarily of commercially generated waste at an annual rate of 15 to 20 thousand tons/year.

This paper forecasts a moderate view of metals recycling in the year 2000 - processing approximately 30-40,000 tons of radioactive scrap per year with 45% from DOE/Department of Defense (DOD) and 50% from commercial

generators. This segment of the nuclear industry will have annual revenues of approximately $200 million per year spread over a half dozen or more competitors. Services provided will primarily be decontamination for free release, melting for fabrication of containers or industrial items and melting for burial.

This paper presents a detailed discussion of the market and RSM industrial processing capacity, the policy and regulatory drivers, and the financial projections.

THE MARKET

What will the market for metals recycling be in the year 2000? By characterizing the market in the first six years of the 1990's and making assumption of the major market drivers, a reasonable projection of the market as we enter the 22nd Century are possible. The market is defined by two components, the generators and the processors.

PROCESSING EXPERIENCE

In the six-year period of 1990 to 1995, an estimated 100,000 tons of RSM from commercial and government generators was sent to processors for "recycling or burial." The introduction of commercial metal melting services in 1993 increased RSM processing by about 10,000 ton per year due to the large savings over burial or decontamination. Of the material that was not directly buried, roughly half was decontaminated and half melted into shield blocks for the DOE/DOD high energy physics program. Table I - RSM Processing and Technologies '90-'95 shows that in 1995, the block melting process consumed half of the total, with direct disposal less than 5%. Decontamination share remained about half the total.

Table I

Disposal costs have increased by a factor of more than three during 1990-1995 for both commercial and government disposal services. This factor of three or more increase has made metal decontamination and recycling via melting more economical than burial. Commercial use of technologies which have been made more cost effective due to the high cost of direct disposal, include metal melting and recycling, and extensive use of new chemical decontamination technologies for metal cleaning.

Fig. 1

MARKET FORCES -- 2000

How much metal will be available in the market? The answer to this question is dependent on the amount produced, the technologies and cost and the decisions and policies that will be made by the DOE and to a lesser extent, the NRC-NEPA.

Technology Available

Over the next five years, repackaging and direct disposal of metals is expected to take a minimal share of the overall metals market. Repackaging, however, will be technologically enhanced by the utilization of metal melting to produce optimum waste density prior to burial. Companies currently in the metal melting business are Manufacturing Sciences, SEG, Nuclear Metals and Aerojet who have existing metal melting capacity. Of these, all will have commercially-available operations by early 1996.

Commercial Production of RSM

Commercial waste generation from 1996 to 2000 will remain essentially flat as compared to the previous six years. Metals from commercial sources will continue to consist of piping, structural items, and decommissioned buildings with a sporadic input of decommissioning wastes from nuclear power plants or industrial facilities. The commercial market

is expected to remain flat with approximately 20,000 tons per year including the large scale decommissioning wastes. Not included in this volume estimate are large components such as steam generators or reactor vessels which are expected to be handled as direct disposal items.

Government Production of RSM

Government metal volumes, while large in inventory, may be relatively small in annual availability due to funding constraints and policy decisions from both the DOE and EPA/NRC. The DOE is developing a policy for metals recycling that is consistent with public safety and cost effectiveness. If approved and implemented, this could result in a annual metals volume that exceeds 50,000 tons per year. Initiatives for metals mining of gaseous diffusion plants and nuclear weapons production facilities may drive high-value metals into the market, providing significant sources of RSM.

In subsequent sections of this paper, the factors that will effect the volume of DOE metals entering the marketplace are examined.

METAL RECYCLING INTO PRODUCTS

Metal recycling to date has been limited to either decontamination and unrestricted release of metals into the scrap metal market or production of shielding blocks for the high-energy physics program. Over the next five years, a market will be developed for high-quality containers manufactured from radioactively-contaminated steel alloys. These containers will be primarily of stainless steel due to the availability of nickel for the stainless alloy and the high market cost of clean stainless. A smaller percentage of the containers will be manufactured out of carbon steel, but due to the higher cost of radioactive container fabrication versus clean containers, only specialized applications of these carbon steel containers will be utilized.

Table II -- DOE Metal Container Consumption Forecast for 2000 shows the annual requirements of the DOE for both stainless and carbon steel in the year 2000. DOE is likely to be the sole customer for radioactively-contaminated steel containers. These containers will be utilized primarily for low-level waste disposal containers. In some cases, specialized uses of stainless steel containers, i.e., for TRU waste will be piloted on a demonstration basis during the time period. Large scale utilization of vitrification, storage or transportation casks or other large consumer markets for radioactive steel recycling will not materialize until after the year 2000, due to the developmental lag time in these demanding applications.

Table II

Due to cost constraints, radioactively-contaminated container manufacturing is expected to occupy approximately 1/5 to 1/3 of DOE's annual radioactive waste containerized waste requirements. This will be produced from RSM removed from DOE decommissioning sites that cannot be economically cleaned for unrestricted use.

UNRESTRICTED USE MATERIAL

Standards and limits are in place now allowing for unrestricted use of materials into the commercial metal recycling market. Criteria for release of materials for recycling in an unrestricted manner, e.g., conventional scrap metal recycling, have been based on criteria established by the Nuclear Regulatory Commission in the mid 1970's. These criteria are the same as Reg. Guide 1.86, criteria for unrestrictive release of facilities. The same criteria have been used to apply to release of materials from recycling operations. Functionally, this

criteria equals to 100 counts per minute above background with a thin window g-m detector. For alpha contamination, this translates to a rate of approximately 2 cpm above background using a alpha probe. In 1995, approximately 10,000 tons of metal was recycled from the industrial processors providing this service. An additional 2,000 tons was released directly by commercial generators.

Over the next five years, the amount of material released is expected to remain essentially constant. Unless standards are changed (see below), there will not be any new developments in technology that significantly change the amount of material that would be released.

RISK BASED UNITS

Table III, Numeric and Risk Based Limits, compares the existing and future standards for contamination on and in recycled metals. Over the next five years, individual states, and possibly the Federal Government, will adopt standards that are not based on numerical limits like Reg. Guide 1.86, but based on dose to man or risk-based limits. Tennessee, by virtue of its pre-eminent position in regulating the radioactive waste processing industry, is leading other states (with the exception of possibly Texas) in the development of risk based limits. By the year 2000, it is likely that Federal guidance on this issue will be forthcoming. This will help the development of large scale RSM container recycling for restricted use within the radioactive waste industry. States such as Tennessee will adopt risk based limits that allow limited release of radioactive contamination on metal but produce insignificant doses to the public, e.g., less than 1 mR/year to the maximally-exposed individual. This action by individual states will promote full scale metals recycling industries.

Table III

In the worst case, the standards will remain the same which would preclude widespread recycling of metals into container feed stocks.

If the Federal Government acts quickly and establishes risk based limits before the year 2000, the impact on the metals recycling industry would be large. A standard on the order of two pico curies/gram could increase the available processing capability of the industry significantly by providing an outlet for low-cost, high-quality RSM.

COST FOR DECONTAMINATION OR PROCESSING

The cost of decontamination or recycling of metals varies widely depending on the material type, contaminant and physical configuration of the metal itself. Surface contaminated items composed of large carbon steel or similar materials with a large mass to surface area ratio can be decontaminated and surveyed for unrestricted use for less than $1/lb., excluding disposal of reject wastes. Typical materials composed construction materials, stairways, cabinets, piping and similar items with a medium surface to weight ratio cost approximately $1.50/lb. for decontamination and free release. Complex items such as small diameter piping, cabling, thin gauge steel cost $1.85/lb or more. Heavily contaminated items, items which have ingrained contamination and require aggressive surface removal, chemical cleaning or sectioning, frequently have a decontamination cost of $2.00-$2.50/lb. including survey.

All of these figures are offset somewhat by the scrap value of the material. The various classes of decontamination costs and scrap values are shown in Tables IV and V, respectively. Table IV, Processing Cost Projections and Market Pricing 1995, shows the class of material and its

approximate decontamination cost. Table V, Scrap Values, shows the current market scrap value for various material types.
Table IV
Table V
In the year 2000, these costs are expected to be about the same as in 1995. It is assumed that the rate of inflation will remain very low during this period and that the value of the scrap metals will remain essentially flat. Costs which are increasing, such as labor and disposal, will be offset by production efficiency gains resulting in a net annual cost increase of less than 5%.
Melting and disposal or recycling costs are currently in the $1.70 to $2.50/lb. range. The low end represents low activity materials which are already cut into sizes which fit the existing furnace dimensions. The high end represents heavily contaminated austenitic steels which must be decontaminated, cut and blended into the melt to meet the specifications. To date, the only significant recycling activity has been the production of shield blocks for the DOE. In the future, radioactive waste containers will become a significant portion of the market on an economic basis while comprising a somewhat smaller share on a mass basis.
DOE POLICY
The DOE has a fundamental policy decision to make regarding the recycling of radioactive metals. If the DOE adopts a recycle policy, including the disposal of most, if not all radioactive metals, the market for radioactive metals recycling will be very large. Conversely, if the Department bases all recycling decisions on economics alone in terms of current cash cost, it is unlikely that any significant volume of metals recycling into useable products will occur.
It is likely that the DOE will adopt a policy for RSM recycling to the extent that it can be accomplished for a cost which is within 10% of all available alternatives. This will result in some products being manufactured from radioactive metals but probably not more than 20% of the metals currently available for recycling. The production and utilization of low-cost items, made from carbon steel, is probably not feasible based on economic cost models developed by The Association for Radioactive Metals Recycling (ARMR). Where a high value component is needed, such as stainless steel drums, the economics associated with recycling and utilization of these containers is more favorable as compared to the straight option of burial and procurement of new, nonradioactive containers.
MARKET FORECAST -- 2000
With the adoption of a DOE policy for recycling and issuance of a risk-based standard for metals recycling, the government sector of the RSM market would be able to produce 50,000 to 100,000 tons of metal per year for recycling. This forecast projects that only about half of this would actually be available, due to economic and technical constraints. Figure 2 - RSM Production Forecast with Policy and Regulations-- 1996-2000, shows the result of these projections.
Fig. 2
The processing technologies that will make up this industry will still largely rely on conventional means, e.g., decontamination or melting for volume reduction and burial. The government's policy on recycling of RSM containers will drive only the high-value, stainless steel containers to be economical.

Figure 3, RSM Processing Share Forecast with Policy and Regulations, shows large share of government origin decon and smelting business that grows significantly between 1997 and 1999.
Fig. 3
FINANCIAL PROJECTIONS
Using the pricing and assumptions listed above, the 1996 RSM market (excluding direct disposal costs) is a $90 million dollar per year industry. The growth, due to a favorable DOE policy and risk-based regulations, to the year 2000 would increase this industry by roughly $100 million or by the year 2000, a RSM industry would generate $200 million per year in revenues.
Employment in the RSM industry in 1995 is estimated to be about 450, using an average of $200K of revenue per employee. The growth in this industry will result in an industry employment of 1000, a doubling in size by the year 2000. Table VI, Revenue Projection for RSM Processing -- 1996-2000 shows the components of revenue and the dramatic impact of government processing on the total. In the year 2000, government processing will account for about 60% of the industry's revenue.
Table VI
CONCLUSION
The RSM industry in the year 2000 will be between $100 and $200 million per year with an employment of 500 to 1000 people directly involved in processing RSM. The factor of two variation is dependent on the outcome of federal government action to establish RSM recycling policy and standards. The industry has poised itself to respond to the large, potential demand. Based on current technology and experience, the possibility of a $200 million per year industry is realistic and achievable.
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RECYCLING FOR PROFIT THROUGH QUANTUM-CEPTM
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ABSTRACT
M4 Environmental L.P. provides the DOE a unique opportunity to commercially recycle mixed waste or convert depleted uranium hexafluoride (UF6) while simultaneously purifying radioactive scrap metal (RSM). No other Low-Level mixed low-level (LLMW), or RSM processing technology offers this unique recycling/conversion solution to the multiple waste challenges faced by DOE. The dissociation and partitioning capabilities

inherent to Quantum-CEP allow the creation of useful products from the elemental constituents of a multitude of wastes. This co-processing approach will allow safe, economical disposition of the majority of DOE wastes. The life cycle costs associated with co-processing RSM, LLMW, and depleted UF6 utilizing the Quantum-CEP technology are anticipated to be significantly lower than individually processing these wastes/materials through separate, standard industrial processes.

Potential RSM feedstocks for co-processing LLMW and depleted UF6 include: 1) approximately 200,000 tons of contaminated ferrous and non-ferrous metals in the current national DOE inventory (1); 2) barrier nickel in the intact buildings that house the gaseous diffusion operations; 3) PCB-contaminated ductwork, copper instrument tubing and copper wire (with and without insulating encasement) present in the still-intact gaseous diffusion plants; 4) aluminum from compressors and other process equipment in the still-intact gaseous diffusion plants; and 5) ferrous alloys present in the still-intact gaseous diffusion plants.

LLMW streams such as inorganic sludges, organic sludges and liquids, soils, scintillation fluids, inorganic debris, batteries, laboratory packs and reactive metals can be co-processed with the RSM. Products from the co-processing of LLMW include a synthesis fuel that can be used as either a fuel or a chemical feedstock; ingots of RSM for 1) feed material for the fabrication of radioactive waste containers; 2) shield blocks for DOE operations; or 3) free-release into the European Community or other areas that have existing release standards; and a stable ceramic phase that contains most of the radioactive contaminants from the LLMW and RSM, which can either be reused within the DOE or disposed of at a site such as Envirocare's facility in Utah.

Review of DOE's inventory of depleted UF6 indicates approximately 50,000 cylinders containing more than 1.2 billion pounds of material that can be co-processed with radioactive scrap nickel, copper, aluminum or ferrous alloys. Products generated from the Quantum-CEP conversion of the depleted UF6 include anhydrous hydrogen fluoride (AHF), aluminum fluoride, a reusable uranium oxide ceramic, a reusable uranium-iron alloy for feedstock to the Atomic Vapor Laser Isotope Separation (AVLIS) uranium enrichment process, and purified ingots of the RSM.

M4 ENVIRONMENTAL L.P.

M4 Environmental Management L.P. (M4), formed in August 1994, is a limited partnership between Lockheed Martin Corporation and Molten Metal Technology, Inc. (MMT). The company has the exclusive license to provide Mat's proprietary CEP and Quantum-CEP technologies to the U.S. Department of Energy, the U.S. Department of Defense, and the United States Enrichment Corporation for processing hazardous, radioactive, and mixed wastes into useful products. Quantum-CEP refers to CEP technology when applied to radioactive material.

M4 recently announced the commissioning of its second Quantum-CEP system, a commercial system which processed DOE LLMW in December 1995. M4 activated its first system to process depleted uranium hexafluoride at the M4 Technology Center in October 1995. The completed Technology Center will have required an approximately $40 million capital investment and will house several CEP and Quantum-CEP systems that have a range of nominal molten metal capacities from 20 to 3000 lb. The company also was recently selected as one of three finalists, out of 23 competitors, for the U.S. Army's Alternative Technology Chemical Demilitarization Program.

OVERVIEW OF CEP/QUANTUM-CEPTM

Quantum-CEP is an adaptation of CEP technology for radioactive and mixed waste streams. Quantum CEP allows both destruction of hazardous components and controlled partitioning of radionuclides. This leads to decontamination and recycling of a large portion of the waste components to commercial products as well as volume reduction and concentration of radionuclides for final disposal.

At the core of both CEP and Quantum-CEP technology is a molten metal bath that acts as a catalyst and solvent in the dissociation of the feed, the synthesis of products and/or the concentration of radionuclides in the desired phase. Upon introduction to the bath, feeds dissociate into their constituent elements and go into metal solution. The critical criteria for evaluation of feeds for processing, therefore, lie not in their molecular structure, but rather in their elemental makeup, since most materials will be dissolved through an elemental intermediate regardless of molecular structure at introduction. Once in this dissolved elemental state, the addition of co-reactants enables reformation and partitioning of desired products by either remaining in the molten metal phase, moving to the ceramic phase directly above the metal, or exiting the reactor in the gas phase. The partitioning control afforded by co-reactant addition is a distinguishing feature of CEP and Quantum-CEP.

The versatility of the CEP process is attributed to the 'singular' dissolved elemental intermediate through which reactions proceed. For example, the metal bath ensures that in the dissociation of an organic feed, dissolved carbon (C) is the only carbon intermediate. This is in contrast to open flame systems, in which multiple free radical intermediates can be formed, leading to hard-to -control side reactions. While a traditional non-catalytic technology (e.g. incineration) operates in a strongly oxidizing atmosphere and hence produces large quantities of carbon dioxide and water vapor, the strongly reducing atmosphere of CEP produces carbon monoxide and hydrogen gas, with a CO:CO2 ratio on the order of 10,000:1.

Given the key role of the metal bath as a solvent, the thermodynamic forces governing the solubility of elements in liquid metal baths are important in understanding how the operating conditions of the CEP system can be manipulated to synthesize desired products of commercial value. The reducing strength of dissolved carbon is an important factor in CEP's flexible recycling ability. Specifically, CEP provides the flexibility to recycle organics to synthesis gas, recover metals and inorganics, such as halides, sulfur and phosphorus, and separate and concentrate compounds of the actinide elements. The dissolution of carbon in the liquid iron solvent ensures a homogeneous reducing environment and, hence, robust control of product composition and quality.

MELT PURIFICATION OF RSM

CEP radioactive waste processing and recycling has been studied extensively by MMT under a DOE-sponsored Program Research & Development Award. This $38 million research program was designed to investigate the "Recycle of Contaminated Scrap Metal." One of the program focus areas was radionuclide partitioning, for which 152 tests were performed, consuming 1,390 hours of hot metal time on the bench-scale, pilot-scale and demonstration-scale CEP systems in the Fall River, Massachusetts facility. Hafnium was used as a surrogate for the actinides in these studies, which involved partitioning in both molten iron and nickel. In iron, the lower detection limit (LDL) for hafnium is 0.2 ppm; in nickel it is 1.74 ppm. Results consistently showed residual radionuclide

surrogate concentrations in the matrix metal that were below the LDL with no radial or axial gradients identified. Additional partitioning tests were performed with similar success on the bench scale at Scientific Ecology Group's Quantum-CEP facility, using both cerium and uranium as contaminants.

DOE national laboratories have extensively researched the decontamination of metals using melt refining/slagging ("passive partitioning"). The documentation of these experiments indicates that among all of the ceramic phase compositions tried, the best results occurred with the use of the borosilicate and high silica. For instance, decontamination factors (DFs) between 600 and 1,100 were achieved for the removal of uranium dioxide (UO2) and (PuO2) from nickel with these ceramic phases (1). Note that here DF is defined as the initial weight fraction of contaminant in the as-received RSM divided by the final weight fraction of contaminant in the purified RSM. Quantum-CEP enhances decontamination through continuous processing, selective reactant addition, superior mass transfer characteristics, unique reactor configurations, and alternate materials selection ("active partitioning"). Therefore, higher DFs than those benchmarked from the previous DOE "passive partitioning" work are anticipated.

A thermodynamic analysis of melt purification can be used to approximate the magnitude of the maximum theoretical separation under equilibrium conditions where the partitioning of the contaminants is to an oxide-based ceramic phase. A favorable theoretical separation is a "green light" to proceed, but the calculated result will not likely be achieved in practice because of system nonidealities. The Quantum-CEP system under consideration consists of a molten metal bath, an oxide-based ceramic phase floating on the molten metal, and a head space of oxygen. These phases are assumed to be in equilibrium. The equilibrium state of a closed system is that state for which the total Gibbs energy (G) is a minimum with respect to all possible changes at the given temperature and pressure. This means that at equilibrium, the change in the Gibbs energy (DG) is zero. For a given chemical reaction aA + bB = cC + dD, the expression for the change in the Gibbs energy is given as:

Eq. 1

where

1. DGo is the standard Gibbs energy of formation at a given temperature
2. ai is the activity of the chemical species = gixi.
3. R = 8.314 J mol-1 K-1.
4. T = temperature.

The two reactions (for RSM with one contaminant) that are at equilibrium in this system are (matrix metal = Mm; contaminant metal = Cm)

Eq. 2

Eq. 3

Because DG1 = DG2 = 0 at equilibrium,

Eq. 4

To determine the maximum theoretical DF, rearrange the above equation to obtain an expression for aCm. Then, use the following set of assumptions to simplify:

1. aCm = gCmxCm (where gCm is the thermodynamic activity coefficient for the contaminant metal in the matrix metal, and gCm is the mole fraction of contaminant metal remaining in the matrix metal after melt purification).

2. aCmxO2y and aMmxO2y in the ceramic phase are approximated by xCmxO2y and xMmxO2y, their respective mole fractions.
3. aMm may be approximated as unity.
4. T is the temperature of the system.

Solve for xCm, and convert to the appropriate weight fraction using the appropriate molecular weights. Finally, an assumption must be made as to the value of gCm.

M4 TECHNOLOGY CENTER

Overview

M4's state-of-the-art Technology Center is located in a light industrial/commerce park in Oak Ridge, TN. The three-story facility has more than 75,000 ft2 of factory floor space and 25,000 ft2 of office space. Quantum-CEP systems in the M4 Technology Center are designated as Radioactive Processing Units (RPUs). The following provides a description of each of the systems that have been installed or are being installed in the facility.

Description of LLMW Facilities

RPU-2

RPU-2 provides a bench-scale facility for research and development and treatability studies. Each unit is constructed with feed preparation equipment (excess water removal, etc.), two replicate Quantum-CEP test units, a common gas mixing system and three redundant gas handling trains (GHTs).

The gas delivery set-up is capable of mixing three reagent gases and an inert gas for submerged lance injection into a molten metal or metal and ceramic phase system. The gas composition is controlled using individual flowmeters and a gas mixing chamber, and can be directed to either test unit via a three-way valve at the exit of the gas mixing chamber.

On-line analysis of the process gas stream during waste injection is performed with, for example, a Total Hydrocarbon Analyzer (THC) mass spectrometer, CO/CO2 analyzer, etc. The on-line analysis allows quantification of the gas phase product quality for comparison to larger Q-CEP systems. Additionally, metal and ceramic phase samples can be taken during operation. Post-run elemental analysis is performed on metal, ceramic, containment, head space, etc., for material balances.

RPU-3

RPU-3 is a refractory-lined Catalytic Processing Unit (CPU) which can be fed either through a submerged lance, a lock hopper or a bottom tuyere. A tuyere is a concentric pipe mounted from the bottom of the reactor where flowable feed with a small nominal particle size (i.e., < 1 mm) and co-feeds are injected into the reactor. The submerged lance and lock hopper are typically used for adding bulk solids to the CPU. A sludge/ slurry injection system is under development to enhance the capability of RPU-3 to process a wide range of materials. The RPU-3 has a nominal molten metal capacity of 800 lb. A prototype ceramic and metal tapping assembly has been developed for the RPU-3. This system allows for continuous removal of the two phases from the CPU on a continuous basis. Throughput for RSM may be up to 60 lb/hr in this system.

Thermocouples (for measuring gas and/or metal temperature), off-gas plumbing and reactor relief stations are installed at the top of the reactor. A containment monitoring system is also installed. An angled sample port is provided for intermittent contact measurement of temperature as well as withdrawal of ceramic and metal samples. Product gases are withdrawn from the reactor through a gas handling train (GHT).

RPU-4

RPU-4 is rated for the largest waste throughput of any processing unit in the Technology Center. The CPU is an inductively heated, refractory-lined steel pressure vessel with a nominal molten metal capacity of 3,000 lb. The unit is equipped with ceramic and metal tapping capabilities that allow up to 1,400 lb/hr of RSM to be decontaminated.

A gas handling train will allow off-gas cooling and collection of dust including volatile heavy metals and metal halides. The ability to separate alkali halides from the lead, zinc, and other volatile heavy metals will be accomplished through as dual-stage cooling and filtration train. A small purge stream containing the volatile heavy metals is required for the system. Appropriate polishing steps, such as carbon bed absorption and caustic scrubbing, are utilized to ensure off-gas quality prior to being sent to storage or flare system. The carbon bed is fully recyclable to the CEP system after saturation.

Planned melt purification work to be done in these three CEP systems in 1996 includes co-processing commercial mixed waste (chlorinated and/or fluorinated organic liquids) and nickel with 2.1 wt. % ThO2. Other potential RSM feedstocks for co-processing LLMW in 1996 and out years include: 1) approximately 200,000 tons of contaminated ferrous and non-ferrous metals in the current national DOE inventory; 2) barrier nickel in the intact buildings that house the gaseous diffusion operations; and 3) ferrous alloys present in the still-intact gaseous diffusion plants. It is expected that the Quantum CEP Technology will be economical for the processing of mixed wastes in comparison with other available technologies; by using RSM in the processing, the economics will be in hand.

UF6 Program Background

In May 1995, M4 signed a one-year contract with the USEC to demonstrate the applicability of Quantum-CEP to convert UF6 process tails to a stable uranium form (i.e., UO2/U3O8 or bulk uranium metal) with reuse of the fluorine. A successful technology demonstration and a decision to proceed would lead to the construction of one or more production facilities to convert USEC's stored and newly generated tails. This would provide a solid experience base for pursuing the DOE market, reported to be 556,000 metric tons of UF6, as well as the international market.

The experimental plan has been constructed with three process flowsheet options. In the baseline approach, the uranium content of the UF6 is converted to either UO2 or U3O8, and the fluorine content is recovered as anhydrous hydrogen fluoride (AHF). A general representation for this process is:

Option 1 (Baseline): UF6 + H2O + CO2 = UOx + AHF + CO + H2

In Option 2, a metal oxide and a carbon source are co-fed to the CPU to produce either UO2 or U3O8, the fluoride of the metal that was fed as an oxide, and synthesis gas (CO + H2). The option differs from the baseline in two respects: it feeds a solid material (such as CaO, MgO, or Al2O3) and it does not need the elaborate gas recovery system required for AHF since the fluorine values are recovered as a salt (CaF2, MgF2, or AlF3). This option is expected to have less attractive economics than the baseline approach, which is the focus of planned development and demonstration activities. A general representation for this process is:

Option 2 : UF6 + MOy = UOx + MFz + CO + H2

Finally, as a third option, a reactive metal (Al, Mg, or Ca) can be fed into the process as a reductant, producing a uranium alloy with the bath

metal (i.e., iron) and the fluoride of the reactive metal. A general representation for this process is:

Option 3: $UF_6 + M = U + MF_x$

The dense, stable storable uranium oxide produced in the first two process options can either be sent to a disposal site or made into products for shielding. The uranium alloy produced in Option 3 could be used for such things as AVLIS metal feed or shielding. The bath metal for Options 1 & 2 may be copper, nickel or iron, which can be provided from RSM in DOE inventories; the preferred bath metals for option 3 are iron or carbon steel, which can yield an alloy that melts significantly lower than either iron or uranium alone. Option 3 is unique in that it would allow the use of radioactive scrap aluminum from idle gaseous diffusion plant equipment (such as compressors) as feedstock for conversion to AlF_3. The aluminum fluoride could be collected from the gas phase so that, with proper filtration, it is free of radioactivity and is of very high purity (i.e., 99% minimum). Currently, AlF_3 is available commercially in purities of 92% and 97% and has a value of around $0.50/lb. Also, in a production plant designed to convert the DOE UF_6, the UF_6 cylinders will provide significant metal input and up to 70,000 tons of steel product for resale from Quantum-CEP operations. By feeding excess bath metal to the process, Quantum-CEP purified RSM can be cast and made available for additional reuse.

Specific RSM feed being lined up for RPU-1 includes: 1) copper contaminated with depleted uranium from the H-1 Foundry at the Oak Ridge Y-12 Plant; 2) a 180-lb Monel UF_6 cylinder from the K-25 Site in Oak Ridge; 3) aluminum compressor blades contaminated with slightly enriched uranium also from the K-25 Site; and 4) barrier nickel from the gaseous diffusion plants (either part or all of a 100-lb ingot or "cornflakes." The cost of Quantum CEP conversion of depleted UF_6 are expected to be in lined with quoted competitive conversion process cost (two dollars to eight dollars/kg of UF_6); RSM addition to the process will further lower conversion costs.

Description of UF_6 Demonstration Facility

RPU-1

The M4 RPU-1 experimental facility incorporates the general capability to feed gaseous UF_6 and various reactants (co-feeds) necessary for the chemical conversion of UF_6 (e.g., H_2O, H_2, O_2, CaO, and CH_4) at prescribed rates, temperatures, and pressures to a CPU and analyze the resulting off-gas in near real time. HF generated by the conversion process is chemically reacted with chemical sorbents for disposal. The capability to sample the metal melt and ceramic phase is available, but bulk removal is not. The design allows quantitative operation of the CPU at pre-established process conditions for sufficient periods of time (e.g., one to six hours) to achieve steady state and collect meaningful process performance and material balance information. The overall design allows quick turnaround of the experimental equipment to accommodated two test periods per week. Each test period can determine the steady state results for several values of an experimental variable.

RPU-1 consists of six major components: 1) Feed System, 2) CPU, 3) Off-Gas System, 4) Purge System, 5) Analytical and Control Instrumentation, and 6) Containment. A summary description of the facility is given below.

The Feed System consists of an electrically-heated environmental enclosure for generating a UF_6 vapor feed, gas manifolds for various gaseous components, steam generator, and solid feeder. Solid co-feeds

(e.g., Ca, UO3, CaO, Al, and CaF2) may be batch charged to the CPU (along with the initial metal charge) as well as lance fed to the CPU in a continuous mode from a feed hopper.

The CPU is a crucible-based reactor designed by MMT specifically for UF6 service. The unit will operate with a variety of molten metals, including iron, nickel and copper. The nominal temperature range is 1000o to 1700o C. The crucible design will accommodate a nominal 45 kg of metal and will allow the collection of sufficient ceramic product to support a six-hour experimental campaign. Multiple tests may be performed for each experimental campaign. The CPU can be operated under a broad range of processing conditions to enable parametric mapping of process performance for the assumed reaction scenarios. The CPU operates at essentially atmospheric pressure. Metal and ceramic phase tapping is not provided although the design allows metal and ceramic phase sampling.

The Off-Gas System consists of a ceramic process filter to remove entrained metal and ceramic particulates from the CPU, chemical traps to quantitatively react the CPU-generated HF and residual UF6, and HEPA filters for final particulate removal.

The Purge System consists of a cold trap, chemical traps, and HEPA filter to remove bulk process gas from the experimental system after completion of an experimental campaign.

The Analytical and Control System consists of near real time, on-line instruments for the quantitative analysis of process gas for process control and process performance determination. Off-line analytical equipment is provided for analysis of the ceramic and metal phases. In this case, the crucible will be removed from the CPU after completion of an experimental campaign and bisected for quantitative analysis. X-ray fluorescence or energy dispersive x-ray will be used to provide quantitative elemental distributions in the crucible, metal and ceramic (as well as any interface). Scanning electron microscopy (SEM) will be utilized to identify any distinct phase regions of interest in the crucible, metal, ceramic and any interfaces. Electron probe microanalysis (EPMA) will be used to determine the chemical identity of the distinct phases identified by the SEM.

The Containment System provides multiple physical barriers to double and triple contain radioactive gas and solids, as well as flammable and toxic process components, at the source in the event of a primary process breach. The Feed System, CPU, Off-Gas System, and Purge System are located in separated, independently ventilated, secondary containment structures. The CPU will be readily removable from the containment structure for post-mortem examination. A hooded work area is also provided for disassembly of the crucible after a test. All components of the facility are located in a third containment structure for ultimate environmental isolation and protection of workers. HF, H2, and CO monitors with alarm circuits are provided in the various containment structures to alert the operator to process leaks.

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PARTNERING BETWEEN DOE SITES AND COMMERCIAL UTILITIES TO REDUCE COSTS

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ABSTRACT

The Westinghouse Savannah River Company (WSRC) is partnering with a number of commercial nuclear power plants in order to directly implement utility best practices as a means of improved treatment, storage and disposal of low-level radioactive waste. Initial focus has been on development and implementation of an aggressive pollution prevention/waste minimization program. Through a series of benchmarking visits and technical exchanges, WSRC has identified and begun implementation of a series of changes to work practices and material/product specifications that are expected to result in substantial cost savings and extension in the life of on-site disposal facilities. The utility participants in this partnership have been exposed to a number of technologies with significant applicability to current and planned operations at commercial nuclear plants. This partnership is consistent with the Department of Energy's (DOE) initiative to utilize commercial technologies/practices to an increased degree and the U. S. electrical industry's continuing practice of cost reduction and economic competitiveness.

INTRODUCTION

The Savannah River Site (SRS) is a key DOE facility focusing on national security work, environmental and waste management activities, and economic development and technology transfer initiatives. Owned by the DOE and operated under contract by the Westinghouse Savannah River Company (WSRC), the complex covers 310 square miles near Aiken, South Carolina.

From its construction and initial operation in the 1950s until the early 1990s, SRS operated in a manner consistent with its national defense mission. There was minimal contact with the commercial world, with all process-related research and development performed on site. Production goals, and employee and public health and safety were paramount. Cost effectiveness was secondary. The situation was similar at the other sites in the DOE Weapons Complex.

With the fall of the Soviet Union in the early 1990s, the DOE complex began a transition period with an often traumatic effect on its mode of operations. Site missions changed from production to environmental management, plans were made to completely shut down some facilities, budgets were cut as the Federal Government struggled with an increasing budget deficit, and thousands of workers were laid off as funding for

their projects vanished. DOE and its contractors quickly recognized that a basic change in operating culture was required both for basic survival and to complete remaining missions within available budgets. This change in operating culture involved, among others:

- privatization
- outsourcing
- use of commercial best in class technologies
- change in basic structure of operating contracts

The cost effectiveness/savings initiatives undertaken by DOE and its contractors mirror the efforts begun by the U.S. electrical utilities with commercial nuclear plants in the early 1980s. Specifically in the area of low-level radioactive waste management, the commercial power plants were faced with spiraling costs for waste disposal. Burial costs were increasing steadily from a few dollars per cubic foot in the late 1970s to several hundred dollars per cubic foot today. This problem was compounded by the formation of regional compacts which presented the possibility that the majority of generators would be denied access to available disposal capacity.

As the cost of disposal of low-level radioactive waste became an increasing part of the nuclear plant's operating budget, this portion of the operation came under increased scrutiny as the utility endeavored to reduce costs and improve its situation in what was quickly becoming a highly competitive electrical generation and distribution industry. Strategies were developed and implemented to:

- reduce waste generation at the source improve the packaging of waste that is generated
- decontaminate material where practical
- volume reduce waste volumes to the maximum extent
- develop storage concepts where required

IMPLEMENTING A COST REDUCTION CULTURE AT SRS

The Solid Waste Management Department (SWMD) at SRS is responsible for the treatment, storage, and disposal of all solid waste streams generated at SRS with the exception of high-level waste stored in underground tank farms. The disposition of these waste streams are as follows:

- hazardous waste is shipped off site for incineration at a commercial facility
- mixed waste is stored on site awaiting operation of the Consolidated Incinerator Facility
- transuranic waste is stored on site awaiting opening of the Waste Isolation Pilot Plant
- sanitary waste is disposed of at an off-site commercial landfill
- low-level radioactive waste is disposed of in on site engineered disposal vaults

As part of its program to reduce the cost for treatment, storage and disposal of the site's solid waste, SWMD has pursued some of the more traditional approaches including:

- privatization of sanitary waste disposal
- development of needed technologies under the Vendor Forum Program
- awarding of a contract for off-site volume reduction of low-level radioactive waste
- utilization of subcontracts with commercial vendors for selected tasks

In addition to these initiatives, it was recognized that there was a significant source of proven, world-class best practices within easy grasp of SRS; the 100 or so commercial nuclear plants. These plants

readily share their best practices among themselves without regard to utility ownership. Discussions with several of these plants found them willing and eager to share their best practices with SRS as well.

The SWMD at SRS therefore formed an informal Waste Management Partnership with a number of commercial nuclear plants. The objective of the partnership was to identify and directly incorporate as many utility best practices as possible at SRS that would assist our cost reduction initiative. To ensure effectiveness of this partnership, certain classical paradigms had to be overcome at SRS:

What we've done for the past 40 years has worked and is safe. Why change now?

Utility practices are not as technically rigorous as we require.

We have a lot of alpha contamination which invalidates many utility practices.

We've got to study it some more.

The areas in which SRS could benefit from the partnership are:

waste minimization/source reduction strategies and practices

decontamination strategies/practices

volume reduction techniques

container selection and design

waste storage

treatment, storage and disposal of greater than Class C waste

filtration, demineralization and other treatment of dilute liquid radioactive waste streams

hazardous chemical control programs

waste characterization and certification programs

reuse/recycle of radioactive scrap

Although the cost of low-level radioactive waste disposal at SRS does not provide the same hard economic driver as exists in the commercial nuclear industry, radiological performance requirements for low-level radioactive waste disposal facilities have resulted in an almost doubling of unit disposal costs. This results from a 1995 change from shallow- land burial to disposal in engineered vaults. The need for a new focus on pollution prevention/waste minimization at SRS resulted in this area becoming the initial focus of the Waste Management Partnership. SRS benchmarking teams therefore visited the following commercial nuclear plants:

Susquehanna Steam Electric Station near Berwick, Pa.

Beaver Valley Nuclear Station near Pittsburgh, Pa.

McGuire Nuclear Station near Huntersville, N.C.

Plant Vogtle near Waynesboro, Ga.

V.C. Summer Nuclear Station near Jenkinsville, S.C.

These plants are but a few of the plants with demonstrated track records of significant reduction in the volume of low- level radioactive waste generated and disposed of over the past ten years. During the benchmarking visits, plant personnel openly discussed their programs, what had worked (and what hadn't worked), the need for senior management commitment and support, employee education and motivation, and opportunity identification and assessment. Plant walk downs were key ingredients of the visits as most of the members of the benchmarking team were representatives of SRS' larger waste generators.

WHAT THE PARTNERSHIP HAS MEANT TO SRS

The Waste Management Partnership has been the cornerstone for the successful development and implementation of SRS' pollution prevention/waste minimization program. The Partnership will also

represent the engine for continuous improvement of the program. Whereas a typical commercial nuclear plant has only a handful of individual waste generators, a DOE site the size and complexity of SRS has approximately 55 individual waste generators. The development and implementation of a site-wide pollution prevention/waste minimization program is therefore faced with a number of unique hurdles and barriers.

The Partnership provided SRS' SWMD with a litany of utility success stories that were crucial in receiving senior management buy in and commitment. The information gained from the Partnership was used to develop a brochure entitled "Savannah River Site, The Commercial Connection" that was targeted at the site's division vice presidents. The brochure described the successes and cost reductions achieved by the utilities and described the techniques and work practices that, when applied at SRS, would result in an overall reduction in SRS' radwaste budget. The utility success stories also provided operating experience to overcome skepticism that implementation of common sense changes to operating practices could have a significant impact on radwaste volumes and disposal costs. The Partnership clearly allowed SRS to immediately benefit from 10 years of utility field work.

Following the example of the commercial nuclear industry, SRS has formed a Solid Waste User's Advisory Board to represent the focus of site-wide implementation of pollution prevention/waste minimization initiatives. As a result of the Waste Management Partnership, the following initiatives are being pursued (both on a pilot and across the board basis) at SRS facilities:

- Segregation of Radiation Buffer Area (RBA) Waste - Green is Clean program
- Segregation of all waste types at point of generation
- Elimination/reduction of throw away items (especially plastic and tape)
- Maximize use of launderable items; e.g., waste bags, rags, etc.
- Select materials of construction consistent with incineration in the Consolidated Incinerator Facility
- Designation of contaminated tool cribs
- Clearance of tools in regulated areas with an automated tool frisker
- Decontamination of tools
- Elimination of double bagging of LLW

Once fully implemented, it is estimated that these programs have the potential to reduce waste generation volumes by 30% and reduce annual disposal costs at SRS by $2-3 million. These efforts will also extend the life of the site's engineered disposal vaults, thereby delaying the need for scarce capital project funding.

In addition, as SRS and other DOE facilities intensify their recycling efforts, the utilities represent a potential end user of the "products" manufactured from the recycled materials.

WHAT'S IN IT FOR THE UTILITIES?

A reasonable question is, what can the commercial nuclear plants learn as part of this Waste Management Partnership? The answer is quite simple and straightforward: additional opportunities for cost effectiveness/reduction. Although the mission of a commercial nuclear plant and a facility at SRS or other DOE site differ, the tail end of the nuclear fuel cycle offer numerous areas of commonality as discussed earlier in this paper. The lifting of the veil of secrecy at SRS and other DOE sites offers the utilities the opportunity to investigate those best practices in these areas of commonality. In addition, nuclear plant

personnel frequently become involved in issues such as off-site incineration and design and operation of regional disposal facilities that directly affect the operation of their plants. The exchange of design and operating information on SRS facilities such as the low level radioactive waste disposal vaults and the Consolidated Incinerator Facility will assist nuclear plant personnel perform this part of their responsibilities.

In the case of Susquehanna Steam Electric Station, sufficient interest was generated during the SRS benchmarking trip in May 1995 that two utility personnel visited SRS in August 1995. The utility personnel were interested in learning more about SRS' solid waste operation with specific emphasis on disposal vault design and design and operation of the SRS Consolidated Incinerator Facility. The utility representatives' visit to SRS was also used to allow them to walk down two of the site's larger generators and critique their waste minimization practices. Once again, by working together, the DOE facility and the electric utility were able to develop a win-win situation.

PARTNERING BEYOND THE SAVANNAH RIVER SITE

There are geographically convenient commercial nuclear plants available to most DOE facilities. These commercial plants offer numerous opportunities to the DOE facilities for partnering on day-to-day operations issues as well as, one- time events such as decommissioning. With many DOE facilities actively undertaking D&D programs, the potential for exchange of world class practices and techniques do exist.

Another partnership that has developed at a Westinghouse operated DOE facility is the one between the waste management organizations at the Hanford DOE site and the commercial nuclear plant operated by the Washington Public Power Supply System (WNP) on the Hanford reservation. WNP personnel had initially participated in the annual Westinghouse GOCO Waste Management Operations Workshop at Hanford in November 1994. This relationship has now grown into one that has proven beneficial to both organizations. WNP personnel actively participate in Westinghouse Hanford Company's monthly pollution prevention meetings. WNP personnel have learned from the DOE pollution prevention program and is implementing many of the philosophies, such as pollution prevention opportunity assessments, recycling/reuse and materials substitution to minimize waste generation. Additional Westinghouse Hanford has provided WNP assistance in development of procedures and programs for control of chemicals and management of hazardous waste.

In return, WNP has shared their initiatives on radioactive waste minimization with Westinghouse Hanford. The use of anti-contamination clothing designed to eliminate waste generation are being considered both at Hanford and West Valley. The use of launderable tarps to replace herculite and other plastic sheeting is also being pursued.

The partnership between WNP and Westinghouse Hanford has and will continue to provide opportunities for exchange of information with mutual benefit. The partnership has provided the key vehicle -- an open line of communications.

CONCLUSION

Yes Virginia, DOE facilities can learn from commercial nuclear plants and commercial plants can learn from DOE facilities. Once the NIH (not invented here) factor is overcome, partnering between DOE facilities and commercial nuclear plants provides a powerful vehicle for exchanging best practices as both cultures strive for improved cost competitiveness.

Partnering between the SWMD at SRS and waste management organizations at a number of commercial nuclear plants has provided the impetus for the SRS pollution prevention/waste minimization program that is expected to result in significant cost savings. The partnering commercial plants have gained information and knowledge needed to further make their operations cost effective. This partnering can be expanded to any DOE/utility pair as shown by the Hanford/WNP experience.

As we all look to make our operations more cost effective and competitive, remember, put the NIH monkey back in its cage and look beyond your culture for opportunities for continuous improvement.

ACKNOWLEDGMENT

This work was sponsored by the U. S. Department of Energy Office of Environmental Restoration and Waste Management under DOE Savannah River Operations Office Contract No. DE-AC09-89SR18035.

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AN INDUSTRY RESPONSE TO RECYCLE 2000

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ABSTRACT

The United States Department of Energy (DOE) is expected to issue a policy early this year articulating DOE's position on the recycle of DOE radioactive scrap metal. In anticipation of this "Recycle 2000" initiative, the nuclear industry has formed a new trade association called the Association of Radioactive Metal Recyclers (ARMR). This article describes the Recycle 2000 initiative, provides some background on ARMR and its membership, and identifies industry views on the actions to be taken and issues to be resolved if Recycle 2000 is to become a reality.

RECYCLE 2000

The U.S. Department of Energy (DOE) has a huge volume of radioactive scrap metal (RSM) in inventory at sites across the DOE complex. In March 1995, an inventory report prepared for the Office of Technology Development documented the existence of 157,501 tons of contaminated carbon steel at thirteen (13) sites (1). Others have estimated that, when metal generated during the future dismantlement of contaminated DOE facilities is included, the total volume of DOE RSM could reach 3 million tons!

In December 1994, the "Recycle 2000" concept was articulated by DOE at a meeting in Denver in conjunction with a diverse group of stakeholders including citizens groups, environmental organizations, unions, industry and various government organizations. The overall objective was to determine stakeholder support for DOE to remove RSM from the waste disposal stream and "beneficially reuse" the DOE metal in applications where its small radioactivity content would not be a detriment. Specifically, DOE proposed that, by the year 2000, at least 50 percent of the low-level waste disposal containers (B-25 boxes) used by DOE would be fabricated from DOE-generated radiologically contaminated carbon steel defined as RSM. Such a proposal would recycle natural resources (metal)

and vastly reduce the DOE disposal volume that would otherwise be required if the radioactive scrap metal was buried.

Since this successful Recycle 2000 concept introduction, other meetings have been conducted by DOE in Salt Lake City and Knoxville with considerable emphasis placed on, 1) developing a cost model comparing the cost of recycling RSM into waste containers to the cost of burial, and 2) evaluating the dose received (risk) by industry workers and the general public if the Recycle 2000 concept was implemented.

There appears to be considerable support for recycling not only in industry but throughout DOE and EPA. Consider the following:

USDOE Assistant Secretary for Environmental Management, Thomas P. Grumbly, has stated that, "It is our policy to reuse or recycle these scrap metals to the maximum practical extent" (2).

USDOE Director of Environmental Restoration, James Owendoff, expressed in a Weapons Complex Monitor interview a preference to use waste disposal containers fabricated from recycled DOE metal as opposed to containers made from virgin steel, even at a large cost premium (3).

USDOE Director of the Office of Eastern Area Programs, Jim Fiore, has stated, "Pollution prevention and waste minimization activities such as scrap metal recycling are supported and encouraged by Headquarters" (4).

USDOE Director of the Waste Minimization Division, J. Kent Hancock, estimated that between $28 to $43 billion could be saved within the complex over the next 75 years by implementing pollution prevention programs including recycling (5).

The USEPA has established pollution prevention as a "national objective" requiring that pollution be prevented or eliminated at the source wherever feasible, recycled when prevention is not feasible, and disposed of only as a last resort.

Presidential Executive Order #12780 requires that DOE "promote cost-effective waste reduction and recycling of usable materials from waste generated by Federal Government activities," and that DOE "integrate these programs to assist in addressing the nation's solid waste disposal problems."

Because of the strong support for recycling both within industry and government, the industry consensus is that the Recycle 2000 Policy will be adopted by DOE and promulgated in early 1996.

ASSOCIATION OF RADIOACTIVE METAL RECYCLERS (ARMR)

In 1995, strong industry interest in the beneficial reuse concept and in DOE's Recycle 2000 initiative spawned the formation of a new trade association called the Association of Radioactive Metal Recyclers (ARMR), headquartered at the University of Tennessee in Knoxville.

ARMR is a trade organization of the holders, decontamination processors, metal fabricators and end users of recycled radioactive scrap metal. Through the activities of the Association, the Members seek to stimulate and support the movement of radioactive scrap metal to useful products with the following specific objectives:

Coordinate the exchange of information among the Members regarding Radioactive Scrap Metal (RSM).

Foster the identification of reuse products, including the free release of decontaminated metals.

Support the development of the industry infrastructure to permit the beneficial reuse of RSM.

Encourage research into cost-effective recycle of RSM.

Advocate regulatory practices and national standards supportive of RSM recycle.

Promote the recycle of RSM as an environmentally favorable and cost effective alternative to disposal.

Hold or participate in industrial society meetings to promote the recycle of RSM.

Provide information to the interested public and speak with the single voice of a trade association.

Advance other activities supportive of the purpose of the Association.

Since its formation, ARMR has worked closely with DOE in refining the Recycle 2000 policy. In an October 13, 1995 letter to ARMR, Tom Grumbly, DOE Assistant Secretary for Environmental Management, stated that "We welcome your (ARMR) thoughts on how to implement that decision (Recycle 2000) efficiently and quickly. We also encourage you to share your ideas on how the National Recycle Program might be expanded in the future." Although ARMR has a strong focus on DOE, the organization will also address material generated by the commercial nuclear power industry and the fabrication of many different products in addition to low-level waste containers.

A current list of ARMR organizational members include:

- Advanced Recovery Systems
- Alaron
- Allied Erecting and Dismantling
- American Technologies, Inc.
- Brandenburg Industrial Service Co.
- CDM Federal Programs
- Carolina Metals
- Corpex Technologies
- Fernald Environmental Restoration Management Corporation (FERMCO)
- Hake Associates
- Lockheed Martin Energy Systems, Inc.
- MSE Western Environmental Technology Offices
- M4 Environmental Management, Inc.
- Manufacturing Sciences Corporation
- Molten Metal Technology
- Princeton Plasma Physics Lab
- Scientific Ecology Group
- University of Tennessee
- U.S. Ecology
- Westinghouse Savannah River Company

The formation of ARMR and the list of organizational members should leave no doubt in anyone's mind that industry is ready to support the DOE Recycle 2000 effort!

INDUSTRY IS READY!

Recycling options available today for radioactive scrap metal include both decontamination/free release of metal and metal melting/beneficial reuse.

Decontamination/Free Release Capability

The nuclear services industry not only strongly supports the Recycle 2000 concept but has already made considerable capital investment in preparing to respond to DOE for policy implementation. In fact, nuclear industry emphasis on waste volume reduction beginning in 1980 has spawned a number of companies that specialize in removing contaminated material, including contaminated metal, from the waste disposal stream. These companies are

typically licensed by agreement states, states authorized by the Nuclear Regulatory Commission to issue radioactive material licenses, and permitted to receive radiologically contaminated metal for subsequent decontamination, survey, and free release in accordance with NRC Regulatory Guide 1.86 and DOE Order 5400.5.

A list of companies currently operating commercial facilities offering decontamination and free-release service is as follows:

ALARON	Pittsburgh, PA
ATG	Richland, Washington
Hake Associates	Memphis, TN
Manufacturing Sciences Corporation	Oak Ridge, TN
SEG	Oak Ridge, TN
US Ecology	Oak Ridge, TN

The metal most suitable for cost-effective decontamination and free-release normally has a high mass to surface area ratio with easily accessible surfaces for decontamination and survey. Following metal decontamination and free release, the metal is typically sold to commercial scrap metal dealers.

Metal Melting/Beneficial Reuse Capability

In direct support of the Recycle 2000 initiative, industry has recently expanded its metal recycling capability going beyond simple decontamination/free release. Specifically, a number of companies now offer metal melting services for that metal that cannot be easily decontaminated -- a capability required to support Recycle 2000. In fact, a true niche marketplace has developed with considerable competition.

SEG

In 1990, SEG was the first company to offer commercial metal melting services to process metal not suitable for economic decontamination, that is, metal with low mass to surface areas, inaccessible surfaces or otherwise difficult or impossible to decontaminate.

The melting process reduces the bulked volume of radiologically contaminated scrap metal by a factor of approximately 15. The process also concentrates most of the RSM's radioactivity in the "slag" which is removed for burial. Since the resulting metal product still retains a small amount of volumetric radioactivity, the product cannot be "free released" under current regulations in the United States. SEG has contracted with Los Alamos to provide shield blocks fabricated from this metal - an application where the slight volumetric contamination of the metal was not a detriment to the products intended use - shielding (6). SEG currently has the capacity to melt and process more than 15,000 tons per year.

In addition to its metal melt facilities in Oak Ridge, SEG has acquired and now operates a container fabrication facility in Carlsbad, New Mexico.

SEG's effort has been followed up and expanded by other companies in an attempt to anticipate the commercial opportunities growing out of the Recycle 2000 concept.

MANUFACTURING SCIENCES CORPORATION (MSC)

MSC, in a joint venture with British Nuclear Fuels Limited (BNFL), recently completed construction of a new, 115,000 square foot metal recycling facility in Oak Ridge. The facility features automated indoor storage of all incoming contaminated scrap metal, remotely operated material handling, semi-automated mechanical and chemical decontamination processes and vacuum induction melting. The new plant will operate in

conjunction with MSC's existing metal rolling and fabrication facility. Estimated plant capacity for complete metal recycle is 10,000 tons per year.

MSC is ideally positioned to respond to Recycle 2000 in the short-term since the company possesses not only the capability to melt RSM into ingots but also to roll the ingots into metal sheet and fabricate waste containers.

MSC also has a contract with DOE to convert four buildings at Rocky Flats to commercial use for RSM recycling. These buildings collectively contain nine vacuum induction melting furnaces and two rolling mills. This so-called National Conversion Pilot Project, when completed in 1997, will be offered by DOE to commercial enterprises and be available to support DOE Recycle 2000 efforts.

CAROLINA METALS

Carolina Metals operates a metals processing facility in Barnwell, SC which is licensed to handle radioactive materials. In 1995, the company worked with MSC in a demonstration to fabricate stainless steel containers for Westinghouse Savannah River Company using 20 tons of volumetrically contaminated heat exchangers, cooling water piping, and slug buckets from activities at SRS. The company has an estimated capacity to melt 1,000 tons of RSM annually.

Carolina Metals currently does not provide any decontamination services but has teamed with Alaron to melt materials which they have generated in their decontamination activities.

RECOMMENDED ACTION

With industry infrastructure in place, it's now time for DOE to consider the privatization of DOE recycling efforts. Privatization would enable commercial companies to recycle DOE metal operations under NRC or Agreement State rules - and not be bound by DOE Orders which frequently increase cost. The concept has many advantages:

Speed - DOE recycling efforts can move forward quickly if DOE will fund large recycling projects by commercial companies, essentially privatizing the function in support of Recycle 2000 objectives.

Cost Reduction - A large-scale privatization effort will provide significant manufacturing "economies-of-scale" resulting in reduced DOE costs. In addition, DOE contractor labor costs now incurred in fragmented recycling efforts and in the custodianship of existing RSM piles would be reduced.

Backlog Reduction/Contaminant Migration Mitigation - The huge DOE scrap metal backlog could be eliminated quickly, reducing storage costs and the continued migration of contaminants into the environment from numerous DOE scrap metal piles.

Visible Cleanup Progress - Real cleanup progress will be visible at a time when visible progress is very important.

The strongest argument for action is very pragmatic. Many of DOE's scrap metal piles have been in existence for well over a decade. For example, the Oak Ridge Scrap Metal Program was established in 1986, but the scrap metal piles are still there, in outdoor uncovered storage, with the metal corroding and with small amounts of radionuclides escaping into the environment. DOE has the ability to recycle this material now! The RFP issued in November

1995 by Lockheed-Martin Energy Systems in Oak Ridge for the recycling of approximately 710 tons of RSM is a small first step supporting DOE recycling efforts. A national procurement contract should be considered

so that all sites can participate in the achievement of Recycle 2000 objectives. Features of a national procurement contract could include: 1) standardization of terms and conditions. 2) specification of the new family of standard DOE waste disposal containers, and 3) definition of secondary waste disposition responsibility.

KEY ISSUES

Despite broad support within government and industry for the Recycle 2000 effort and industry readiness to support DOE, there are a number of issues that appear to be impeding DOE recycling efforts.

Funding

Current federal budget realities in general and DOE budget cuts in particular put recycling efforts in competition with other cleanup activities for limited dollars. With funding limited, the decision is frequently made to let RSM piles grow in size and deal with them later. If funding for recycling is not provided, it will be difficult or impossible for the Recycle 2000 policy to succeed.

Recycle Cost Premiums

Budget constraints are related to the "recycle cost premium". In many cases, it costs more to recycle metal than to simply bury it. The problem is particularly acute for those DOE sites planning on-site disposal cells where the recycle cost premium can be especially significant. Sometimes the issue is whether to recycle or take what appears to be fiscally expedient action, at least in the short-term.

Life Cycle Costs

The size of the "recycle cost premium" depends heavily on the true life cycle cost of waste disposal. A recent study performed by the Oak Ridge National Laboratory's Center for Risk Management for the DOE's EM Waste Management organization stated that the total annual cost of waste disposal at DOE sites is, in most cases, considerably higher than the burial charge used in most recycle vs. disposal cost comparisons. For example, the full-cost of disposal at the Nevada Test Site in FY-95 was determined to be $24 per cubic foot, double the published disposal price (7). In fact, the weighted average DOE disposal price in FY95 was calculated to be $38.50 per cubic foot. By understating real disposal costs, economic analyses comparing recycle to disposal will almost always favor disposal.

Decentralization of DOE Recycling Efforts

DOE recycling efforts are underway at a number of sites across the complex including Oak Ridge, Savannah River, Rocky Flats, INEL and Fernald. Due to the nature of DOE management and operation (M&O) contracts, these positive M&O efforts are often uncoordinated, overlapping and thus unable to take advantage of the substantial "economies of scale" that would result from large-scale DOE procurements to rid the complex of unwanted RSM inventories. It is recommended that DOE centralize recycling activities, utilizing a National Procurement Contract to assure Recycle 2000 policy implementation.

CONCLUSION

The DOE Recycle 2000 Policy has wide support, both in industry and government. Industry has invested millions of dollars in the infrastructure required to support DOE recycling efforts. Now is the time for DOE to move ahead to eliminate the Department's huge backlog of radioactive scrap metal.

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INCENTIVES FOR RECYCLING IN THE U.S. DEPARTMENT OF ENERGY

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ABSTRACT

A 1994 change in public law (P.L. 103-329, Section 608) enables Federal agencies to receive and use funds from the sale of materials recovered through recycling or waste prevention programs. The law states that these funds must be used for specified environmental programs or other authorized employee programs. Numerous recycling programs exist across the complex; however, revenues generated from these programs are either not captured or managed, are being used to fund unrelated programs (e.g., employee incentive programs), or are being donated to charities. This change in public law provides a valuable incentive for agencies to maximize the recycling of wastes otherwise bound for disposal, especially in light of recent and projected Federal budget cuts. This paper discusses the provisions of the public law, the Department of Energy (DOE) recycling progress, and incentives for recycling within DOE.

PUBLIC LAW 103-329, SECTION 608

The change in public law occurred in 103rd session of Congress and was reauthorized by the 104th Congress. (P.L. 104-52 was signed by President Clinton on November 19, 1995). The law states that, in addition to other funds provided to agencies, Federal agencies can receive and use funds from the sale of materials recovered through recycling or waste prevention programs for three specific purposes:

1. Acquisition, waste reduction and prevention, and recycling programs as described by Executive Order 12873, "Federal Acquisition, Recycling, and Waste Prevention," including any such programs adopted prior to the effective date of the Executive Order (October 20, 1993).
2. Other Federal agency environmental management programs, including but not limited to, the development and implementation of hazardous waste management and pollution prevention programs.
3. Other employee programs as authorized by law or as deemed appropriate by the head of the Federal agency.

DOE RECYCLING REVENUE

Based on recycling information contained in the draft 1993 "Annual Report on Waste Generation and Waste Minimization Progress," DOE facilities recycled 60,717 metric tons sanitary waste (33 percent of the sanitary waste stream). Recyclable materials were grouped into four main categories: precious metals, scrap metals, paper, and other (aluminum cans, glass, plastic, Polystyrene, toner cartridges, batteries, engine oil, wood, tires, food waste, and concrete).

The amount of material diverted for recycling in 1993 increased 248 percent compared to 1991. This increase can be attributed to an increase in recycling activity, improved tracking systems, and an increase in the number of materials being tracked. Using current market conditions for scrap metal, paper, and other recyclables, these materials could generate over one million dollars if sold on the open market to local recycling processors. This revenue could be as much as two to three times higher if DOE facilities market directly to recycling mills, rather than going through intermediate recycling processors.

An informal survey of DOE facilities showed that waste recycling revenue often goes into a general overhead account at the facility where it is difficult, if not impossible, to track how it is spent. A few sites reported that revenue is returned to the U.S. Treasury. In some cases, the revenue is used to pay for the cost of the recycling program or offset future purchasing costs. Most sites reported that revenue from the sale of aluminum cans is used by employees (i.e., donated to charity). Previously, all revenue from Federal recycling programs administered by the General Services Administration (GSA) was donated to day care centers operated for the Federal employees. Decisions regarding the use of recycling revenue are currently made at the discretion of the sites.

INVESTMENT IN RECYCLING

The Department of Energy is continually searching for ways to minimize the wastes and prevent wastes from being generated in the first place. The "1995 Baseline Environmental Management Report" showed that the eventual costs of dealing with the wastes generated across the DOE complex over the last 40 years will be hundreds of billions of dollars unless certain things are done to reduce the waste streams from routine operations and dismantlement activities. Recycling is one means to reduce the volume of DOE waste and typically represents activities that are "doable" for the near term.

High Return on Investment Projects

In 1994, the DOE Pollution Prevention Executive Board approved funding for 17 pollution prevention projects that were identified as providing a high Return on Investment (ROI) to the Department. Of these 17 initial (Round I) projects, five were eventually funded through other mechanisms or rescoped, and 12 have gone on to implementation. The expected net present value savings to the Department (ten years after project completion) are more than $28 million.

The ROI approach has proven to be a successful mechanism for fostering pollution prevention activities within the Department. In 1995, the Office of Pollution Prevention solicited proposals from the field for a second round (Round II) of ROI proposals. The Environmental Management organization has approved 22 projects for funding with an anticipated net present value savings of about $109 million. Table I provides a project summary and the projected ROI, costs and net present value savings for the six high ROI projects that involve recycling.

Table I

SUMMARY

Executive Order 12873 defines recycling as "the series of activities, including collection, separation, and processing, by which products or other materials are recovered from the solid waste stream for use in the form of raw materials in the manufacture of new products other than fuel for producing heat or power by combustion." In the past, recycling activities relied almost totally on the extra effort by a few champions who wanted to save resources and "do the right thing" environmentally. There were few, if any, resources available from upper management to make the programs work cost-effectively. Recent management emphasis on self-supporting program activities has given "new life" to the Department's recycling efforts.

Within DOE, the Office of Pollution Prevention (EM-77), is responsible for coordinating implementation of Executive Order 12873. Waste Minimization/Pollution Prevention coordinators at the field office level are responsible for incorporating recycling into their facilities' operations. Recent budget cuts have resulted in a substantial reduction in the Office of Pollution Prevention's overall budget in FY 1996. The opportunity to obtain additional revenue from previously unavailable sources is a powerful incentive for the Department to maximize the efficacy of its recycling programs.

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Co-chairs: Connie Cicero, WSRC;
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ION EXCHANGE PERFORMANCE OF COMMERCIAL CRYSTALLINE SILICOTITANATES FOR CESIUM REMOVAL*

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ABSTRACT

A new class of inorganic ion exchangers called crystalline silicotitanates (CST), invented by researchers at Sandia National Laboratories and Texas A&M University, has been commercialized in a joint Sandia-UOP effort. The original developmental materials exhibited high selectivity for the ion exchange of cesium, strontium, and several other radionuclides from highly alkaline solutions containing molar concentrations of Na+. The materials also showed excellent chemical and radiation stability. Together, the high selectivity and stability of the CSTs made them excellent candidates for treatment of solutions such as the Hanford tank supernates and other DOE radwastes.

Sandia National Laboratories and UOP have teamed under a Cooperative Research and Development Agreement (CRADA) to develop CSTs in the powdered form and in an engineered form suitable for column ion exchange use. A continuous-flow, column ion exchange process is expected to be used to remove Cs and other radionuclides from the Hanford supernatant. The powder material invented by the Sandia and Texas A&M team consists of submicron-size particles. It is not designed for column ion exchange but may be used in other applications.

IONSIV IE-910 CST powder has been manufactured by UOP in commercial quantities and tested under a wide range of conditions. Cesium distribution coefficients (Kd) of up to 1,000 mL/g are measured in DSSF-5 simulant, a Hanford waste that is 5M in Na, and 0.5M in K and has high concentrations of OH-, NO-3, NO-2, and other ionic species. The engineered form of IONSIV IE-911 CST has been prepared by UOP under commercial manufacturing conditions and tested for cesium Kd in batch and kinetic tests. Tests with Hanford DSSF-5 and related simulants resulted in a 50% breakthrough of Cs of about 550 column volumes (CV) at a flow rate of 3.75 CV/hr. A column test on IONSIV IE-911 using an ORNL Melton Valley Tank W-27 simulant also resulted in a 50% breakthrough of 500 CV at a flow rate of 3 CV/hr. Column tests on INEL groundwater simulant with a flow rate of 10 CV/hr showed no Cs breakthrough to 4500 CV and excellent performance for removing Sr.

The experimental results have been integrated with an effort at Texas A&M University to model the CST's equilibrium and kinetic behavior. For the

IONSIV IE-910 exchanger, the removal of Cs and Sr from solution can be estimated to within about 10% for a variety of solutions ranging from 2M acid to 6 M hydroxide. The model also predicts the effect of high concentrations of potassium, sodium, and rubidium on the selectivity for Cs. Kinetic models have been developed that accurately estimate the breakthrough curve through the 50% breakthrough point for IONSIV IE-911 as conditions are varied.

Data are also presented confirming the excellent stability of the commercial CSTs over a broad pH range and the high radiation stability of IONSIV IE-910 and IE-911 exchangers. In addition, data are provided that demonstrate the high physical strength and attrition resistance of IONSIV IE-911, critical properties for column ion exchange applications.

INTRODUCTION

A new class of inorganic ion exchangers, invented by researchers at Texas A&M University and Sandia National Laboratories, has been successfully commercialized by UOP. The materials, crystalline silicotitanates (CSTs), demonstrate high cesium distribution coefficients in acidic, alkaline, and neutral, solutions despite the presence of high concentrations of competitive ions such as Na+ and K+ (1,2). The affinity for Sr++ in neutral and alkaline wastes is also found to be high (2). The CST is stable in Hanford tank waste simulants for long time periods and to exposure to at least 109 rads (2).

UOP and Sandia teamed under a Cooperative Research and Development Agreement (CRADA) to develop commercial CST materials. UOP commercially-synthesized IONSIV IE-910 CST powder demonstrates high capacity and selectivity for Cs+ and Sr++ over a wide range of pH conditions and competitive ion concentrations. In simulated and actual waste tests, Cs+ distribution coefficients are high over a pH range of 0 to 14 in the presence of varying concentrations of Na+ and K+. Distribution coefficients for Sr++ are also high at neutral and alkaline pH. The chemical stability of the IONSIV IE-910 exchanger is also demonstrated at pH from 0-14, and no effect on performance is noted after radiation exposure to 109 rads.

The UOP-Sandia CRADA was targeted toward the development of a commercial engineered form of the CST (beads, pellets, or granules) to meet the requirements anticipated for the removal of Cs, Sr, and other radionuclides from the Hanford alkaline tank waste supernates. These requirements included high selectivity and capacity for Cs+ and Sr++ from highly alkaline, high-sodium waste solutions, that have good radiation and chemical stability. The column ion exchange process likely to be used at Hanford for radionuclide removal also requires an ion exchanger that can be easily loaded and unloaded from the columns and that provides rapid ion exchange kinetics with acceptably low pressure drop and good mechanical strength characteristics. A final critical requirement was that the ion exchanger needed to be compatible with final waste forms, such as borosilicate glass, that are likely to be used at Hanford. An ion exchanger meeting all the above requirements has the potential to offer significant waste-treatment cost savings. Because other DOE sites also have a need for radionuclide removal, the product developed under the CRADA was to be as broadly applicable as possible. Work is presented that shows that UOP IONSIV IE-911 ion exchanger has met these criteria in a wide range of test conditions.

Application Conditions; High pH, High Salt, Low Cs

The principal target use for the CST ion exchanger is in the remediation of liquid wastes containing radioactive Cs from DOE operations. UOP has developed this material into commercial products: the CST powder is IONSIV IE-910 ion exchanger, and the engineered form is the IONSIV IE-911 ion exchanger.

The primary waste-treatment application is at Hanford, Washington, where 177 tanks containing highly radioactive waste are currently stored. Most of the gamma radiation results from 137Cs. A significant amount of radiation is also due to the presence of 90Sr. The liquid portion of this waste has a wide range of compositions, but all of it can be characterized as high pH (typically pH 11.5 to 2 M OH-) with high dissolved salt concentrations and a low molar concentration of Cs. The concentrated salt solution contains several cationic metals (mostly sodium and potassium), which compete with Cs for ion exchange sites and can affect both the selectivity and the ultimate capacity of the exchanger for cesium. The waste also contains significant concentrations of anions including NO-3, NO-2 and Al(OH)-4.

Three typical waste liquids; Double Shell Slurry Feed (DSSF), Neutralized Current Acid Waste (NCAW), and Complexant Concentrate (CC). are highly concentrated sodium salts (mostly nitrate and nitrite) with low concentrations (at the parts per million level) of Cs. The NCAW tends to have a relatively high concentration of 137Cs but comparatively low potassium concentration. The DSSF has relatively high potassium concentration, while CC tends to have more residual organic material, including some original complexing agents combined with their hydrolytic and radiolytic by-products.

The overall objective of the remediation project is to reduce the volume of High Level Waste (HLW) for disposal. The approach to be used is to extract the Cs from the concentrated salt solution by selective ion exchange and, thereby, convert the treated solution to Low-Level Waste (LLW) for easier disposal. The concentrated 137Cs along with the other Cs isotopes would then be vitrified and placed into long-term storage as HLW.

Product Requirements: Capacity/Selectivity, Stability; IE-911

The need to produce a minimum volume of concentrated 137Cs from a dilute Cs solution, in the presence of high concentrations of competing metal cations, sets severe demands on the capacity and selectivity of the ion exchanger chosen. In addition, the ion exchanger needs to operate in the high pH environment without loss of its high capacity and selectivity and physical integrity. Finally, the exchanger needs to operate without loss of performance resulting from the effects of prolonged exposure to high levels of radiation.

These requirements are all taxing. Nevertheless, the CST invented by Sandia and Texas A&M University (TAMU) and commercialized by UOP as the IONSIV IE-910 and IE-911 ion exchangers has these characteristics and can deliver the performance needed to ensure the successful decontamination of the Hanford waste.

A standard method of exchanger evaluation involves determination of the equilibrium distribution coefficient, or Kd value. The Cs Kd is dependent on test conditions such as pH, ion concentration, and relative abundance of competing ions. The Maintenance of a high Kd value in the operating environment is critical to achieving the maximum reduction in waste volume. As will be illustrated later, the IONSIV IE-910 and IE-911

products maintain high Cs Kd values across a broad range of conditions spanning the expected range at Hanford.
The distribution coefficient, Kd, is calculated using the equation:
Eq. 1
Kd: distribution coefficient
Cs: equilibrium activity in the ion exchanger solid
Cl: equilibrium activity in the liquid
Ci: initial concentration of the ion of interest in the liquid
Cf: final concentration of the ion of interest in the liquid after the period of contact V: solution volume
M: mass of ion exchanger used, as received basis
F: F-factor = ratio of (mass of dry exchanger) to (mass of as-received exchanger)
Figure 1 shows the effect of pH on the Cs distribution coefficient (3). The Kd value decreases noticeably as pH becomes increasingly alkaline. However, even at the highest pH, CST maintains a high coefficient. The high selectivity for Cs at low pH enables it to resist acid leaching.
Although the principal focus of the IE-910 and IE-911 development was the selective removal of radioactive Cs, (Fig. 2), the CST is also effective for Sr exchange at alkaline pH (3). This capability was also highlighted in later studies with simulated Hanford ground water, neutral ground water (INEL simulant), Oak Ridge waste, West Valley waste, and also in DSSF actual waste at PNL (4).
Fig. 1
Fig. 2
The abundance of Na+ present in the Hanford solutions would be expected to compete with Cs+ for the ion exchange sites within the CST. Increases in both (Na+) and Na:Cs ratio are expected to influence the ion exchange selectivity for Cs+. This effect is illustrated in Figs. 3 and 4 for both the IE-910 CST powder and the IE-911 (engineered form) (3). The actual solutions at Hanford have high Na:Cs ratios and tend to have high molar concentrations of Na+. An important feature of CST is that it maintains a high selectivity even under these extreme conditions. All of the data in this paper have been obtained with representative simulants at room temperature.
Another significant, or competing, cation present in the Hanford waste is potassium. Potassium would be expected to have an ion exchange affinity more similar to that of Cs and, therefore, a more competitive influence on Cs+ selectivity (distribution coefficient). This effect is illustrated in Fig. 5 for the IE-910 powder (3). In this comparison, the (Na+) in the DSSF simulant is at a typical concentration representative of much of the Hanford waste, and the (Cs+) and (K+) are varied across the typical ranges expected. As expected, the presence of potassium reduces the Cs Kd values. The effect is most pronounced at low (K+), but further increases in (K+) don't appear to have much additional impact on selectivity. As noted previously with the effect of both pH and Na:Cs ratio, the IONSIV ion exchanger maintains a high selectivity in the presence of realistic concentrations of competing potassium that is representative of actual wastes.
Fig. 3
Fig. 4
Fig. 5
Because of the wide range of solution compositions and the competing effects standardized tests were established for use during ion exchanger

development: Batch and Column tests. These tests allowed rapid, accurate comparison of relative exchanger selectivities, capacities, and ion exchange kinetics.

Batch Ion Exchange tests were used to evaluate comparative equilibrium Cs Kd values at various times. The test consists of contacting a small amount of ion exchanger (typically about 0.1g) with a fixed amount of waste simulant (typically about 10mL), such as DSSF, and then vigorously shaking or swirling the mixture. Experiments are established for several time durations to evaluate the rate of approach to equilibrium. The simulant composition is prepared in the laboratory to be representative of its respective waste type. Results of this testing are expressed as Cs Kd values vs. time. Comparisons between materials can be made using their equilibrium values (data > 24 hr). The Kd values obtained are expressed on an exchanger weight basis (mL solution per gram exchanger). This method can provide a crude estimation of large differences in mass transfer efficiency. However, if small differences in physical properties exist between ion exchanger samples or subtle differences in mass transfer rates occur, then the Column Ion Exchange test is more representative of performance.

Comparisons can be made between materials of widely differing physical properties. For example, if the density of an exchanger is known, then a volumetric distribution coefficient can be estimated. This "Cs-lambda" (units: mL of solution per mL) value is derived by simply multiplying the exchanger density by its respective Cs Kd value. Comparison on this basis can estimate the relative waste-volume reduction achievable by each ion exchanger being evaluated, a key factor for consideration. However, a value for the exchanger density must be representative of its density in the waste solution. Many organic resins change density depending on their degree of hydration and solution pH. The IONSIV IE-911 does not change density with hydration or solution pH.

Performance of IE-910 and IE-911 Ion Exchangers

Several different ion exchangers (Table I) have been under evaluation for this application. Each was tested using the standard Batch Ion Exchange method.

The results are summarized in Fig. 6 (4). The figure shows Cs-lambda values, which provide a relative volumetric capacity for the ion exchangers listed. The key features to note are the values at 24 to 72 hours, where all materials have achieved equilibrium (note that the Cs-lambda values are shown on a log scale). A comparison of equilibrium Cs-lambda values shows that both IONSIV IE-910 and IE-911 ion exchangers achieve similar high distribution coefficients compared with all of the other candidates evaluated. Comparison of IONSIV IE-911 with the parent IE-910 powder shows that it has a slightly lower value because of the presence of the inert binder acting as a diluent.

Fig. 6

At test times much less than 24 hours, an obvious difference shows in the apparent Cs-lambda value for the IE-910 powder and the IE-911 30x60 mesh material. This apparent difference in performance is almost entirely due to differences in mass transfer efficiency caused by the particle size. The IE-910 powder particles are approximately 0.2 microns in size. The size for the 30x60 mesh IE-911 material is a nominal 250 to 600 microns. Evaluation of ion exchangers is usually reported on a weight basis (as is the data in this paper). The UOP IE-910 and 911 ion exchangers outperform all of the other candidates. The SuperLig 644 is the next best performer.

However, in the type of column ion exchange system envisioned for Hanford, the low density of the SuperLig is a detriment. This candidate would require additional column volume (therefore, capital cost) because of the low density.

Comparisons of how regenerable organic and nonregenerable inorganic ion exchangers are integrated into radwaste processing are a complex issue. The regeneration of organic ion exchangers will generate large volumes of acidic waste that must be concentrated and processed for vitrification. This waste may require interim storage. Disposal of the resins is another consideration. The present flow sheet calls for the spent organic ion exchanger to be decontaminated and sent to the low level glass plant. There it must be integrated into the redox chemistry of the wastes being fed to the LLW glass melter.

In contrast, the IONSIV IE-910 and 911 CST materials are not designed for regeneration and will permit the design and operation of a much-simpler facility. Studies to-date show that the IE-911 composition is compatible with high-level waste vitrification. This inorganic exchanger also seems compatible with extended lag storage options.

In addition to Batch Ion Exchange evaluations, the second type of test uses a small-scale simulation of the column ion exchange process. This Column Ion Exchange test also uses simulated waste liquids as a feed (actual waste has been used but is less typical). Multiple columns are sometimes used in series to simulate the large-scale process.

Figures 7 and 8 illustrate the performance of a sample of IE-911 exchanger (3,5). The feed compositions in each of the illustrations are representative of the different waste streams. The figures show the relative effluent Cs+ concentration for a nominal 10 ppm Cs in the simulant feed plotted against the cumulative amount of feed processed expressed in relative Column Volumes (CV). Exchanger capacity is estimated by observing the point at which the relative effluent concentration reaches 50% of the feed concentration. Assuming that the breakthrough curve is approximately symmetrical, the 50% breakthrough point gives an estimate of the amount of feed solution that can be treated at equilibrium saturation of the exchanger.

In both illustrations, with DSSF or Melton Valley simulants, the example shows that a preliminary development sample of IE-911 exchanger has the capacity to treat more than 500 CV of waste (at 10 ppm Cs) before becoming saturated. Testing at Oak Ridge with actual Melton Valley waste confirmed good, though somewhat lower, performance with 350 CV treated to 50% breakthrough in W-27. Retesting at Sandia National Labs has confirmed that the difference between Melton Valley simulant and actual waste data resulted from a higher sodium concentration and higher pH in the W-27 waste. This performance has been further improved to 660 CV capacity in DSSF with an improved material (Fig. 9). This high capacity coupled with high selectivity demonstrates a level of performance sufficient to allow use without regeneration or further concentration. Eliminating the need for regeneration or elution offers significant cost savings in the construction of the large-scale unit.

Fig. 7

Fig. 8

Fig. 9

Figure 7 also includes predictions of column ion exchange performance from the model developed at Texas A&M University (5). The experimental

results closely match the prediction and demonstrate the utility of the model of Zheng , Anthony, et. al..

Samples of IE-911 exchanger have also been evaluated with lower pH West Valley waste and with contaminated ground water. Although still alkaline, both of these applications are at significantly lower pH, than the Hanford type wastes. The lower pH would be expected to further improve the performance of IE-910 and IE-911 beyond that already discussed because of the large increase in Cs Kd values that accompany decreases in pH. A test at West Valley with actual waste (pH 11.5) demonstrated more than 400 CV capacity without breakthrough to give a Cs decontamination factor of 105-106 (3). Tests with Neutral Ground Water simulant (pH 8 to 9) also demonstrated decontamination factors of more than 500 even though the Cs+ feed concentration was only 50 ppb.

In addition to the high Cs decontamination factors, significant Sr removal occurred in both the West Valley actual waste and Neutral Ground Water studies. The Neutral Ground Water study saw a slight breakthrough of Sr with a decontamination factor more than 1800. Testing with the actual West Valley waste gave a decontamination factor of 104. A modest removal of uranium and plutonium cations, represented by a decontamination factor of about 100, also occurred.

Kinetics and Particle Size

In addition to selectivity and capacity, a number of other considerations need to be taken into account when designing an ion exchange column. As noted previously, smaller particles give improved mass transfer rates. Therefore, using the smallest practical particle size is to the designer's advantage. However, as average particle size decreases, flow resistance (pressure drop) through a packed bed of particles increases sharply.

The commercial development effort behind the IE-911 exchanger focused on achieving the optimal mass transfer efficiency from the optimal particle size. The 30x60 mesh particle size for the standard IE-911 exchanger represents a balance of ion exchange kinetics and design engineering requirements. Although the 30x60 mesh product represents an engineering optimum, other size ranges, both larger and smaller (for example, 20x50 or 60x100), could be made available on request.

Early in the development of the CST engineered form, researchers recognized that an understanding of the ion exchange kinetics would be useful to optimize the design of the ion exchange plant. An effort was undertaken, under the direction of R.G. Anthony (coinventor of CST) at the Kinetics, Catalysis and Chemical Reaction Engineering Laboratory of Texas A&M University, to model Cs ion exchange. This work was done in parallel with the commercial development of IE-911 exchanger. The model developed by R.G. Anthony, and others, has clearly demonstrated its utility as a predictive tool. It can be used to predict the effects of variations in waste chemistry and column ion exchange design parameters. Readers are referred to published papers by R.G. Anthony for further details (5).

Physical Properties of IE-910 and IE-911 Ion Exchangers

UOP IONSIV IE-95 and IE-96 and TIE-96 ion exchangers are zeolite-based ion exchangers developed, and used successfully, for earlier wastewater and ground water remediation. The IE-911 exchanger was engineered to match the strength and attrition characteristics of these earlier successful products.

Both the IE-910 CST and the inert binder used to make the IE-911 exchanger are highly resistant to chemical attack. Thus, the exchanger can maintain its physical integrity under virtually all pH conditions. Samples of IE-911 have been tested for seven days in contact with extremely high pH DSSF-5 simulant to determine whether there might be a loss of strength typical of the Hanford type wastes, might occur during exposure (3). Strength after exposure was, by the standard UOP ion exchange attrition test, comparable to material not exposed to DSSF-5. The results were within the acceptable range for successful commercial nuclear ion exchange materials such as IONSIV IE-95 exchanger. Additional long-term tests that include the effects of time and temperature are currently underway.

The Cs loaded CST powder could not be significantly eluted using 3M nitric or 3M formic acids. Neither could Cs-CST be eluted with 8M NH4NO3, 2M Ca(NO3)2, 1.5M Pb(NO3)2 or 2M Mg(NO3)2 (4). This experience correlates with the extremely high distribution coefficient observed at low pH. This property, combined with its high capacity, makes IE-910 powder and IE-911 exchanger an ideal, single-use materials. Their inorganic composition makes them compatible with direct vitrification.

The CST has also been evaluated to determine the effects of thermal exposure. Dry heating of Cs-loaded CST to several hundred degrees should not cause any loss in the retention of Cs. The CST shows no loss in performance or structural integrity during exposure to ambient temperatures in alkaline solution. However, temperatures should be maintained at less than 60C when CST is exposed to concentrated acids or extremely alkaline solutions for extended times. Long-term exposure to concentrated nitric acid (> 6M) results in leaching of some Ti (4).

The IE-910 and IE-911 materials have rigid inorganic structures that resist significant swelling or shrinkage with changes in temperature, pH, and ion exchange levels. This resistance to physical changes sets it apart from the organic ion exchange resins. Organic ion exchange materials shrink and swell with variations in pH, temperature, and salt concentration. These variations cause deterioration in organic IX beads, that leads to flow maldistribution, bed plugging, and other problems. The bulk density of IE-911 exchanger is approximately 1 g/cc (62.4 lb/ft3). Typical water contents determined by loss on ignition (LOI) are about 12wt-% for the IE-910 powder and about 20 w-t% for the IE-911 engineered-form exchanger.

Samples of IE-910 powder have also been exposed to high levels of radiation (109 Rads) in various solutions and showed no loss of structure or performance. This resistance to performance degradation was not seen for the organic ion exchange resins (6,7). Testing of the IE-911 engineered-form exchanger, is currently in progress.

SUMMARY

Commercial-grade CSTs in the powdered form (IONSIV IE-910) and the engineered form (IONSIV IE-911) have been developed and manufactured by UOP. Both materials have been tested under a wide range of conditions in simulants and actual waste solutions. The IE-910 and IE-911 materials exhibit the high radionuclide selectivity and capacity and the chemical and radiation stability of the CST developed by Sandia and Texas A&M. A comparison of equilibrium cesium lambda values shows that both IONSIV IE-910 and IE-911 materials achieve high distribution coefficients compared with all of the other candidates evaluated. In addition, the IONSIV IE-

911 ion exchanger exhibits physical strength and attrition resistance equivalent to the best inorganic ion exchangers manufactured by UOP. The commercial CST products are not intended for regeneration. Thus, the design of a cost-effective, safe, and reliable cesium ion exchange process does not require the capital or operating expenses of a regeneration system. The commercial CST products are also compatible with the batch ion exchange processes or with backfill barrier applications not involving a column ion exchange process. The cesium radionuclide-exchanged CST appears to meet the requirements for long-term storage and for incorporation into final waste form, such as glass and concrete. IONSIV IE-910 and IE-911 ion exchangers provide high capacity, high selectivity, high decontamination factors, physical strength, and resistance to chemical and radiolytic attack. These properties make the IONSIV IE-910 and IE-911 products ideal for treating a wide range of wastes across the U.S. DOE complex and in commercial applications.

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Acknowledgements of Support:
U.S. Department of Energy at Sandia National Laboratories under contract DE-AC04-94AL85000 Westinghouse Hanford Company, TWRS Program U.S. Department of Energy, ESP Program

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40-3

REDOX POTENTIAL MONITORING IN GLASS: FERROUS-FERRIC RATIO DETERMINATION WITH ADAPTATION FOR IN-CELL USE

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ABSTRACT

The ferrous-ferric (Fe^{2+}/Fe^{3+}) ratio in glass may be determined using an ammonium acetate buffer/1,10-phenanthroline reagent scheme similar to Standard Method 3500-Fe D following dissolution of glass in a hydrofluoric acid (HF) and sulfuric acid (H_2SO_4) mixture. This method is within the 95% confidence level for t-test comparison with the method of Schreiber. This method has been adapted for remote use in shielded cells for determination of waste and vitrified waste redox potential.

INTRODUCTION

It has been shown that the control of the oxidation-reduction potential in ceramic glass melters, for high-level nuclear waste vitrification, effects glass product quality and may extend the melter's useful life (1-5). The ratio of ferrous ion (Fe^{2+}) to ferric ion (Fe^{3+}) is one indication of the overall redox conditions in the melter and is readily determined in the laboratory. A review of the current iron ratio methods suggests that a spectrophotometric method may be the one most readily adaptable to hot cell use (6).

A simple colorimetric method for ferrous-ferric (Fe^{2+}/Fe^{3+}) ratio determination developed by Jones (7) was suggested for monitoring the redox conditions in crucible- and pilot plant-vitrification. Developed for the commercial glass industry, this method, as modified by Schreiber (8) (Fig. 1), involves dissolution of a glass in an HF/H_2SO_4 mixture, reagent addition, and then pH adjustment with HCl or NH_3 to a pH of 3.3 to 3.5. Reduction of the ferric ion is accomplished with the addition of hydroquinone followed by a 30 minute reaction period. The resulting orange-red solution is analyzed spectrometrically at 510 nm.

Fig. 1

A West Valley Demonstration Project (WVDP) adaptation of this method was attempted In-cell and was not readily performed. It was difficult and time consuming to perform the pH adjustment step In-cell with dropwise addition of acid or base, particularly for large batches of samples that include blank and quality control samples.

A review of the literature for a more adaptable method that would produce acceptable results suggested that Standard Method 3500-Fe D, Phenanthroline Method (9), would be a useful starting point. This is because of the minimum number of reagents required for complete analysis and the acceptable sensitivity of the method in the range desired. The addition of excess buffer to establish an acceptable solution pH instead of acid/base addition for pH adjustment is readily accomplished In-cell. The method calls for the use of an ammonium acetate/acetic acid buffer, which has a pH ~4.0.

Method development was performed in the laboratory before use in hot cells. A laboratory comparison was then conducted between the WVDP iron

ratio method and that of Schreiber. The spectrophotometers used were Perkin-Elmer Lambda 1A for bench-top analysis and a Hewlett Packard HP8452A equipped with Custom Sensors & Technology fiber optics and a dip probe for remote, In-cell analysis.

METHOD

Required reagents:

1. Ammonium acetate-acetic acid buffer: Dissolve 250 grams of ammonium acetate ($NH_4C_2H_3O_2$) in type II water. Add 700 ml of glacial acetic acid and stir.
2. 0.1% 1,10-phenanthroline solution: Dissolve 100 mg of 1,10-phenanthroline monohydrate in 100 ml of type II water. A couple of drops of HCl will hasten dissolution.
3. Hydroxylamine hydrochloride (NH_2OHHCl), reagent grade.
4. Concentrated hydrofluoric acid (HF).
5. Concentrated sulfuric acid (H_2SO_4).

The WVDP method is summarized in Fig. 2. A finely ground glass sample weighing 50 to 80 mg was placed in a 125 ml disposable beaker or 125 ml poly bottle. One milliliter of conc. HF was added and allowed to react. One milliliter of conc. H_2SO_4 was then added and allowed to react. Approx. 100 ml of type II water was added. Approximately 2 ml of this dissolution is transferred to a 60 ml analysis bottle that contains 15 ml of reagent water, 10 ml of ammonium acetate-acetic acid buffer and 5 ml of 0.1% 1,10-phenanthroline solution. The solution is diluted to 50 ml with Type II water.

Fig. 2

After 10 minutes, each solution is analyzed spectrophoto-metrically at 510 nm for Fe^{2+}. Approximately 0.1 to 0.2 grams of solid hydroxylamine hydrochloride (NH_2OHHCl) is added to each solution and heated to a gentle boiling in a microwave oven. After a 15 minute cooling period, each solution is again analyzed spectrophotometrically at 510 nm for total Fe^{2+}. Ferric ions are determined by the difference, and the Fe^{2+}/Fe^{3+} ratio is computed.

DISCUSSION

A single ground glass sample that is used as a reference material at WVDP was analyzed by the bench-top method and by the In-cell modification. Fifty Fe^{2+}/Fe^{3+} data points (Fig. 3) were collected: the bench-top data average was 0.343 0.048 and the In-cell average was 0.310 0.066.

Fig. 3

The bench-top method was then compared with the iron ratio method of Schreiber. Twenty data points for each method were collected (Fig. 2). The WVDP method average was 0.600 0.053 and the Schreiber method average was 0.627 0.079 (Fig. 4). A t-test comparison of the data points indicates that the means for the two methods are similar to a 95% confidence level.

Fig. 4

For the In-cell method, the analytical device is a fiber optic dipping probe connected to the spectrophotometer located outside the shielded cells. The dip probe was found to be much easier to use and maintain In-cell than a flow-cell. A "C"-shaped stainless steel fixture is used as a sight guide In-cell for the 50 ml dilution prior to analysis; this fixture is sized to fit 60 ml poly bottles. During method development, all reagents were dispensed In-cell. It is current WVDP practice to prepare dissolution and analysis bottles Ex-cell. The only reagents added In-cell are water and hydroxylamine hydrochloride. Pre-weighed

hydroxylamine hydrochloride is introduced in scintillation vials for each analysis. The WVDP is investigating forming pellets with this reagent for ease of handling.

SUMMARY

WVDP has developed a method based on Standard Method 3500-D which is readily adapted for In-cell use and found to produce comparable results to the widely cited method for Fe2+/Fe3+ determination. Results may be obtained in less than one hour from the time of sample grinding.

ACKNOWLEDGEMENT

The author would like to thank the members of the Analytical & Process Chemistry Staff for performing the analyses during method development.

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40-4

GLASS DURABILITY RESULTS FROM THE DWPF STARTUP TEST PROGRAM

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ABSTRACT

The Defense Waste Processing Facility (DWPF) at the Savannah River Site (SRS) produced seventy-one canistered waste forms containing simulated waste glass during the Startup Test Program. Glass from these canisters, along with glass samples taken during filling of the canisters, was characterized as part of a continuing effort to demonstrate compliance with the Waste Acceptance Product Specifications (WAPS).

Each of the glass samples taken during the Startup Test Program was subjected to the Product Consistency Test (PCT). The PCT is an ASTM crushed glass leach test that measures the releases of several elements over a period of seven days. The acceptance requirement related to this testing states that the PCT results of all of the glass samples analyzed

must be at least two standard deviations below the mean PCT results of the Environmental Assessment (EA) glass. The acceptance requirement was met for each of the three hundred and ninety-two glass samples characterized which demonstrated readiness of the DWPF to transition to Radioactive Operations.

INTRODUCTION

High-level radioactive waste at the Savannah River Site (SRS) will be immobilized in a durable borosilicate glass and poured into stainless steel canisters in the Defense Waste Processing Facility (DWPF). The canistered waste form will then be sent to a geologic repository for final disposal. The Department of Energy's Office of Civilian Radioactive Waste Management, which is responsible for the design and operation of the repository, has defined requirements which the canistered waste forms must meet to be acceptable for disposal in the repository. These requirements are the Waste Acceptance Product Specifications (WAPS).(1) To demonstrate compliance with the WAPS and to ensure that the DWPF was prepared for Radioactive Operations, the DWPF developed a Startup Test Program. As part of the Startup Test Program, the DWPF performed five non-radioactive melter runs which were designed to simulate the process expected during Radioactive Operations. Four of the simulated melter runs, the Qualification Runs, were used to obtain the data required for waste qualification.

A brief description of each of the five melter campaigns during the Startup Test Program is shown in Table I. The first run (FA-13) was not part of the Qualification Runs, but was used to flush the startup frit from the melter and achieve stable operation. In WP-14, the first campaign of the Qualification Runs, neodymium was added to the feed to study the mixing behavior of the melter. The WP-15 and WP-16 campaigns were used to simulate extreme changes in feed composition, and the WP-17 run was used to return to a baseline composition and to introduce noble metals to the DWPF.

Table I

The compositions used for the simulated feeds were based on the projections of various high-level wastes at SRS. While the high iron and high aluminum feeds were designed to test the extreme cases, the composite feed was based on an overall blend of the existing waste. To ensure that DWPF could consistently produce a durable waste form independent of the feed type, the Savannah River Technology Center (SRTC) designed an extensive program to sample the glass from the canistered waste forms.

GLASS SAMPLING

During filling of the majority of the canisters in the DWPF, samples were taken from the glass pour stream using a specially designed sampler. The samples were removed from the sampler by DWPF personnel and sent to the Savannah River Technology Center (SRTC) for characterization. Sixty-two of the seventy-one canisters were sampled by this method during filling. Samples of glass from the canisters were also characterized. The canisters were prepared for glass sampling by two methods: sectioning with a bandsaw and wall removal using arc-air cutting. Sectioning allowed three entire cross sections of the canister to be glass sampled. Wall removal only allowed the glass near the canister wall to be sampled. Generally, the first three canisters filled during a campaign (the period of the maximum change in composition) and the last three canisters filled (the period of closest approach to steady state), were cut at three

levels which separated the canister into four sections. For each of these canisters, samples were taken at each level. For each level, samples were taken as a function of radial position in the canister. A minimum of twelve samples were removed from each sectioned canister. A total of 295 samples were removed from the sectioned canisters.

A window at least twelve inches wide and twelve inches high was removed from the canister wall for all other canisters scheduled to be sampled. Wall removal allowed the canister to be available for glass sampling much quicker than by sectioning, but only one glass sample was taken from the wall removal canisters. Thirty-five samples were analyzed from the canisters with a wall removed.

Table II provides an overview of the number of canisters sampled, along with the total number of glass samples characterized during each melter run. Only six canisters were sampled during FA-13 because the data was not used for waste acceptance purposes. The sampling and characterization methods were tested during FA-13 to ensure readiness for the Qualification Runs. The last three canisters of WP-17 were not sectioned since the material at steady state was the composite feed that was characterized during WP-14.

Table II

A total of 392 glass samples were removed from the canisters during the DWPF Startup Test Program. Each sample taken was characterized by determining its composition and its durability. The results discussed below concentrate on the durability testing.

GLASS TESTING AND RESULTS

The durability of the glass was assessed by leaching it according to the Product Consistency Test (PCT) protocol.(2) The PCT is a crushed glass leach test that measures the releases of boron, sodium and lithium from the glass at 90C in ASTM Type I water over a period of 7 days.

The PCT was performed on all of the glass samples taken during each melter campaign. The tests were performed in triplicate and each test included the appropriate blanks and standards. The results of the standards indicated that the tests were within an acceptable range.

For each sample, the measured PCT responses for boron, sodium and lithium were normalized by the weight fraction of that element present in the glass. The normalized releases were compared to the Environmental Assessment (EA) glass (3) to determine the effectiveness of the Glass Product Control Program (GPCP).(4) The acceptance criterion of the GPCP states that the glass produced by the DWPF must have PCT results which are at least two standard deviations better than the PCT results of the EA glass.(5)

The average normalized boron release and the associated standard deviation for each campaign can be found in Table III along with the boron release of the EA glass. Tables IV and V show the same information for sodium and lithium, respectively. These results show that the glass produced by DWPF during the Startup Test Program was significantly more durable than the EA glass. Therefore, the acceptance criterion of the GPCP was met.

Table III

Table IV

Table V

The composite feed contained higher concentrations of silicon and aluminum than the high iron feed, WP-15. Since silicon and aluminum have a positive effect on the glass durability, it was not unexpected that the

glass from the composite feed was more durable than the glass from the WP-15 high iron feed. Figure 1 graphically depicts the changes in the boron release during WP-15 and WP-16. The results for each canister were averaged and were plotted in the order of canister filling. Although there is some scatter, the general trends can be observed. Even at the maximum value, the boron release for the DWPF glass is significantly below the EA glass.

Fig. 1

CONCLUSIONS

The characterization of the glass taken from the canisters produced during the DWPF Startup Test Program showed that the DWPF can produce a durable glass even with extreme changes in feed types. Nearly four hundred glass samples were removed from the canisters and subjected to the Product Consistency Test to measure the chemical durability of the glass. The PCT results indicated that all of the samples tested met the acceptance requirement when compared to the PCT results for the EA glass. The DWPF was therefore able to demonstrate readiness of the DWPF to transition to radioactive operations.

ACKNOWLEDGMENTS

This paper was prepared in connection with work performed under Contract No. DE-AC09-89SR18035 with the U.S. Department of Energy.

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VITRIFICATION OF HIGH LEVEL NUCLEAR WASTE INSIDE AMBIENT TEMPERATURE DISPOSAL CONTAINERS USING INDUCTIVE HEATING: THE SMILE SYSTEM*

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ABSTRACT

A new approach, termed SMILE (Small Module Inductively Loaded Energy), for the vitrification of high level nuclear wastes (HLW) is described. Present vitrification systems liquefy the HLW solids and associated frit material in large high temperature melters. The molten mix is then poured into small (~1 m3) disposal canisters, where it solidifies and cools. SMILE eliminates the separate, large high temperature melter. Instead, the HLW solids and frit melt inside the final disposal containers, using

inductive heating. The contents then solidify and cool in place. The SMILE modules and the inductive heating process are designed so that the outer stainless can of the module remains at near ambient temperature during the process cycle.

Module dimensions are similar to those of present disposal containers. The can is thermally insulated from the high temperature inner container by a thin layer of refractory alumina firebricks. The inner container is a graphite crucible lined with a dense alumina refractory that holds the HLW and frit materials. After the SMILE module is loaded with a slurry of HLW and frit solids, an external multi-turn coil is energized with 30-cycle AC current. The enclosing external coil is the primary of a power transformer, with the graphite crucible acting as a single turn "secondary." The induced current in the "secondary" heats the graphite, which in turn heats the HLW and frit materials. The first stage of the heating process is carried out at an intermediate temperature to drive off remnant liquid water and water of hydration, which takes about 1 day. The small fill/vent tube to the module is then sealed off and the interior temperature raised to the vitrification range, i.e., ~1200C. Material volatized (e.g., small amounts of cesium) during the liquefaction process is retained by an internal lower temperature "cold trap" inside the module. Liquefaction is complete after approximately 1 day. The inductive heating then ceases and the module slowly loses heat to the environment, allowing the molten material to solidify and cool down to ambient temperature.

The process cycle requires approximately one week, with most (60%) of the time used for cool-down. During the process cycle, the outer steel can is maintained at near ambient temperature by air cooling to remove heat generated by parasitic induced currents and conductive heat transfer through the insulating firebrick layer. These parasitic effects increase the amount and cost of energy required to process module; however, the cost is still very low. For example, a SMILE module containing 1.25 m3 of vitrified HLW, (equivalent to the baseline HLW disposal module for the Hanford TWRS program) will require approximately 10 mWh. At a cost of 10 cents per kWh, this corresponds to only $1200, which is negligible compared to the repository fee of several hundred thousand dollars per module. Heat transfer analyses for the SMILE process are described, together with projected glass compositions based on Hanford HLW feeds. Using appropriate frit materials, the composition of SMILE-Hanford HLW glass can be made very similar to that projected for Savannah River waste. Because of its modularity and the elimination of long-term material problems, there do not appear to be major technical issues for the SMILE concept.

DESCRIPTION OF THE SMILE CONCEPT

The present vitrification approach is based on large, high cost, centralized high temperature glass vitrification facilities that melt the solid waste in large furnaces and then pour the molten glass into the final steel disposal containers. In SMILE, the waste solids would be vitrified in-situ inside closed, individual containers that serve as the final geologic disposal containers. The outer SMILE container is maintained at a relatively low temperature while its inner contents are heated to high temperature by an external low frequency (e.g., hertz) inductive heating coil. Thermal insulation between the inner contents and outer container minimizes heat leakage from the hot interior. The SMILE concept appears to offer substantial cost savings over the baseline

approach. It also appears to be more reliable and maintainable, is much less demanding on materials, and essentially eliminates the problems of radioactive emissions from hot melters. Besides its advantages of lower costs and simpler operation, SMILE has the unique capability that the vitrified waste inside disposal containers can be reheated to high temperature without degrading canister containment. If long-term radiation damage to the vitrified waste were to prove unacceptable, the inner contents of SMILE containers could be annealed when necessary.
Figure 1 shows the proposed SMILE concept. The empty container consists of a graphite cylinder with an inner alumina liner. The graphite is inductively heated to high temperature by an outer solenoidal coil that is connected to an external power source. The solenoidal coil operates at a low frequency (e.g., 30 hertz), and acts as the primary of a transformer, with the secondary being the graphite cylinder. The graphite cylinder is enclosed in a stainless steel jacket (not shown) with a layer of thermally insulating refractory ceramic, e.g., alumina fire brick (also not shown), between them. The graphite can operate at high temperature, e.g., 1200C or higher, while the enclosing steel jacket is maintained at a relatively low temperature [200C, for example] by auxiliary air cooling. The stainless steel jacket completely enclosed the graphite cylinder and its contents except at the top of the container, where the fill/vent tube is initially open.
Fig. 1
The SMILE version in Fig. 1 uses graphite as the conducting scepter that is heated by the inductive heating coil. Other versions are possible. For example, the graphite cylinder could be replaced by an iron container, which would couple even better to the inductive coil, making the heating process more efficient. The iron scepter version of SMILE would have a somewhat lower temperature capability but would still be high enough (i.e., ~1150C) to vitrify HLW into borosilicate glass. It also is possible to have multiple tubes or cylinders (graphite or steel) at several points in the HLW solids to enable faster and more uniform heating. Here, however, we only analyze SMILE designs with a single graphite scepter that contains all of the HLW solids to be vitrified.
The external fill/vent tube at the top of the container allows the waste slurry (solids plus water) to be introduced and also allows the gases from the drying phase (e.g., steam) to leave. The central internal porous ceramic vent tube collects the gases generated inside the annular porous bed which then allows them to flow to the external vent. The steam travels at most about 12 inches (average about 6 inches) through the porous solid zone to reach the central vent tube, compared to a distance of 15 feet if there were no tube.
Figure 1A illustrates the drying operation. The container is filled with a wet slurry of HLW solids and frit material. Excess water is pumped out through the central vent and external fill/vent tube, leaving behind a settled bed of wet solids. The contents are then sequentially inductively heated to a sufficiently high final temperature (e.g., ~800C) to first drive off all liquid water, then the water of hydration, and finally any residual decomposition gases. The remaining solids then contain only dry refractory waste oxides and frit.
The external vent/fill tube is then sealed off and the contents inductively heated to a higher temperature (i.e., ~1200C) to vitrify the HLW (Fig. 1B). No radioactive gases, e.g., volatized cesium, are released during vitrification, since the fill/vent tube has been sealed. This

minimizes radioactive release and contamination of the vitrification facility. After the HLW solids have been vitrified inside the sealed container, it cools to near ambient temperature, and is discharged to a temporary storage facility. Eventually it would be shipped to an off-site geologic repository.

If an additional protective barrier is desired, the container can be overpacked with an outer steel container. The process is illustrated in Fig. 2A. The open outer steel container would be positioned below the inner container and inductive heating coil during the drying and vitrification phase of the process.

Fig. 2

After vitrification is complete and the inner container has cooled to near ambient temperature, the outer container is raised to enclose the inner container (The outer container fits inside the heating coil). The outer container would then be lowered with the enclosed inner container, and removed through an exit transfer lock at the bottom of the pit. At the beginning of the sequence, both the inner and outer steel container come in through an entrance transfer lock at the bottom of the pit. The inner container enters first, is raised up by the hydraulic lift and then held in place by the center and adjustable side supports. The hydraulic lift is then lowered and the outer container brought in through the transfer lock. The treatment sequence is then initiated. The inner container is filled with wet HLW solids (#1); the water is removed by moderate heating (#2); the vent/fill tube of the inner container sealed (#3); and the HLW solids vitrified at high temperature (#4). In contrast to previous HLW vitrification processes, the SMILE process will not result in any significant release of radioactive volatiles. In the drying step (#2), the only volatiles are steam and decomposition gases (e.g., from residual nitrates) which are trapped externally; in the vitrification step (#4) all volatiles are fully retained inside the inner container.

After the heated, sealed inner container cools to near ambient temperature, the hydraulic lift raises the outer steel container around it (Step #5). The outer/inner container combination is then lowered to the bottom of the process pit and removed through the exit transfer lock. The open top of the outer steel container is then covered with a lid, and a welded seal made remotely, completely sealing off the vitrified waste inside a double barrier container (Step #6). The completed inner/outer container combination is shown in Fig. 2B. The finished container would be temporarily stored and then sent to an off-site geological repository. The projected time line for processing SMILE containers is given in Table I. A more detailed description of the SMILE concept is given in the BNL report, "SMILE - A New Approach for the Vitrification of High Level Wastes," by J. Powell, et al (1).

Table I

ANALYSIS OF THE SMILE CYCLE

Dimensions for a typical SMILE container are given below. These values are based on an initial study of the SMILE concept, and may change somewhat after further, more detailed studies. Outer radius of central vent tube = 5 cm; outer radius of HLW glass region = 32.5 cm; outer radius of alumina cylinder = 33.5 cm; outer radius of graphite cylinder = 41.0 cm; outer radius of thermal insulator = 46.0 cm; outer radius of stainless steel jacket = 47.5 cm; length of canister = 450 cm.

The drying of the wet HLW/frit solids involves a complex conductive/convective transient heat transfer process. Exact analysis of the process is beyond the scope of this paper. However, the transient behavior can be approximately determined from analytic solutions given by Schneider (2) in the Handbook of Heat Transfer. The following thermophysical properties are taken for the bed of HLW/frit solids: k = 1x10-2 watts/cmK, 70% solids volume fraction in bed, rCp = 2 Joules/cm3K [per cm3 of bed], with a corresponding thermal diffusivity of a = k/rCp = 5x10-3 cm2/sec. At the beginning of the drying phase, the temperature of the graphite alumina cylinder is raised from its original value, To, to TG and held constant. Heat is conducted/convected radially inwards, with steam and other gases being driven off through the central vent tube. The temperature ratio, TR, relating the time dependent (Q = time) temperature at the axis of the HLW/frit bed (i.e., at the central vent tube) to the original temperature To, is a function of the Fourier number, Fo,
Eq. 1
Taking TG = 1150 K(877C) and the Fourier number = 0.5, the analytic solution given by Schneider yields a value of TR = 0.90 (center temperature = 1065K). For the condition of RB = 32.5 cm and a = 5x10-3 cm2/sec, the time required is QDRY = 29.4 hours. At the end of the drying phase, the temperature of the coolest portion of the bed (i.e., at the central ceramic vent rube) is 792C, with the average bed temperature being ~800C. The temperature of the graphite/alumina cylinder is then increased to a higher constant value, TG*, so that the HLW/frit solids can be vitrified. Fixing a precise value for TG* will require experiments but it is expected that TG* will be on the order of 1200C (1473K). Taking temperature ratio, TR*, for the vitrification phase is 0.85, the corresponding Fourier number is Fo* = 0.4. The time required for vitrification is QVIT = 23.6 hours. The bed temperature at the ceramic vent tube at the conclusion of the vitrification phase is T* (0,QVIT) = 1150C, based on values of TG = 1200C and To = 800C.

The total energy input to the bed for drying and vitrification supplied by the graphite/alumina heater is QTB = QDRY + QVIT = 1.16x109 Joules/meter, based on the change in enthalpies of the original components. Two additional energy inputs are supplied by the graphite/alumina heater: 1) energy to raise the heater from a cold state to its final operating temperature, and 2) energy lost to the cool outside steel container by conduction through the insulating alumina brick layer during the drying and vitrification phases. The first term equals 0.75x109 Joules/meter, the increase in enthalpy of the graphite cylinder and alumina liner. The second term is considerably greater, 2.42x109 Joules/meter. The temperature, TSS, of the outer steel container is maintained at a relatively low value, e.g., 500K (227C) by air cooling. The thermal conductivity of the insulation is taken as equal to that given by Marks (3) for #16 insulating brick refractory. The total energy input from the heater then equals QTOT = QDRY + QVIT + QHEATER + QINSUL = 4.33x109 Joules/meter. This corresponds to a heater energy input that is approximately 3 times greater than that required to dry and vitrify the HLW/frit solids themselves. The total energy input for a 5 meter long container is 2.16x1010 Joules or ~6000 kWh. At an electrical energy cost of 10 cents per kWh, this corresponds to a cost of ~$600 per container. The average electrical power input during the drying and vitrification phases is 113 kW(e).

There is an additional energy input and cooling load associated with parasitic I2R losses generated in the outer stainless steel container by the inductive heating coil. The SMILE container is equivalent to a single turn secondary of a transformer, with the primary being the inductive heating coil. The large size of the container, plus the low heating rate enables the use of low AC power frequencies, i.e., 30 hertz, in the inductive heating coil. The alternating magnetic field produced by the inductive heating coil will induce currents in the outer stainless steel container as well as in the graphite cylinder, however, and energy losses to the steel must be included as part of the overall energy losses. They will not affect the internal temperature distribution, but do affect the cooling load on the outer steel container. Analyses of the inductive heating process find that the parasitic currents in the steel container approximately double the required input power, from 113 kW(e) to a total of 230 kW(e).

COMPOSITION OF HLW GLASS PRODUCED BY SMILE

SMILE can be used with all processing alternatives, including enhanced sludge washing and acid dissolution (e.g., TRUEX). As an example, HLW from the TRUEX D process proposed for the Hanford TWRS program can be combined with suitable frit material to yield a final borosilicate glass composition that closely approximates Savannah River HLW glass.

Table II gives the weight percent of the various solid oxides (after heating) in the TRUEX D waste (4) and compares it to the weight percent distribution for Savannah River waste (5). There are substantial differences, and some components fall outside of the quoted possible range. However, by addition of appropriate frit material, the combination of Hanford TRUEX D waste and frit can closely match the composition of Savannah River waste glass. Table III shows the resultant composition produced by adding frit in the ratio of 3 parts frit by weight to 1 part of TRUEX D waste. The resultant composition matches that of the Savannah River glass for all components except those marked with an asterisk. For those that are different, the discrepancies are relatively small, with the principal differences being a higher amount of Al2O3 (6.9 vs. 2.8 wt.%) and a somewhat lower amount of Fe2O3 (10.1 vs. 14.5 wt.%). These differences are not expected to change glass properties significantly, however. These relatively small differences could be further reduced by minor adjustments in the concentrations of the other components.

Table II

Table III

SMILE DEVELOPMENT

There do not appear to be any major technical feasibility issues for SMILE. The inductive heating coil is low tech and thermal insulation techniques are well established. The handling and lifting equipment appears straightforward. Material behavior during the vitrification phase should be confirmed; however, it appears molten glass can be contained without problems in both alumina and graphite. Because the SMILE concept involves small modules, and it does not require the high temperature components to operate for long periods, it should be possible to readily demonstrate the concept at full scale, and show that containers can be reliably produced. Technology modifications can be made quickly and easily, since the time to process a container is only a few days. This eliminates waiting for many months to detect problems, and then waiting for many more months to see if the required modifications are successful.

Borosilicate glass can be produced in alumina, steel, or graphite containers. Issues relating to heat transfer and energy cost do not appear to be significant. The issue of how much gas is released inside a SMILE container during the vitrification phase, and whether it would result in an objectionable pressure increase, can be resolved at an early stage by laboratory tests on simulated wastes. As discussed earlier, if the release amount is considered excessive, the container could be vented during vitrification with an attached cold trap to catch whatever cesium would volatize.

There is a substantial information base on the properties of the various materials that would be used for SMILE, and the thermal and electrical designs of the SMILE container and process facility appear to be straightforward. However, SMILE will require the normal engineering development associated with any new process, which involves going from bench scale to pilot plant to full scale.

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HANFORD HLW IMMOBILIZATION IN SYNROC

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ABSTRACT

Current strategies for the remediation of Hanford tank wastes envisage four HLW product streams: Cs, Tc, Sr/TRU and insoluble residues. All of the Cs and Tc products can be immobilized in 27 metric tonnes of Synroc that could be produced in a plant of 2 MT p.a. capacity over the time scales envisaged for Hanford remediation. There is a possibility that further segregation of the major heat generating HLW can be achieved without significant increase in the amount of Synroc by including the Sr/TRU stream with Cs and Tc. The Synroc would require extended intermediate storage prior to geological disposal but the benefits of low heat generation and the absence of hard g from 137Cs in the major HLW stream would outweigh the costs. Results are also presented of a Synroc/glass composite with high waste loadings produced by melting. Crystalline zirconolite and perovskite, which have very stable analogue minerals in nature are the hosts for TRU in this composite wasteform.

INTRODUCTION

About 240,000 MT of process chemicals contaminated with about 250MCi of radioactive elements are currently stored at Hanford in 177 steel tanks (1). The wastes are dominated by sodium nitrate and nitrites. Recent

descriptions of the waste compositions have been given (1) but considerable uncertainty exists because the wastes are inhomogeneous, resulting from various reprocessing and subsequent chemical processing campaigns and numerous intertank transfers. Current strategies (2) for the remediation of Hanford Tank Wastes envisage prior separation of most of the Cs, Tc and Sr/TRU from the existing liquid components of the waste, including sludge washings, which are then destined for solidification as low-activity waste. The insoluble components of the sludges together with the three separated components, will thus form four separate HLW products.

Borosilicate glass is the only currently accepted high-level waste form in the US. At glass waste loadings of 25-45 wt.% as many as 30,000 standard 0.6m diameter by 3m long waste canisters will require geologic disposal. Significant incentives exist to minimize the volume of conditioned HLW because repository space is limited. High waste loadings in glass require melting temperatures significantly higher than the 1150oC currently envisaged in conventional ceramic glass melters. Waste loading in glass is subject to two constraints: product quality and ease of processing (3). High temperatures in glass making may lead to extensive loss of Cs and Tc through volatilization. Crystallization of refractory crystalline phases (4) at high waste loadings is detrimental to the operation of Joule melters, but does not affect melters with short residence time and with vigorous mixing such as high frequency induction heated cold-crucible melters.

The separation of the HLW fraction of Hanford wastes into four distinct products provides an opportunity to explore possible synergies between Synroc, a crystalline ceramic wasteform underdevelopment at ANSTO, and borosilicate glass in the context of Hanford Waste remediation. Two aspects will be discussed in this paper:

The immobilization of Cs and Tc in Synroc by ceramic process technologies to provide a stable wasteform for Cs135 and Tc99, both of which have very long half-lives and significant solubility and mobility in most geologic media.

The results of early studies in the development of a Synroc/glass composite waste form that can be produced by melting at temperatures of about 1400oC.

SYNROC

Synroc-C, a formulation developed for the immobilization of HLW from the reprocessing of commercial LWR spent fuel, consists mainly of zirconolite, $CaZrTi_2O_7$, barium hollandite, $Ba(Al,Ti)_2Ti_6O_{16}$, perovskite, $CaTiO_3$, and excess titanium oxide. A combination of the first three phases has the capacity to accept, in solid solution, most of the elements present in HLW. Under the redox conditions chosen for Synroc fabrication, a number of waste elements such as Ru, Rh, Pd, and Tc, are reduced to the metallic state and form alloys that are microencapsulated within the titanate phases. The alloys also contain Mo, Ni, Te and some Fe. Hollandite is the host for Cs, Ba and Rb; perovskite is the major host for Sr; and zirconolite and perovskite are the hosts for rare earths and actinides.

Titanium metal powder is added to calcined Synroc at the 2 wt.% level for redox control during consolidation by hot-pressing. This is an important aspect of Synroc chemistry. An excess of reduced rutile provides Synroc with the capability to maintain the desired phase assemblage (by changes in the relative abundance of the phases) even if unexpected fluctuations

occur in the HLW stream composition. This process flexibility is evidenced by the ability to use the same Synroc precursor composition for waste loadings in the range 0-30 wt.% without deleterious effects on the chemical durability. The composition and mineralogy of Synroc-C are given in Table I.

Table I

The density of Synroc-C is 4.48g/cm3, the thermal conductivity is 2.5 W/m/K and it has an incongruent melting point at about 1350oC. These properties permit the consideration of higher waste loadings than in glass with the accompanying high temperature in the wasteform.

The major phases in Synroc are analogous to titanate minerals that have successfully immobilized (5) naturally occurring radioactivity (eg. U,Th) in a wide range of geochemical/geological settings. Natural samples of zirconolite have been studied (6) which have immobilized U, Th and their decay products for periods approaching one billion years.

The data base on the aqueous chemical durability of Synroc is extensive (7, 8). Protection is afforded by hydrated films of TiO_2 which form on Synroc and the leach rate decreases rapidly with time even with frequent replacement of leachant. At long dissolution times, the normalized leach rate of Cs and Tc at 90oC is less than 1x10-4g/m2d and for the actinide elements is less than 1x10-5g/m2d in water saturated with air. Matzke (9) has shown that the dissolution rate of Synroc at 150oC is less than 0.15 nm per day. Consequently, Synroc can be considered to be an important independent barrier to the return of radioactive waste elements to the biosphere. Radiation stability of Synroc and its phases has been discussed in the recent review by Ewing et al (10).

The key parameters for Synroc processing are effectively dictated by the need for a dense, fine grained product, ie > 98% of theoretical density, and control of redox potential at all key stages of the process to eliminate the possible losses of volatile species whilst maintaining chemical control to ensure appropriate partitioning of radwaste species into targeted phases. The redox potential is controlled during rotary calcination of the Synroc precursor (Table I) and HLW solutions at 700oC by a counter-current flow of 3.5% H_2/N_2 reducing gas that prevents the formation and subsequent loss of potential volatile species such as Cs, Tc and Ru. The calcine after blending with 2 wt.% titanium powder is transferred to stainless steel bellows containers, normally containing 35 kg of Synroc, before hot pressing at 1180oC. The use of the innovative bellows containers, Fig. 1, that are eventually stacked in waste canisters for storage, transport and disposal, avoids unnecessary generation of process wastes. The bellows containers also facilitate heat transfer from Synroc to the waste canister. The metallic top and bottom plates provide a short heat conduction path to the canister, and hence high waste loadings are possible for wastes to be stored for extended periods in engineered facilities prior to disposal. Direct methods of quality control are possible through sampling of Synroc in the bellows on a statistical basis. The process technology of Synroc production by ceramic hot-pressing has been demonstrated on a non-radioactive basis at 10kg/hr scale (8) and the operating experience has provided the basis for a conceptual design of a radioactive plant.

Fig. 1

Whilst the Synroc-C formulation was specifically designed for HLW from commercial reprocessing of LWR fuels, it is flexible and has useful properties for waste partitioning strategies in commercial fuel

reprocessing (11). Applications of zirconolite-rich Synroc-C for excess plutonium disposition have been discussed recently (12, 13). Synroc-D, developed by Ringwood et al (14) for the immobilization of defence wastes at Savannah River, demonstrated the overall flexibility of the Synroc concept in adapting to different wastes. High concentrations of inert contaminants such as Na2O, SiO2 and iron oxides, coupled with relatively low levels of actinides and fission products, required a compromise for high waste loadings to be achieved. In Synroc-D, hollandite was no longer thermodynamically compatible with silicate phases. Nepheline ($NaAlSiO_4$) was introduced as the host phase for Cs, Rb and Ba. Compatible spinel phases are used in Synroc-D for the excess inert components in defence wastes to yield acceptable waste loadings with a leach resistance superior to defence waste glasses. The Synroc-D strategy employing ceramic processing is not capable at present of reaching the high production rates required for immobilization of the sodium-rich Hanford wastes.

HANFORD Cs AND Tc WASTES

About 2000kg of Cs and 1500kg of Tc are likely to be extracted from the Hanford Tank wastes. The 137Cs constitutes about 26% of the total Cs and the total decay heat from Cs is 220kW. The maximum Cs loading in Synroc can be constrained by solubility limits or by the permissible heat loading.

The maximum heat loading at the La Hague, France, vitrification plant (15) is 3kW for a container holding 400kg of glass. This is a much higher thermal loading than the Hanford specification of about 0.4kW for a similar weight of glass. The separation of 137Cs and 90Sr followed by conditioning in stable matrices (eg. borosilicate glass or Synroc) and storage for about 150 years has been advocated by Northrup et al (16) to reduce the size of geological repositories for HLW disposal. An acceptable center line maximum temperature in Synroc bellows for intermediate storage loaded in a French canister is estimated (11) to be about 650oC corresponding to 8kW/400kg Synroc.

Recent studies by Hart et al (17) indicate that the solubility of Cs in Synroc is about 10 wt.%. On the basis of 7.5 wt.% Cs (1.95 wt.% 137Cs) in Synroc from Hanford Tanks, the heat loading on 400kg of Synroc is 3.3kW which is only 10% higher than the specification at La Hague. Consequently, the separated Cs from Hanford wastes could be immobilized in about 27 MT of Synroc. This could be achieved in a small Synroc plant of less than 2 MT p.a. within the envisaged time scales for HLW solidification at Hanford.

The Tc, immobilized together with the Cs, would be in the metallic state. This Synroc with 7.5 wt.% Cs and 5.5 wt.% Tc, has the ability, if required, to immobilize significant quantities of the separated Sr/TRU product at Hanford. On a series of tests on Cs/Sr added in molar ratios of 2:1 to the standard Synroc-C precursor, Hart et al (17) observed the correct phase assemblage existed in Synroc containing 10 wt.% Cs2O and 3.4 wt.% SrO. Since this Synroc also contains perovskite and zirconolite there is the additional capacity to immobilize TRU as well. The Sr will increase the heat loading on Synroc and may require some increase in the total weight of Synroc over that for Cs and Tc immobilization. Nevertheless, the bulk of Sr and TRU and the remaining Tc in the Hanford HLW are expected to remain in the insoluble residues stream. It is premature to speculate further at this time of the amount of Synroc required to immobilize the Cs, Tc and Sr/TRU streams at Hanford until

better estimates are available of the composition and quantity of the Sr/TRU extracted from liquid phases during Hanford Tank waste remediation.

The chemical durability of Synroc has been studied extensively in MCC-1 type tests with frequent exchange of leachant. The normalized differential leach rates for Cs and Tc decrease with leaching time. In de-ionized water at 90oC, the normalized leach rates for Tc after 90 days are < 1 x 10-4g/m2/d in air saturated water, and <1x10-6g/m2/d under anoxic conditions (17). The Cs leach rates in long term tests are also<1x10-4g/m2/d (8, 13).

SYNROC/GLASS COMPOSITE WASTEFORMS

Development of a Synroc/glass composite wasteform for the sodium-rich insoluble HLW sludges at Hanford has been initiated at ANSTO and preliminary results have been published (11, 18). Our strategy has been to formulate a wasteform that retains the extensively characterized refractory phases perovskite and zirconolite of Synroc-C as hosts for the long-lived TRU elements. Developments in cold-crucible melters (19) and the demonstration that Synroc can be produced through melting (20) in cold-crucible melters suggest that the technology for high-temperature and high throughput melting of crystalline/glass composites is feasible. The initial formulation was based on hypotheses about which elements could be removed from the solids in Hanford Double Shell Tanks (DST) by washing of sludges. The composition of the simulated DST average waste chosen is shown in Table II. The TRU were simulated by rare earth oxides.

Table II

To encourage the formation of zirconolite and perovskite, CaO and TiO_2 were added in either 1:1 or 1:2 molar ratios and SiO_2 was added to increase the durability of aluminosilicate phases. Melting was carried out in Pt-5%Au or Al_2O_3 crucibles under an argon atmosphere at 1400-1500oC. The melts were cooled at 2, 5 or 20oC/min. and no significant differences were observed in their microstructures.

The microstructure of a Synroc/glass composite containing 70 wt.% of the simulated DST waste in Table II and $TiO_2/CaO/SiO_2$ of 11.9, 8.3 and 9.8 wt.%, respectively is shown in Fig. 2. The phase assemblage consists mainly of zirconolite, perovskite, spinel and glass in the ratio of 25:10:10:54. The rare earth simulants of TRU were found to be distributed between zirconolite and perovskite (18). The durability of this composite wasteform was good as indicated by the following 7-day 90oC PCT test leach rates (g/m2) : Na (0.3), K(0.3), Si(0.1). For the remaining elements the leach rates were all < 0.07g/m2. Preliminary results for simulated sludge in wastes from NCAW and NCRW tanks suggest that acceptable formulations with waste loadings higher than 50 wt.% are achievable to meet the PCT criteria (18). Similar waste loadings have been reported by Hrma and Bailey (3) on glasses with other crystalline phases.

Fig. 2

DISCUSSION AND CONCLUSIONS

Technetium and 135Cs, because of their long half-lives and ability to migrate in the environment, contribute significantly to the long-term risk associated with radioactive waste disposal. Synroc can accept waste loadings of 7.5 wt.% Cs and 5.5 wt.% Tc. About 27 MT of Synroc is sufficient to immobilize all of the Cs and Tc product streams from Hanford originating from current strategies for remediation of the tank wastes. The high 137Cs content in the Synroc will require extended

intermediate storage prior to geological disposal. However, it may well be that two half-lives of 137Cs may be required before a geological repository is available, particularly if the bulk of the remaining conditioned HLW is disposed first. A very small Synroc process plant of 2 MT p.a. would be required to immobilize the Cs and Tc over the envisaged time scales of Hanford remediation.

There are advantages in also immobilizing in Synroc the Sr/TRU fraction from Hanford waste pretreatment. The amount of extra Synroc production capacity over that required to immobilize Tc and Cs is expected to be small. Such a strategy would reduce the heat loading in the remaining fraction of HLW that could be conditioned in borosilicate glass. There would be accompanying benefits in the ease of handling the HLW glass because of the elimination of the hard g from 137Cs decay and the reduction of heat loading enables more glass blocks to be disposed early in the first geological repository in the U.S. The cost of the interim store for heat decay of the much smaller amount of Synroc would be insignificant compared with savings in the handling and disposal costs of glass.

The Synroc process technology, based on conventional ceramic processing, combined with the innovative use of bellows containers during hot pressing, generates less secondary waste compared with large glass melters that have to be disposed.

Preliminary results also indicate that high waste loadings are possible in a Synroc/glass composite waste form produced by melting of the insoluble stream of HLW products from Hanford. The TRU component in these wastes is likely to be partitioned in the crystalline Synroc phases, zirconolite and perovskite, for which very stable analogue minerals are available in nature.

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Session 41 -- UTILITY WASTE CHARACTERIZATION/SCALING FACTOR
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41-1
USING THE 3R-STAT COMPUTER CODE TO DETERMINE THE QUANTITIES OF I-129 AND TC-99 IN LOW LEVEL WASTE
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ABSTRACT

This paper describes the use and application of the computer code, 3R-STAT to determine the quantities of I-129 and Tc-99 in LLW generated by operating nuclear plants. A topical report has been submitted to the U.S. NRC for review on July 20, 1993. After more than two years of review, the U.S. NRC has approved the use of 3R-STAT as an acceptably accurate method to determine the quantities of I-129 and Tc-99 generated by nuclear power plants. The NRC approved the use of 3R-STAT for LLW Disposal Site Developers, and Nuclear Utilities in their waste management programs. The paper discusses how the code is used in terms of the proper execution of the code. The paper also discusses the nature of the 3R-STAT code results and how these results are used to report the quantities of I-129 and Tc-99 shipped to disposal site operators.

INTRODUCTION

The computer code, 3R-STAT, was developed by Vance & Associates to determine, through computer modeling, the quantities of I-129 and Tc-99 in low level waste. The computer code was developed as an alternative to costly and difficult radiochemistry measurements on actual waste samples from nuclear power plants. A Topical Report was prepared and sent to the US NRC for review and approval in July of 1993 and it was approved in August of 1995. The NRC approved the use of the computer code by LLW site developers, nuclear utilities and LLW disposal facility operators.

DISCUSSION

The basic modeling developed for the 3R-STAT code was derived from the chemistry and physics of the release of the short-lived radionuclides from reactor fuel in a reactor relative to the release of the long-lived isotopes of I-129 and Tc-99. The input to the code are the measured reactor coolant concentrations of the eight (8) gamma radionuclides:

I-131	Cs-134
I-132	Cs-137
I-133	Co-60
I-134	
I-135	

From the relative concentrations of these radionuclides, the computer code determines the corresponding reactor fuel release conditions and based on these conditions determines the corresponding release rates of I-129 and Tc-99.

Typically, the reactor coolant isotopic data are analyzed on a fuel cycle basis to determine the average release rates of I-129 and Tc-99 over the fuel cycle. The computer code analyzes the isotopic data for each day that a sample of reactor coolant has been collected. The code determines the release rate of I-129 and Tc-99 for each day in units of mCi/MWD. The average release rate for a fuel cycle is simply the average of all of the days that samples were collected. For plants that sample reactor coolant daily, there could be 300 to 400 I-129 and Tc-99 release rates in the computed average for a fuel cycle. Table I displays the average fuel release rates and radionuclide conditions over a Beaver Valley fuel cycle. Figure 1 shows a graph generated by 3R-STAT displaying the activity release rate for I-129 (mCi/MWD).

Table I

Fig. 1

The code has been used to develop the inventory projections of I-129 and Tc-99 for new LLW disposal facilities in the following States:

- Nebraska
- Illinois
- Pennsylvania
- New York
- North Carolina
- Connecticut
- Texas

The inventory projections, were based on data from past fuel cycles (3 to 4 fuel cycles) from those plants planning to ship waste to the LLW facility. The unique future of 3R-STAT is that the code can be applied retroactively to fuel cycles long since completed, if the isotopic data can be retrieved. Each fuel cycle for each plant was analyzed using the computer code to yield average release rates of I-129 and Tc-99 for each fuel cycle and an average for each plant. The calculations for the inventory projections is summarized as follows:

Eq. 1

The NRC also approved the use of the computer code by nuclear utilities to determine the quantities of I-129 and Tc-99 contained in low level waste shipped from their power plants to an operating disposal facility. The method approved by the NRC for utilities would involve continuing to include MDA values on shipping manifests for I-129 and Tc-99. Then on an annual or fuel cycle basis, the utility would report the values produced by 3R-STAT for the reporting period. The calculation is as follows:

Eq. 2

In the 3R-STAT Topical Report, it was proposed that a semi-annul or annual report be provided by the reactor generators to the site operator. This report would provide the total quantities of I-129 and Tc-99 shipped in LLW for the reporting period.

An illustration of the actual calculations for several of the Beaver Valley fuel cycles is given in Table II.

Table II

The NRC also approved the use the computer code by LLW disposal site operators for the purposes of verifying the results submitted by utilities. The site operator would receive the raw unedited isotopic data from a utility for a given fuel cycle. The site operator would analyze the data using 3R-STAT and compare the results against the results submitted by the utility.

41-2

IMPACTS OF 10CFR61 SCALING FACTORS FOR UTILITY LOW-LEVEL WASTE ON WASTE FACILITY PERFORMANCE ASSESSMENT

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ABSTRACT

Importance of accurately estimating the scaling factors for 10CFR61 radionuclides is investigated with respect to waste facility performance assessment. Scaling factors used in the analysis are estimated based on the shipping manifest information from Barnwell LLW site. Sensitivities of scaling factors of each radionuclides on the estimated dose are evaluated for a test case problem in a humid permeable site and key radionuclides and waste streams are identified. For 129I and 99Tc, the

use of scaling factors based on the lower limits of detection (LLD) values are compared with the case where more accurate scaling factors are used. Based on the test case analyses, the key radionuclides in LLW performance assessment among the 10CFR61 nuclides were found to be 14C, 129I, and 99Tc. Others that could produce non-negligible dose include 59Ni, 239Pu, 94Nb and 3H. It was found that the changes in scaling factors for key waste streams could potentially have a significant impact on performance assessment. Important waste streams were Class C and B dewatered resins, Class A unstabilized mixed DAW, Class C solid irradiated equipment, and Class C dewatered cartridge filters. It was also found that 129I and 99Tc will remain as important radionuclide in performance assessment even after more accurate non-LLD based scaling factors are used. This was due to the contributions from non-utility generators.

INTRODUCTION

The primary objective of low-level radioactive waste (LLW) characterization is to accurately assess radionuclide inventory in a waste disposal facility. In utility waste management, scaling factors are used to quantify the concentrations of difficult to measure (DTM) radionuclides within LLW. These scaling factors are known to have large variabilities depending upon the types of waste streams and radionuclides ranging several orders of magnitude. For I-129 and Tc-99, the lower limits of detection (LLD) are often reported by the laboratory analyzing the samples when these nuclides have not been detected or measured. When the LLD, rather than actual analyzed concentration, are used in the scaling factors the concentrations of these radionuclides in the wastes are over-estimated. This over-estimation of I-129 and Tc-99 inventory was found to range by factors of 10 to 10,000 (1,2). According to some of the recent performance assessments of LLW facilities, I-129 and Tc-99 are one of the major nuclides of concern. Therefore the overestimation of inventory of I-129 and Tc-99 may potentially have a significant implications in the licensing of any proposed LLW site.

The purpose of this work is first to identify the radionuclides and waste streams of importance in performance assessment and to quantitatively characterize the impact of variabilities of scaling factors of these radionuclides on the performance assessment. The implications of over-reporting I-129 and Tc-99 inventory is also investigated.

INVENTORY DISTRIBUTION OF 10 CFR 61 NUCLIDES IN LLW STREAMS

Performance assessment of a waste disposal facility is directed by the inventory of radionuclides stored in the facility. Accurately estimating the distribution of radionuclide inventory in major waste streams are thus important in performance assessment.

Table I shows this information for the 10CFR61 radionuclides associated with the use of scaling factors. This information was calculated using the radionuclide concentrations estimated by Chem-Nuclear Systems, Inc. based on shipping manifests collected at Barnwell, SC (3). Utility wastes account for the majority of inventory for all the radionuclides listed in Table I, except for Am-241 (only 6% from utilities). In fact, for many of these radionuclides, more than or close to 90% of the total activity disposed was estimated to be from the utility waste: 99.2% for Ni-59; 99.8% for Ni-63; 89.3% for Co-60; 91.3% for Sr-90; 99.4% for Nb-94; 95.8% for Pu-238; 94.3% for Pu-239; 95.1% for Pu-241; 100% for Cm-242; and 99.9% for Cm-243. For C-14, I-129, and Cs-137, about 80% of total activity disposed was estimated to be from utility generators. In the

case of Tc-99, utility waste accounted for about 66% of the total activity. The contributions from non-utility generators are separately listed in Table II.

Within the utility wastes, dewatered resins and irradiated equipment were the two major waste streams that make up the majority of inventory for the listed radionuclides except I-129 and Tc-99. The percentages of inventory accounted for by dewatered resins were: 60% for C-14; 20% for Tc-99; 37% for Ni-59; 34% for Ni-63; 78% for Sr-90; 88% for Nb-94; 78% for Cs-137; 73% for Pu-238; 80% for Pu-239; and 72% for Pu-241. The percentages of inventory accounted for by irradiated equipment were: 13% for C-14; 20% for Tc-99; 37% for Ni-59; 34% for Ni-63; 78% for Sr-90; 88% for Nb-94; 78% for Cs-137; 73% for Pu-238; 80% for Pu-239; and 72% for Pu-241.

For I-129, mixed DAW (~35%), dewatered cartridge filters (~16%), and incinerator ash (~20%) were reported to contain the majority of activity. The activity of Tc-99 was more evenly distributed within various waste streams (22% in DAW, 20% in dewatered resins, 13% in filters, 5% in incinerator ash, and 3% in equipment).

As shown in Table II, the contributions from non-utility generators to the total activity inventory were generally minor except Am-241 (94% from non-utility generators) and Tc-99 (34% from non-utility generators). Among the Tc-99 activity which was estimated to be from non-utility generators, industry produced most of this as DAW (26%) and as incinerator ash (6%). Am-241 in LLW was coming mostly from government and industry as sealed sources or dry solids.

Table I

Table II

TEST CASE PERFORMANCE ASSESSMENT FOR 10CFR61 NUCLIDES

To identify the radionuclides of importance, test case performance assessments are performed for the listed radionuclides. The test case was based on a hypothetical waste disposal facility, characterized as humid permeable site. Major input parameters used are listed in Table III. For the performance assessment, a computer model similar to SYSCPG (4) with modifications in the source inventory descriptions and the pathway/dose analysis was used (5). Results of test case analyses are given in Table IV. Many of the nuclides including Ni-63, Am-241, Co-60, Cs-137, Pu-238, Pu-241, Cm242, and Cm243 showed no or negligible contribution to the projected dose. Radionuclides found to be important include I-129, C-14, and Tc-99 followed by Ni-59 and Pu-239. For H-3, Nb-94, and Sr-90, non-negligible contributions were observed. For the three major nuclides in the assessment. i.e., I-129, C-14, and Tc-99 the peak dose was estimated to be close to 1 mrem/yr which occurred within 1,000 year time frame for the test case problem. Dose from Ni-59 and Pu-239 was estimated to be less than 0.02 mrem/yr with much longer time to the peak (6800 yr and 47600 yr, respectively).

Table III

Table IV

POTENTIAL IMPACTS OF SCALING FACTOR VARIABILITIES ON PERFORMANCE ASSESSMENT

To investigate the impacts of scaling factor changes of key radionuclides on performance assessment, sensitivity calculations were made for the waste streams of importance. The radionuclides considered for this sensitivity analysis include C-14, I-129, Tc-99, Ni-59, Pu-239, and Nb-94. Table V lists scaling factors for these radionuclides estimated in

each of the waste streams considered. These scaling factors were increased by the factor of 10 for the sensitivity analysis.

Table VI shows the results of the sensitivity analyses. Each data points in the table represent the ratio of dose estimate after the increase of scaling factor by a factor of 10 to the base case. The dose estimates were shown to increase by a factor of up to 3.5 for C-14 (Class C dewatered resins), up to 4.1 for I-129 (Class A unstabilized mixed DAW) and up to 2.4 for Tc-99 (Class C dewatered resins). For Ni-59, Pu-239, and Nb-94, the increase in the ratio was estimated to be higher: by a factor of 4.9 for the Class C solid equipment for Ni-59; by a factor of 4.9 for the Class C dewatered resins for Pu-239; and by a factor of 7.6 for the Class B dewatered resins for Nb-94. Almost all of the calculations show that the increase in the estimated peak dose was proportional to the inventory increase due to the increase in scaling factors. One exception observed was the case with Pu-239 where the solubility limit became limiting.

Table V

Table VI

IMPLICATIONS OF INVENTORY OVER-REPORTING FOR I-129 AND TC-99

It is known that the inventory of I-129 and Tc-99 in LLW is highly over-estimated due to the use of LLD in the calculation of scaling factors (1). To investigate this issue, calculations are made with different set of scaling factors for I-129 and Tc-99 based on recent findings (1,2,6,7). New scaling factors used for I-129/Cs-137 and Tc-99/Cs-137 were: (1) the existing scaling factors divided by a factor of 1000 (7), and (2) the existing scaling factors divided by a factor of 10 (1,2). The results of test case performance assessment after these changes were made are shown in Table VII. Results show that decreasing the current scaling factors by a factor of 10 was enough to significantly reduce the inventory of I-129 and Tc-99. With new scaling factors, the peak dose predicted was significantly reduced, by a factor of 3.7 and 5 for I-129 and 2.5 and 5 for Tc-99. Predicted peak dose levels, however, were still significant enough for I-129 and Tc-99 to remain important in LLW performance assessment. This was due to the contributions from non-utility generators. Non-utility generators contribute about 18% and 31% of the total activity inventory of I-129 and Tc-99, respectively, as listed in Table II.

Table VII

CONCLUSIONS/DISCUSSIONS

Based on test case analyses for a hypothetical LLW disposal facility, key radionuclides in LLW performance assessment among the 10CFR61 nuclides were found to be C-14, I-129, and Tc-99. Ni-59, Pu-239, Nb-94 and H-3 were found to be capable of producing non-negligible dose in the performance assessment. The test case analysis showed that the changes in scaling factors for key waste streams could potentially impact the performance assessment significantly. The waste streams that were identified important were Class C and Class B dewatered resins, Class A unstabilized mixed DAW, Class C solid irradiated equipment, and Class C dewatered cartridge filters. It was also found that I-129 and Tc-99 will remain important in LLW performance assessment even after the use of newly proposed, more realistic scaling factors due to the contributions from non-utility generators.

For more complete understanding of the importance of LLW characterization in performance assessment, comparative evaluations of all major parameters in performance assessment will be undertaken as the next step.

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41-3

CONSOLIDATION of WASTE CORRELATION FACTORS at DIABLO CANYON

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ABSTRACT

Diablo Canyon Power plant is a two unit PWR. Solid radwaste in the form of resin and cartridge filters is generated from both unit specific and common plant systems. The number of solid waste streams is potentially large due to the various plant systems. As part of the waste classification program samples from the various waste streams are sent to off site laboratories for radionuclide analysis on an annual basis. The annual cost of these services has ranged from $60,000 to $100,000. Statistical trending of eight years of laboratory data using vendor software has enabled the number of spent cartridge filter waste streams to be reduced from seven to two. This trending examined fission and corrosion product activity across the various systems, unit fuel cycles and the two different units.

Additional data trending enabled constant correlation factors for corrosion and transuranic fission products to be derived. This derivation was accomplished with a comparison across resin, cartridge filter and dry active waste streams over various systems, fuel cycles and the two different units. This has enabled a reduction in the number of nuclides requested in an off site analysis thus, reducing the annual cost of the service. As a result of the consolidation of waste streams and the truncation of the spectrum analysis, the cost of a single analysis has been reduced by 40% and the 1995 laboratory service budget has been reduced to $50,000.

INTRODUCTION

Diablo Canyon Power Plant is located on the Pacific coast near Avila Beach, California. Pacific Gas and Electric Company (PG&E) owns and operates the two unit Westinghouse 1100 MWe PWR units. The commercial operation dates for Units 1 and 2 were May 1985 and March 1986 respectively.

A waste classification program has been established at Diablo Canyon to comply with Federal regulations. Appendix F, 10 CFR 20 indicates the requirements for manifesting, certification, control and tracking of low level waste transferred to a land disposal site. Section 61.55 of 10CFR 61 requires waste to be classified. Wastes suitable for land disposal must fall into one of three categories; Class A, B, or C.

Wastes are determined to fall into one of the three classes based upon the concentration of particular radionuclides in the waste versus the limits set forth in Table 1 and 2 of Section 61.55.

Many of the nuclides listed in 10 CFR 61 are considered difficult to measure (DTM) since they emit no gamma radiation. Such nuclides must be measured using radio chemical separations. Since lab facilities at nuclear power plants are limited to gamma and tritium detection, such nuclides must be analyzed at off site vendor laboratories. These analysis are time consuming and expensive.

The NRC's Branch Technical Position on waste classification requires that the method of radionuclide content determination be accurate to within a factor of ten. The ratio of non gamma emitting nuclides to key gamma emitting nuclides from off site analysis enables correlation or scaling factors to be derived. These factors, derived from off site analysis, can be used in concert with subsequent on site gamma analysis to determine waste container activity. These factors can also be used to determine waste streams and trend waste stream spectrums over time. The Branch Technical Position allows the use of this technique as an accurate, cost effective and timely means of compliance.

The Branch Technical Position specifies that for Class B and C waste a confirmatory analysis should be performed annually to insure accuracy to within a factor of ten. Confirmatory analysis for Class A waste should be performed at least biennially.

PLANT WASTE STREAMS

Three types of low level radioactive waste are generated at Diablo Canyon. These are spent ion exchange media, spent cartridge filters and dry active waste (DAW). Spent filters and ion exchange media are produced by several different systems within the plant. Each unit has a spent fuel pool cleanup system, a chemical and volume control system (CVCS) and a boron recycle system. Within the spent fuel pool system is a refueling water cleanup system and within the CVCS is a letdown cleanup and a seal water injection/return cleanup system. In addition, a common radwaste treatment system processes liquid waste from both units.

Spent cartridge filters are produced from each system listed above. Since there are three unit specific systems and a common radwaste system at the plant a possible total of seven spent filter waste streams exist.

Spent ion exchange media is transferred from each system to one of two spent resin storage tanks. In general, high activity resin is stored in one tank and medium activity resin in the other. High activity resin consists of letdown mixed beds and spent fuel pool beds. Medium activity resin consists of all other CVCS beds, boron recycle evaporator feed beds, and all liquid radwaste beds. Since ion exchange media is blended

within these storage tanks there is a possible total of two such waste streams per year.

In practice a waste stream is any waste product or mixture of waste products where the DTM nuclide concentrations can be inferred by use of a single set of correlation or scaling factors (1). Intuitive grouping of waste streams would lead one to suspect that all cartridge filters would likely be similar in waste spectrum. Conversely, a cartridge filter spectrum should be very different from that of ion exchange media since a filter will not remove significant quantities of soluble nuclides.

Early in plant life, Diablo Canyon obtained annual cartridge filter samples for off site analysis from each system when waste was generated. The plan was to collect data for three or four years and plot the results. It was anticipated that the results would demonstrate that the cartridge filter spectrum from various systems would be the same. Although the magnitude was known to vary greatly between systems dependent on the proximity to the fuel, the relative abundance of nuclides were expected to be similar. If the spectrum were modeled as a color, say blue, one would expect a dark blue in letdown and refueling water filters with lessor shades of blue for boron recycle and liquid radwaste.

In 1990, a comparison of the cartridge filter correlation factors from all off site analysis was performed. The results of this study showed that both Unit 1 and 2 filter waste streams possessed similar characteristics. The filters removed the same nuclides regardless of the system. Cesium was not found in significant quantities on cartridge filters. Cobalt was shown to be of no influence in cartridge filter waste classification while non gamma emitters, particularly 14C, controlled classification. The study further found that all corrosion and fission product correlation factors were well within a factor of 10, except for 14C/60Co. It was concluded that further analysis to resolve the variability of 14C should be conducted.

EPRI reviewed the 14C content in cartridge filters at Diablo Canyon in 1991. This review concluded that the variability of 14C was partially influenced by the method of off site analysis. The vendor laboratory was not determining the gamma activity of the aliquot for destructive 14C analysis for normalization to the prime sample gamma spectral analysis. The off site lab specification was revised in 1992 to correct this problem and annual samples were obtained for three additional years.

In 1994, Diablo Canyon performed another comparison of cartridge filter sample results. This time the 10CFR61 Sample Analysis Program produced by D W James and Associates was used to trend and analyze the data.

SAMPLE ANALYSIS PROGRAM

The Sample Analysis Program enables rapid trending of numerous radionuclide sample results. Correlations or scaling factors can be determined for each sample and these factors can be compared for groups of samples as requested by the user. The program produces plots of historical scaling factors, see Fig. 1. The program calculates a median scaling factor and plots the factor of ten limits above and below the median for visual comparison. Any data points beyond these factor of ten limits are noted to be outliers. The user may select and remove outlier data points and produce another plot.

Fig. 1

In addition, the program calculates the dispersion of the data to quantify the variation. In this program the dispersion is defined as the

anti log of the standard deviation calculated from the log ratios (2). The lowest variability off site laboratories can achieve for the same sample is a factor of two. Any data plot from the Sample Analysis Program with a dispersion near two indicates that there is minimal variation in the data.

CARTRIDGE FILTER WASTE STREAM CONSOLIDATION

The basic strategy for consolidating several waste streams is to compare all historical scaling factor results as a data set. This comparison encompasses different fuel cycles, different systems and different reactors. If the dispersion for each scaling factor is near 2, then a common waste stream is indicated. If the dispersion approaches 5 for any single scaling factor, it suggests that more than one waste stream exists.

In order to consolidate the various filter waste streams at Diablo Canyon, fission product and corrosion product scaling factors for all historical data were plotted. All of the scaling factors met the general dispersion of two test except 14C /60Co, see Fig. 1, and 55Fe/60Co. Individual filter waste streams were removed from the grouping to determine if a single system was the cause of the increased dispersion. In addition, unit specific plots and annual plots to depict individual fuel cycles affects were generated.

The result of this comparison was that there are two cartridge filter waste streams at Diablo Canyon. Boron recycle filters are a separate waste stream because they have a higher 55Fe content, and a 14C content which is more than an order magnitude greater than all other filters, see Fig. 2. All other filters make up the other waste stream because a single set of scaling factors meet the general dispersion of 2 test.

Fig. 2

This evaluation reduced the number of filter waste streams requiring annual sampling from seven to two. With analysis costs ranging from $3,000 to $7,000 per sample depending on the turn around time, this reduction resulted in a substantial cost savings.

CONSTANT TRANSURANIC SCALING FACTORS

Based upon the success of the filter waste stream consolidation, an evaluation was conducted to determine if constant scaling factors could be determined from historical data for all waste streams, at Diablo Canyon. Industry data suggested that individual transuranic nuclides would correlate very well with 239Pu. Data plots from all waste streams were generated for the various transuranic nuclides versus 239Pu. These plots resulted in dispersions very near the factor of two test and a set of constant scaling factors.

These transuranic scaling factors were valid for any waste, any fuel cycle and either unit at Diablo Canyon. If the 239Pu/60Co ratio is determined for any future waste, the remaining transuranic nuclides can be determined by use of the derived scaling factors.

CONSTANT CORROSION PRODUCT SCALING FACTORS

Once again, industry data suggested that constant scaling factors could be derived for corrosion products 55Fe, 59Ni, 63Ni. Unlike transuranic nuclides, constant corrosion product scaling factors were likely to be waste stream specific.

Corrosion product scaling factors for the two filter waste streams, spent ion exchange media and DAW at Diablo Canyon were compared. The results were in keeping with industry data. The Nickel scaling factors had a small dispersion and valid constant values could have been used for all

waste streams. The 55Fe scaling factors had a larger variation between waste streams. As mentioned above the 55Fe concentration was much larger in Boron Recycle filters than other filters. The 55Fe/60Co scaling factor for Boron Recycle filters was nearly 7.7 while all other filters had a value of 2.7.

The 55Fe scaling factor variation resulted in two waste streams for ion exchange media. Medium activity resin and inorganic zeolite proved to have the same set of constant corrosion product scaling factors. High activity resin had nickel scaling factors slightly lower than those of medium resin. The dispersion of 55Fe in high activity resin was too large for a constant scaling factor to be derived.

Off site analysis for 55Fe in future spent fuel pool and letdown mixed beds will be obtained. The reason for the large dispersion is due to older spent fuel pool resin results. In 1995, spent fuel pool resin from both units was generated. The 55Fe results from these beds indicate a constant scaling factor for high activity resin will be obtained with a few more data points.

DAW corrosion product data also produced a constant set of scaling factors. These factors were valid for both radioactive trash and clean trash collected in the radiological controlled area. The DAW set of factors were closer to the cartridge filter values than those for resin.

AFFECTS OF FUEL DEFECTS

In late 1994, unit 2 began its seventh fuel cycle. Early on it was noted that a fuel defect was present in this cycle. In order to determine if the fuel defect had any affect on the constant scaling factors which were just derived, a letdown filter was removed and a full spectrum off site analysis performed.

The results of this analysis indicated no variation from the transuranic or corrosion product scaling factors previously derived. The refueling outage for unit 2 is scheduled for March 1996. A full spectrum analysis of a letdown filter and resin bed is planned to verify what affect, if any, the fuel defect has had. At this time, it is expected that this fuel defect will not result in altering the scaling factors derived from historical data.

COST REDUCTION

The economic benefit of deriving site specific constant scaling factors is reducing the number of nuclides analyzed in off site samples. The cost for 55Fe, 59Ni, and 63Ni analysis range from $800 to $1,300 per sample depending on the turn around time. The cost for transuranic nuclides excluding 239Pu range from $900 to $1,400. Thus, the use of constant scaling factors for corrosion products and transuranics saves $1,700 to $2,700 per sample.

Off site sample analysis can be reduced to a gamma isotopic, 14C, 89Sr, 90Sr, 239Pu, 99Tc and 129I for each waste stream. These results coupled with the constant scaling factors derived from historical data enable accurate waste classification to be conducted at a reduced cost.

The consolidation of cartridge filter waste streams and the use of constant scaling factors has resulted in significant cost savings at Diablo Canyon. In the past, $60,000 to $100,000 would be spent annually on off site lab analysis. In 1995, off site analysis for all waste streams generated was reduced to $35,000. This cost includes the analysis of filter samples to determine the affect of fuel defects on unit 2.

CONCLUSION

The comparison of historical scaling factor data can enable the derivation of valid constant values for corrosion products and transuranic nuclides. Such comparisons can also determine the number of waste streams a plant generates. The use of the Sample Analysis Program was critical to performing this comparison at Diablo Canyon.
The waste streams which resulted from this evaluation were based upon valid statistical evidence versus process knowledge. The waste stream results were not intuitively obvious. Significant cost savings can be achieved by accurately determining the number of waste streams. Additional savings can be achieved by deriving and using site specific constant scaling factors. These savings readily pay for the resources needed to perform the evaluation.
The Sample Analysis Program is also a great aid in evaluating waste classification affects of a fuel defect or other event which could shift the radionuclide waste spectrum. The results of new samples can quickly be compared with historical data.
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HUNGARIAN PROGRAM FOR INVENTORY DETERMINATION OF LLW/ILW OF PAKS NPP ORIGIN
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ABSTRACT
In the execution of disposal of low and intermediate level radioactive wastes, it is important to accurately evaluate the kind and quantity of each radionuclide in wastes. For such evaluation, correlation of non-gamma-emitting nuclides based on gamma-emitting nuclides is recommended and regarded as a practical method. This method necessitates a completion of a highly accurate and reliable non-destructive assay system of gamma-emitting nuclide for practical use. In 1992, in support of the new waste disposal program in Hungary, Paks NPP initiated a waste characterization program to determine the radiological properties of its radwastes.
A segmented gamma scanning system has been set up to measure the gamma-emitting nuclides in 200 liter low level drums following in-drum

compaction. In the framework of the program a radiochemical analysis sub-program was started to determine the long-lived non-gamma emitting radionuclides, mainly those listed in the US regulatory document (10CFR61). The radionuclides of interest have been 3H, 14C, 90Sr, 55Fe, 59Ni, 99Tc, 129I and TRUs. Sample preparation techniques and measurement methods have been selected and used. Newly developed or adopted methods have been tested on real liquid radwaste streams such as concentrates, ion-exchange resin and sludge. The measurements taken so far have revealed brand new information and data on radiological composition of waste of WWER-type reactors. In the next stage of the characterization program attempt will be made for providing correlation factors between the gamma and non-gamma emitting radionuclides in different waste streams. Short description of the methods and results on waste inventory are given highlighting the problem areas.

MONITORING PROGRAM

The goal of the monitoring program was to develop a non-destructive gamma-monitoring system that was capable of determining the inventory of gamma-emitting radionuclides in the dry active waste.

As a first step waste assay system was bought from the Canberra-Packard Company.

The description of the hardware and software of the monitor is as follows:

The software runs on an AT bus 386 computer system running under the MS-DOS operating system. The computer contains of a 2 Mb RAM memory, a 120 Mb Winchester hard disk drive, 3 serial and 1 parallel ports, VGA Color monitor, math coprocessor, and 1.44 Mb floppy disk drive.

The system utilizes a Canberra Series 35 PLUS multichannel analyzer with 4K channels of data memory.

The detector is a coaxial high purity germanium detector with a relative efficiency of 30% at 1.33 MeV, and energy resolution of 1.9 keV FWHM. The detector is mounted in a horizontal integral dipstick cryostat with a standard end cap and a 5 cm remote detector chamber. A Canberra Model 2002 preamplifier is mounted to the cryostat.

A lead collimator is used to focus the detector on defined segments of the drum. Collimator dimensions are 10 cm high by 20 cm wide by 20 cm deep. This provides an accurate 10 cm segment of the drum.

The drum assay mechanism is designed to handle containers up to 210 liter drums. These are rotated and moved vertically to accurately measure the gamma radiation emitted from the drum.

The detector collimator support frame is designed to hold the detector, shield, and collimator in the optimum orientation to measure the container being assayed. If a variety of container sizes are being measured the frame can be mounted on a trolley to optimize the detector to sample distance for each container size.

A transmission source shield and shutter is provided mounted on a pedestal and secured to the vertical drive unit to ensure correct alignment between the transmission source, drum, and the detector/collimator for accurate determination of the matrix density of the material.

The analysis software is a SPECTRAN-AT with modifications to optimize the matrix correction routines. Additional routines are provided to correct for matrix attenuation based on the transmission source intensity, and live time correction to correct for count rate losses with high activity containers.

The system is able to detect levels of 3 kBq of 60Co in a drum with a density of 0,5 g/cm3 with a drum assay time of 15 minutes. The minimum sensitivity for 137Cs 6 kBq.
The maximum positional error is less than +/- 25% for sources in different locations in the drum for containers with less than 0,3 g/cm3 density and gamma energies greater than 300 keV.
The maximum drum activity should be less than 500 MBq in the close geometry. If drums with high activities must be counted then the detector trolley should be moved away from the container to reduce sensitivity.
After assembling and testing the hardware and software, the detector was calibrated for efficiency with standard radionuclide sources by the Hungarian Bureau of Standards.
In order to evaluate various forms of miscellaneous waste packages regarding the deviation of density and activity, model waste drums were prepared and tests were performed.
Between 1989 and 1995, the radioactive wastes generated in the plant site of the Paks NPP amounts 2560 drums of 200 l compacted state having the distribution with regard to their content:

1901 drums of compacted waste material (74.2 %),
419 drums of non-compacted waste (16.4 %),
240 drums of solidified sludge (soaked up in prilled diatomaceous earth) (9.4 %).

The results of the waste assay determinations are given in Table I and II and in Fig. 1.
Table I
Table II
It seems quite clear that there are 3 dominating isotopes in the waste drums. 110mAg, 54Mn and 60Co isotopes are the most characteristic components of the waste, both for their frequency of occurrence and activity.
Fig. 1
RADIOCHEMICAL ANALYSIS PROGRAM
The objectives of the radiochemical analysis program were to develop the radioanalytical methods capable of determining the significant radionuclides (3H, 14C, 90Sr, 55Fe, 59Ni, 99Tc, 129I and TRUs) in the various waste forms destined for disposal and, provide the correlation factors between the gamma - and non-gamma-emitting radionuclides to enable to estimate the non-gamma radionuclides in wastes.
Radiochemical separation techniques developed or adapted are given below:
99Tc
Extraction chromatography is used to analyze evaporator concentrates for 99Tc. The technetium is oxidized to pertechnate using nitric acid + hydrogen peroxide and heating. This treatment also destroys most of the organic materials may be present in the concentrate. Amberlite XAD-7 resin is used to remove any organic material remaining after nitric acid and peroxide treatment. Then the pertechnate is separated from interfering elements using TEVA-Spec column. The 99Tc is measured by LSC technique.
Liquid-liquid extraction with dibenzo 18 crown 6 ether (DB18C6) in acetone-toluene mixture was also tried instead of using TEVA-Spec column. More reproducible results were obtained with DB18C6 than with TEVA-Spec. The yield in each method was tested using 99mTc tracer.
129I

The method is based on the separation of iodine from other activities by dissolution of elemental iodine into cold carbon tetrachloride. To assure chemical interchange with the iodine carrier an oxidation-reduction cycle is made. The iodide is oxidized to iodate with permanganate, reduced to iodide with bisulfite, and distilled over as free iodine in the presence of nitrite.

After washing the carbon tetrachloride with nitric acid, the iodine is reduced with bisulfite and back-extracted into water. Acidified silver nitrate solution is added to precipitate silver iodide. The chemical recovery is used as a measure of radiochemical recovery.

Two types of activity measurements were applied. Gamma counting of dried silver iodide precipitate with PGP-1500 detector and liquid scintillation counting (LSC). The AgI precipitate was dissolved by forming a chemical complex with potassium-cyanide. After one week waiting time the activity of the samples were determined by LSC.

A mass spectrometric system for measurement of 129I concentration by 129Xe ingrowth method was also developed. The system consists of a VG-5400 noble gas mass spectrometer and units for quantitative extraction of the dissolved gases from evaporator concentrate samples, purification of the extracted 129Xe after ingrowth period by means of a cryogenic cold trap filled with active charcoal. The mass spectrometer inlet system is computer controlled and capable of fully automated measurement of 129Xe samples. A 129Xe measurement takes about 30 minutes. Calibration of the measurement is done using defined aliquots of 126Xe standard. The 129Xe detection limit of the system is about 1*10-15 ccSTR. An intercalibration of the method with an LSC beta-counting was performed and within the errors of the single techniques no significant deviation was found between the two data sets.

55Fe and 63Ni

Radioactive 55Fe isotope solutions of 661 Bq and 1418 Bq 55Fe were prepared for calibration purposes. These solutions were added to the dry sample, dried with infra-red light, then the sample was powdered and homogenized. The powdered sample was placed into an aluminum sample container (diameter 10 mm, equipped with Mylar foil at the bottom) and measured with Si(Li) semiconductor X-ray spectrometer. The measuring time of the quantitative analysis was 5000 s.

Radiochemical Separation of Pu, Am and Sr Isotopes

The complete analytical procedure consists of the following steps: sample homogenization by filtration; take the residue by HCl; combine the aliquots measurement of the subsample. The further steps of the analysis:

addition of strontium carrier, and plutonium (236Pu) and americium (243Am) tracers

repeated evaporation with 65 % nitrite acid to destroy organic materials (especially complexing agents),

sample dissolution with dilute nitric acid,

separation procedure for plutonium

pre concentration of plutonium with iron hydroxide,

dissolution of the iron hydroxide precipitate with 1M nitric acid,

adjustion of the valency state of plutonium to tetravalent by consecutive reduction and partial oxidation,

chromatographic separation of plutonium on strong basic anion exchange resin from 8M nitric solution followed by the elution of interfering components and stripping of plutonium with a reducing agent (ammonium iodide in 9M hydrochloric acid solution)

evaporation and oxidization of the strip solution,
alpha source preparation by micro-precipitation technique with neodymium fluoride,
alpha spectrometry of the plutonium nuclides: 239,240Pu, 238Pu, 236Pu;
separation procedure for americium
preconcentration of americium by coprecipitation with calcium oxalate,
destruction of the oxalates with 65 % nitric acid,
chromatographic separation of americium on Tru.Spec column from 2M nitric acid solution followed by the elution of americium on Tru.Spec column from 2M nitric acid solution followed by the elution interfering components and strontium, stripping of americium with 4M HCl,
alpha source preparation by micro precipitation technique with neodyminium fluoride,
alpha source preparation by micro precipitation technique with neodymium fluoride,
alpha-spectrometry of the americium and curium isotopes: 241Am, 244Cm, 242Cm.
separation procedure for strontium
preconcentration of strontium by coprecipitation with calcium oxalate,
destruction of the oxalates with 65 % nitric acid,
chromatographic separation of strontium on Sr Spec. column (bis-tertier butyldicyclohexano-18-crown-6) from 3M nitric acid solution followed by the elution of interfering components and stripping of strontium with distilled water, precipitation of strontium oxalate,
determination of the chemical recovery by gravimetry,
checking of the purity of the sample by gamma spectrometry,
source preparation for liquid scintillation counting with Insa gel,
beta spectrometry with LSC: 90Sr, (89Sr).

Highest activity concentration and standard deviation of the radionuclides in the different waste types are given Fig. 2.

Fig. 2

CONCLUSIONS

Safety studies related to the disposal of low - and intermediate - level waste indicate that the long term risk is determined by the presence of longer-lived nuclides such as 14C, 59Ni, 63Ni, 99Tc, 129I and the transuranium elements.

As most of these nuclides are difficult to measure, the correlation between these critical nuclides and some other easily measurable key nuclides such as 60Co and 137Cs has been investigated, mainly for typical waste streams from nuclear power plants, and scaling factors have been proposed by several authors. However, the range of these factors is relatively large for some nuclides and some of therm are recognized to be plant-specific.

An automated gamma-scanning monitor has been developed to determine the inventory of gamma-emitting radionuclides in wastes. Experiences so far have indicated that the monitor can reliably determine the inventory of gamma-emitting radionuclides in the waste packages. Radiochemical methods have been developed to determine significant difficult-to-measure radionuclides.

One of the new methods developed is determination of 129I via 129Xe ingrowth by use of mass spectrometer.

In the next stage of the program the efforts will mainly be directed to the determination of the scaling factors.

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Session 42 -- INTEGRATION OF PUBLIC PARTICIPATION IN DECISION MAKING, BUDGETING & PLANNING AT DOE SITES
Co-chairs: Linda Ulland, Minnesota Pollution Control Agency;
Dennis Hurt, WIPP
42-1
REACHING OUT IN NEW WAYS: SUCCESSFUL COMMUNICATION STRATEGIES OF THE FEDERAL ADVISORY COMMITTEE ON EXTERNAL REGULATION OF DEPARTMENT OF ENERGY NUCLEAR SAFETY

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ABSTRACT
Secretary of Energy Hazel O'Leary appointed the Federal Advisory Committee on External Regulation of Department of Energy (DOE) Nuclear Safety (the Committee) in February 1995. The Committee's charter called for it to recommend whether and how existing and new DOE facilities should be externally regulated. The Committee's deliberations were to be conducted under the Federal Advisory Committee Act (FACA) in public through a constructive dialogue among DOE and external stakeholders. The 24 members of the Committee were drawn from a cross section of public, Federal, State, Tribal, industrial, and academic sectors, representing a diversity of expertise. The Committee's final report was submitted on January 19, 1996. The Committee's recommendations were submitted not only to the Secretary of Energy, but also to the White House Office of Management and Budget, and to the Council of Environmental Quality. The report recommended that DOE's facilities and operations be regulated by existing external entities, the specifics of which would be determined through Administration review or by Congressional legislation.
The Committee's charter and means of arriving at its recommendations presented an unprecedented and unique challenge -- the Secretary of Energy requested that the Committee provide its recommendations within only one year of its inception. The scope of the Committee's task had far-reaching impacts throughout the DOE complex. The regulatory issues involved were broad-ranging, complex, and often difficult to grasp; the affected constituency was large and varying; and the available resources were limited.
The Committee met this challenge through aggressively reaching out to the DOE community (workers, unions, contractors, managers), existing oversight organizations (OSHA, EPA, NRC, DNFSB), State and local governments, environmental advocacy groups, and local community and individual stakeholders. This paper identifies the challenges of the Committee's charter, how those challenges were met through an innovative

outreach program, and the key issues that the Committee gleaned from listening to diverse individuals from varying backgrounds, which led to its recommendations for external regulation.

WHY AN ADVISORY COMMITTEE?

Until recently, the Department of Energy (DOE) and its predecessors, the Atomic Energy Commission (AEC) and the Energy Research and Development Administration (ERDA) operated largely as self-regulatory entities in the area of safety at the Department's nuclear facilities. Currently the DOE complex is regulated under a mix of internal and external regulatory and advisory bodies that administer a maze of laws, regulations, directives, orders, and guidance. DOE's workforce suffers from intense frustration due to serious overlaps and gaps in regulatory requirements, and the failure by DOE to address hazards according to their relative risks, costs and benefits.

In response to the growing sentiment both in Congress and within the DOE and its surrounding communities that DOE "self-regulation" was inefficient and ineffective, Secretary of Energy Hazel R. O'Leary established the new Federal Advisory Committee on External Regulation of DOE Nuclear Safety (the Committee) in January 1995. The charter required the Committee to recommend whether and how existing and new DOE facilities should be externally regulated. The Committee was tasked to submit its recommendations to the Secretary of Energy, and to the White House Office of Management and Budget (OMB), and the Council on Environmental Quality (CEQ).

THE CHARTER

The charter, approved on January 25, 1995, tasked the Committee with the following:

> The Committee shall conduct an assessment of the technical, regulatory, institutional and resource issues related to the potential establishment of an external regulatory regime for nuclear safety with respect to Department of Energy facilities and operations.
>
> The assessment will include an evaluation of the historical separation between commercial and defense nuclear activities; the potential advantages and disadvantages of external safety regulation compared to the existing regulatory framework for environment, safety and health; and the concerns and criticisms that have been expressed about DOE's safety oversight by the Congress, the National Academy of Sciences, the Advisory Committee on Nuclear Facility Safety, the Defense Nuclear Facilities Safety Board, and others. Significant transition issues associated with any new regulatory regime will be identified, including a realistic appraisal of both Federal and non-Federal resource impacts.

Although the charter stated that "approximately twenty (20) Advisory Committee members shall be appointed by the Secretary of Energy for a term of two years", in March 1995 at the first meeting of the Committee, the Secretary requested a report be submitted by December 1995.

HISTORY OF SELF-REGULATION

Over the past half century, the DOE and its predecessor agencies have regulated themselves with regard to nuclear safety, as provided for under the Atomic Energy Act. The manner of oversight established in the Act reflected the agency's primary responsibilities for development and production of nuclear materials and weapons, and for many other classified national security activities appropriate in a defense environment. Production was of high priority during those times,

resulting in a legacy of safety and environmental problems at numerous DOE sites.

The nuclear mission of the Department has changed and much is now focused largely on the stewardship and dismantlement of nuclear weapons, cleanup of contaminated sites, safe decommissioning of facilities. In addition, in recent years, the DOE has become subject to an increasing number of external federal and state regulatory requirements and oversight, while at the same time internal requirements and oversight have intensified.

Activities in early 1994 demonstrated continuing national concern regarding nuclear safety at DOE facilities. In March 1994, Congress held hearings on the Federal Nuclear Facilities Licensing and Regulation Act, H.R. 3920. The Committee on Natural Resources, U. S. House of Representatives, held 2 days of hearings and heard from the NRC, the DNFSB, the EPA and other interested organizations and individuals on the need for regulating DOE nuclear facilities. At the hearings, the Department proposed that these ongoing concerns be addressed through a constructive dialogue within the DOE, other Federal agencies and external stakeholders via the formation of an independent Federal Advisory Committee.

The Secretary took action to establish an independent Committee, under the FACA, to examine whether (and how) new and existing DOE facilities and operations might be externally regulated to ensure nuclear safety. Secretary O'Leary approved the establishment of this new Federal advisory committee on July 28, 1994.

After a search and selection of a very broad and diverse group of distinguished individuals, Secretary O'Leary announced the membership of this Federal advisory committee on February 16, 1995. She named Dr. John F. Ahearne and Mr. Gerard F. Scannell as the Committee's co-chairs. Dr. Ahearne is currently a Lecturer in Public Policy at Duke University and is the Executive Director of Sigma XI Center (a Scientific Research Society). He also is past Chairman, Nuclear Regulatory Commission and the DOE Advisory Committee on Nuclear Facility Safety. Mr. Scannell is currently President of the National Safety Council. He is a former Department of Labor, Assistant Secretary for Occupational Safely and Health and former Vice President, Corporate Safety Affairs for the Johnson & Johnson Company.

The 24-member Committee represented a wide diversity of knowledge and expertise from the private sector as well as other Federal agencies. Thomas H. Isaacs, a senior career DOE official was named Executive Director of the Committee and a small senior federal staff was selected from DOE, the Environmental Protection Agency, and the Nuclear Regulatory Commission.

The Committee got to work immediately. Beginning with its first organizing meeting in Washington D.C., in March 1995, the Committee held eight public meetings in the next nine months. Organizing and planning the public meetings would have been enough of a challenge; however, since the Committee agreed to "get out of Washington" so they could hear directly from workers and the interested public, the majority of meetings were held at or near representative DOE sites.

THE CHALLENGE

The Committee's charter and the complex regulatory issues involved created unprecedented challenges for the Committee and its staff. Because of the unique make up of the Committee, there was no "text book" or

precedent to follow in planning and implementing an outreach strategy. Our challenges included the following:

COMPLEX ISSUES

The broad issue of "externally regulating DOE nuclear safety" is very complex. No group had ever evaluated the issue to the extent that the Committee was tasked. Although the charter mandated that the Committee assess the need to externally regulate "nuclear safety," the Committee found that it could not assess nuclear safety in a vacuum -- nuclear safety was inextricably tied to other safety and health issues, such as existing environmental compliance, worker safety and health, facility safety, and public health and safety. In addition, at the first Committee meeting Secretary O'Leary told the group to take a broad view of regulating DOE -- not to narrow its focus. Thus, the subjects of evaluation became more broad-ranging and even more complex. Most committees established under FACA have more narrow mandates directed at more specific problems.

BROAD CONSTITUENCY

The affected constituency was broad-ranging and all-encompassing. Given that the DOE complex involves such abroad-range of both internal and external stakeholders, the Committee was challenged with identifying and targeting a subset of representative stakeholders to participate in a constructive dialogue. This Committee did not have a "natural" constituency. Instead, the Committee staff was faced with the need to identify a sampling of all affected groups and individuals, a representative set of DOE sites and facilities, and a regional representation throughout the U.S.

SHORT TIME FRAME

The Committee's time frame for completing its charter was unusually short for an issue of this magnitude. In researching the charters and activities of other advisory committees, both within and outside of DOE, the Committee staff found that most committees had much longer time periods to conduct its work. Many committees, such as the DOE Site-Specific Advisory Boards, are under continuing charters. Secretary O'Leary tasked this Committee with only one year to present recommendations on a topic that could have significant impact to DOE workers and managers as well as facilities and operations . The recommendations, it was known from the beginning, could involve significant action by the Administration, Congress, and other state and federal agencies.

MINIMAL RESOURCES

The Committee was provided minimal resources to carry out its charter. This required creative means of leveraging existing resources within the DOE community and other agencies. The Committee's staff involved only 6 full-time professionals.

MEETING THE CHALLENGE

The Committee staff needed to be resourceful and creative to best serve the Committee. The staff worked to serve the Committee members, and this included identifying quickly what resources could best assist them.

ASSIST LEARNING CURVE

Because of the diverse make up of the Committee, many members were selected, in part, because their experience and expertise came from outside the DOE community. However, this meant a learning curve for those members. Therefore, this Committee actively sought and listened and learned from all the invited and public speakers at its meetings. In

addition, the Committee solicited papers and other information to assist in a better understanding of the complex problems related to external regulation of DOE facilities.

IDENTIFY INTERESTED PARTIES

The Committee proactively identified and obtained the participation of a broad representation of internal and external DOE stakeholders. These included:

- DOE workers and managers
- Union representatives
- DOE contractors (workers and managers)
- State and local governments
- Local businesses
- Professional organizations
- Public interest organizations
- Environmental groups

One of the major challenges for the Committee staff was to successfully identify both the internal and external stakeholders and to encourage their participation. We used a variety of approaches to inform the public and workers about the Committee's activities and solicit their input. We understood that input from a variety of viewpoints and sources would provide the Committee with useful information in the development of their findings and recommendations.

Personal Interaction by Pre-Visits with Stakeholders

The Committee staff conducted pre-meeting visits with stakeholders at or near the DOE sites the Committee planned to visit. These visits were usually held 2 to 3 weeks prior to each public Committee meeting and efforts were made to find interested persons including labor representatives and local citizens. We held meetings with federal, tribal, state, and local government representatives, as well as local and national stakeholder organizations. The staff informed every individual and group we met with of the Committee's activities and mission and encouraged them to participate in the upcoming public meeting in their area. We found many of the "usual concerned citizens" that closely follow DOE activities at nearby sites did not know of the Committee or its mission. Many also were not that interested in the topic of external regulation or did not understand that it could mean an entity completely outside DOE to oversee DOE facilities.

To remedy this lack of knowledge and to better educate parties as to the purpose of the Committee, these pre-visits served as a 'heads up" to those who might be interested in the oversight of DOE activities. Usually 2 committee staff members with one contract support person briefed DOE, contractor management, workers, and local stakeholders about the Committee's diverse membership, its origin, activities and meetings to date and how they might help the Committee in meeting its mission and providing recommendations.

Stakeholder involvement was sought at all levels. The following provide examples of the diversity of pre-visit meetings:

- Mayor of Richland, Washington
- Governor of San Ildefonso Pueblo, New Mexico
- Colorado Department of Health
- Snake River Alliance Group (Idaho)
- Chamber of Commerce in Albuquerque, N.M.
- American Nuclear Society in Richland, WA

Frankly, the staff and the Committee were pleased with the level of knowledge that many people and groups possessed. The stakeholders were, in turn, surprised we would actually travel to meet with them in their surroundings and encourage their individual participation at the upcoming Committee meeting. We also found that most parties we met with would provide information and contacts about additional interested individuals and groups. Usually we could personally meet with them while on the pre-visit, but sometimes we spoke by phone or met a day or two prior to the meeting.

In an effort to inform DOE and contract workers about Committee meetings and encourage worker participation, flyers announcing the meeting were posted and electronic mail messages were sent to employees at the respective DOE sites. Articles about the Committee and meeting notices were printed in site news bulletins.

Presentations by Stakeholders at Advisory Committee Meetings

We also involved stakeholders in developing meeting agendas based on issues specific to each site, in suggesting invited speakers to address the Committee, and participate in panel session presentations during Committee meetings. For example two opposing citizen concern groups, "Los Alamos Study Group" and "Responsible Environmental Action League" gave formal presentations at the Advisory Committee Public meeting held in Santa Fe, N.M. and participated in panel session presentations during Committee meetings.

OTHER OUTREACH MECHANISMS

Public Meetings/Stakeholder Comments: The Committee held eight meetings in nine months, all of which were open to the public. Sixty members of stakeholder organizations addressed their concerns about DOE safety. Over 350 members of the public attended the meetings. Public comment sessions were held in conjunction with all the Committee meetings. Forty seven concerned citizens shared their viewpoints and answered Committee questions.

Written Comments: For concerned citizens unable to attend the public comment sessions, different options for the Committee to receive comments and input were offered: U.S. Mail, Facsimile, and Electronic Mail.

Dissemination of Advisory Committee Information: Public information materials such as press releases, Federal Register Notices, and fact sheets were developed and distributed to keep the public informed of Committee meetings and activities. Meeting minutes, transcripts and the Committee's Status Report were also available to the public. This information was distributed upon request, at Committee meetings, and sent to DOE Reading Rooms around the country. The Committee maintained a mailing list of over 500 persons. Over 9,000 pieces of information were distributed during the Committee's tenure.

Information Center: Toll-free phone line. An information repository with a toll-free number was established to respond to information and document requests and answer Committee-related questions. Over 535 calls were received on the toll-free number.

Fact Sheet: We decided that a fact sheet was a good way of distributing important general information about the Committee. We created an "About the Committee" fact sheet which included the purpose of the Committee, described the issue of external regulation, provided information on how to both comment and obtain information, listed the membership, and indicated when and where Committee meetings would be held. The fact sheet was updated prior to each meeting, and provided a vehicle for noting both

changes in membership (due to other commitments by 2 members) and changes in meeting locations when the Committee decided it had collected enough information and needed to focus only on deciding issues and reaching conclusions.

Internet/World Wide Web: Committee information was also maintained on a World Wide Web Page on the Internet, where it could be viewed and/or downloaded.

Early in the Committee's establishment it became clear that there was tremendous interest in the Committee's work and very limited staff resources to deal with requests, so we sought new avenues for reaching out to the public. One of the areas of greatest growth is the World Wide Web. Another is the E-mail part of the Internet. We established an e-mail address specifically for the Committee and posted its address in all the press releases, Federal Register notices, and regularly updated fact sheets, along with the World Wide Web address and other methods for contacting the Committee.

We contacted the DOE-Office of Environmental Management staff and contractors responsible for the EM World Wide Web page and they agreed to create a page for the Committee work. It soon became apparent that the public was only going to use the page if it was regularly updated to provide current information. Prior to each Committee meeting we posted the agenda, press release and Federal Register notice, and after each meeting we posted the transcripts and minutes. In addition, all the subcommittee reports, internal working group reports, and staff papers have been posted. An employee at the Stanford Linear Accelerator recently e-mailed us that he though the Committee's Web page was the best in the Federal government. By the end of January we had over 28,000 accesses to the Committee's page and, after release of the Committee's final report, it was growing by over 1,000 accesses per day.

Letters to the DOE Complex Requesting Criticisms and Solutions: To make certain that all potential issues and solutions were considered, the co-chairs sent a letter to all Operations and Field Office managers, Laboratory Directors, and Program Offices. These resulted in numerous direct responses from across the complex which provided additional insights (concerns, problems, and potential solutions of day-to-day implementation of the current regulatory structure) from the DOE working management level.

Committee Members As Listeners

At Committee meetings, Committee members actively engaged in discussions with concerned citizens, workers, and management concerning safety and health issues at DOE nuclear facilities. Public comments were considered in the development of the Committee's report. There are over 400 transcript pages from public comment sessions.

The Committee listened intently to the invited speakers and the interested public. Meetings often lasted well into the night. At some meetings, the Committee stayed long after the meetings closed to discuss among themselves what they learned in the meetings.

Not only did the Committee hold meetings near the DOE sites, but also toured specific facilities at the sites in order to have a better understanding of the facilities and had some meetings at the sites to encourage worker input.

LEVERAGING EXISTING RESOURCES

To make up for the short time-frame and minimal resources, the Committee staff leveraged existing resources within and outside the DOE complex to

identify and obtain stakeholder involvement, publicize the Committee's activities, and to educate the Committee members on DOE facilities and operations. The objective was to not "reinvent the wheel" but to maximize existing resources and local knowledge. We effectively utilized DOE personnel expertise and their knowledge of the local issues at all of the sites that the Committee visited. The DOE field personnel -- from the Operations Office Managers to the Communications, Public Affairs, and Outreach offices were essential to our successful institutional program. We found the site personnel, both DOE and contractors, eager to help us arrange the pre-visits, site tours (on a very restricted time frame), and press notices. In addition, for each meeting, DOE headquarters personnel provided their distribution mechanisms so we could widely publicize our meetings both inside and outside DOE. Committee staff also attended some SSAB local meetings and SSAB members were speakers and provided other sources of information to aid the Committee members, We contracted with an existing resource, the Office of Environmental Management's Center for EM Information, to respond to inquiries and provide distribution of documents.

BENEFITS FROM STAKEHOLDER PARTICIPATION

DOE Worker Community:

The Committee heard a lot from the DOE worker community (workers, unions). What it learned was that there is a regulatory morass. The workers on the floor saw above them so many layers of management, oversight, and regulatory requirements, including different factions of DOE, that they were confused as to what requirements they had to meet and when, and who was in charge. At every site the Committee visited, the workers communicated that "things were not clear. " The Committee also heard that DOE is a dangerous place to work. They cited a lot of workplace hazards that needed to be addressed, both radioactive and nonradioactive, particularly in light of the changing mission of DOE from production to cleanup. Another concern expressed was that new workers were being brought on site who are not familiar with the very old sites and facilities and their unknown hazards. In addition, the training of subcontractors is not often the same as with the prime contractors on site.

Workers also told the Committee that, due to the manner in which DOE currently prioritizes work, some hazards were going unaddressed while less "risky" activities were being "fixed".

States:

The States were not at all unified in the need for external regulation or how it should be implemented. For example, the State of Colorado believed it could best regulate Rocky Flats. The concerned citizens near Rocky Flats also agreed with State regulation, but with the Federal government providing standards and guidance. However, other States, like the State of New Mexico, believed that DOE should take care of its own problems and that the State did not have a role to play in regulation of a Federal facility.

Public:

The public was unified in its concern for public and worker health and safety. The public generally acknowledged that DOE was more open than it had been previously but, like the workers, did not know for certain who was in charge. Local communities, including local officials, were concerned with the economic well-being of their communities in light of DOE's changing mission and the role of the community in overseeing the

safety of DOE facilities. In general, most members of the public were in favor of independent regulation outside of DOE, meaning, yes, external regulation. However, there was not a consensus among the public as to who was best suited to take on this role. The public, as did the workers, realized the need for clarity in what regulations had to be met.

DOE Managers:

The DOE managers and laboratory directors (former and current) generally agreed that the time had come for external regulation and endorsed the concept. They did not have specific recommendations, however, on who best is suited to provide the external regulation.

Committee Recommendations

All the Committee's recommendations were predicated on the core belief that a meaningful role for citizen's and workers is essential to the development of credibility and trust. The Committee noted that "Two elements are required to implement effective public participation in the regulatory process: timely and adequate information and effective means to influence the outcome." Several recommendations by the Committee were a result of the significant public participation role and the information obtained from the various stakeholders. Included in the Committee's recommendations were several directly related to State and public involvement:

States with programs authorized by EPA, OSHA, or the facility safety regulator should acquire or continue to have roles in the regulation of all respective areas of safety comparable to those they exercise in the private sector;

In principle, States should be able to enact more stringent standards as long as these do not unduly hinder a DOE mission;

Citizen's suit provisions, comparable to those in the environmental statutes, should be available for regulation of safety at DOE nuclear facilities, both with regard to enforcement against DOE and its contractors and with respect to non-discretionary statutory duties of the regulators;

The DNFSB should make its processes more open, including issuing recommendations in draft form for comment.

CONCLUSION

This was a unique challenge -- both from a public policy perspective as well as from an institutional program perspective. The was not your typical advisory committee -- in make up, dedication and cooperation and enthusiasm for the task. This was a difficult assignment for Committee members. No one had tackled a review so great for the need for external regulation. This was an unprecedented outreach effort, which was applauded by all Committee members (despite long public meeting hours) and the stakeholders we reached. This was a very rewarding effort. We met the challenge through using a proactive approach, leveraging of existing resources, and taking advantage of such new outreach tools as the World Wide Web. We've thought of writing a book on this one!

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BUILDING STAKEHOLDER GROUPS
THROUGH TECHNOLOGY

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ABSTRACT
The Nevada Risk Assessment/Management Program (NRAMP) is tasked with performing an independent, integrated risk assessment of the Nevada Test Site (NTS) in close cooperation with stakeholders, who will then use the assessment to prioritize future land use options at the NTS. As part of this process, NRAMP is exploring a variety of methods for involving stakeholders and will develop the most successful methodologies into a model for other DOE sites. The Program has made technology the focus of its methods through facilities available at University of Nevada Las Vegas (UNLV) research centers. The selected technologies include a telephone survey through the Southwestern Social Science Research Center; videoconferencing in the especially equipped rooms of UNLV's TeleMedia Services; a World Wide Web HomePage via the university's Internet connection; and electronic meetings in the Decision Support Laboratory at the Harry Reid Center for Environmental Research. Successes and difficulties with these methods are being formulated into recommendations.

INTRODUCTION

In early 1994, DOE's Office of Environmental Management (EM) created a cooperative agreement that would fund innovative methods to achieve meaningful stakeholder involvement in a risk-based approach toward remediation of DOE sites across the nation. The Nevada Risk Assessment/Management Program (NRAMP) is one of five successful proposals for collecting stakeholder concerns and providing the scientific data stakeholders need to participate in an informed debate about the future. The NRAMP is housed in the Harry Reid Center for Environmental Studies (HRC) at the University of Nevada Las Vegas (UNLV) and focuses on the Nevada Test Site (NTS). Its three primary missions are to (1) work closely with stakeholders, EM, and others in developing independent risk assessments of the NTS; (2) evaluate and prioritize options for NTS restoration and future land use from a stakeholder perspective; and (3) develop a methodology for stakeholder-based risk assessments at other sites.

In Nevada, these missions are impacted by a problem more immediate than the usual human interaction difficulties: four hundred miles separate the two major population centers of Las Vegas and Reno, and even greater distances lie between the more remote regions of the state. People must come together physically before they can come to consensus on the issues. To solve the problem, team members have turned to technology.

Using UNLV facilities and research centers, the Program has focused on relatively inexpensive techniques that can be applied through cooperation with universities at nearly any site. The selected technologies include a telephone survey through the Southwestern Social Science Research Center; videoconferencing in the especially equipped rooms of UNLV's TeleMedia Services; a "HomePage" via the university's Internet connection; and electronic meetings in the Decision Support Laboratory at the Harry Reid Center for Environmental Research. In addition, the team hopes to study stakeholder reactions to electronic presentations by developing a PowerPoint "primer" that could be made available on CD-ROM.

TELEPHONE TECHNOLOGY

The NRAMP grant period (originally 18 months, now extended to 21) did not allow time for the regulation keystone process in forming a stakeholder

working group, and the team made an early decision to accept any and all participants who wished to join the Program. Traditional mailing lists and newspaper advertisements were used to announce the Program to stakeholders, but the team was well aware previous experience had shown groups recruited in these ways to be small and more representative of vested interests than the state's actual demographics.

To achieve representation without moving "keystoning," team member Dr. Dennis Soden from UNLV's Social Science Research Center used telephone technology to sample a broader base of Nevadans' attitudes toward the NTS and to collect their opinions of future land uses of the site. Working with a hired team of graduate students who had experience in telephone surveys, he oversampled the state's rural areas to ensure their representation. In all, he collected 400 opinions from the Las Vegas area, 400 from Reno/Carson City, and 400 from across rural Nevada. This is not to say that telephones reach all of Nevada's citizens. In fact, two members of the current NRAMP Working Group cannot be contacted by phone and have even been reluctant to give a specific home address. Nevertheless, the sample population reached in the telephone survey is certainly more representative of the State's general population than the working group that has been formed.

The telephone survey serves the Program as a check-and-balance on attitudes and opinions collected from the NRAMP Working Group. However, while this technology will certainly be recommended to other DOE sites where rural populations are an important factor, several caveats must be noted:

1. working groups will want to know if the survey carries more weight than their group opinions,
2. teams can expect considerable debate over the content and wording of survey items, and
3. working groups may distrust answers from the sample population and accuse callers of making leading comments - even when a "script" is used to ensure each call follows a standard format.

In general, there seems to be a reluctance to accept human comments when they are not gathered in a face-to-face situation, but given the expense of door-to-door canvassing and the notoriously low return rate for mail surveys, the telephone offers a comparatively quick and inexpensive overview of public opinion and, specifically, of rural opinion that might otherwise be overlooked. It is tempting to speculate on the kinds of survey results that may be achieved when the telephone is fully integrated with the television as has now been done in parts of Canada.

VIDEOCONFERENCING

At the start of the Program, two working groups were formed: a Northern Group in the Reno area and a Southern Group in Las Vegas. Originally, all NRAMP members attended the Southern meeting and three or four flew to Reno for the Northern meeting. However, after two months, the Northern Group remained just half the size of the Southern (i.e., 20-25 vs. approximately 10) and it became increasingly difficult to integrate Northern decisions and opinions with those in the South. In addition, the logistics and productivity loss involved in flying team members to Reno quickly became a burden. Once again, technology provided an answer. NRAMP turned to UNLV's Telemedia Services, which had recently announced its videoconferencing rooms were available to university projects. Working with Mr. Michael Stowers, Telemedia's Executive Director, the team explored its options for long distance communication. The situation

was enhanced by the close connection between University of Nevada Reno (UNR) and UNLV; electronically delivered classes are regularly scheduled between the two universities and most technical difficulties had already been solved.

Team members attended a demonstration session, learned how to use the camera controls, and chose the most effective room arrangement for the size of the NRAMP Working Group. The Southern Group immediately settled into the more appropriate of UNLV's two videoconferencing rooms and has now participated in three meetings at that location. The Northern Group has changed locations once, from the Education Building to the System Computing Services Center, and now appears comfortable in a permanent setting. Aside from minor sound difficulties, which largely involve training speakers to stand in specific locations or address their remarks to a particular microphone, the technology is now very accepted by both groups and, in fact, when the NRAMP returned to a traditional separate meeting for one month, stakeholders in the North expressed great dissatisfaction.

The facilities at the two universities allow the NRAMP to project two views to each location: a view of the room (i.e., the participants) and/or the speaker and, through a document camera, a view of presented materials such as maps, 3-dimensional objects, and overheads. In the future, the NRAMP plans to integrate presentation software such as PowerPoint or Harvard Graphics.

At this point in time, one NRAMP team member flies to Reno and presents portions of the meeting agenda from that location. Presentations are usually reinforced by sending hardcopy handouts to the North either in a pre-meeting mailing or with the attending team member. The expense of this travel, coupled with the universities' videoconferencing fees, means that the overall cost of the meetings has not been reduced (the cost of flying three or four persons to Reno is roughly equivalent to the cost of videoconferencing), but team members generally experience a considerable gain in productivity because all but one can spend the entire meeting day on task rather than travel. The most important gain, however, is in group interaction: though small, the Northern Working Group is now able to present its opinions, participate in general discussions, and be fully represented in any group issue.

Videoconferencing is highly recommended to projects with access to university facilities like those at UNLV and UNR. Cost is the major barrier to widespread use of the technology in the business community, but cooperation with universities can dramatically reduce this factor. Stakeholders adapted very rapidly to the few necessary constraints on meeting procedures, and the NRAMP team considers bringing the Northern and Southern Groups together as one of its major successes.

FUTURE TECHNOLOGIES

With the telephone survey completed and videoconferencing established as the primary meeting methodology, the NRAMP is now focusing on other technologies to enhance group communication: the World Wide Web (WWW), presentation graphics, and Group Decision Support Systems (GDSS). Team members with expertise in these areas are developing interfaces and strategies to draw more participants into the Working Group and to increase participation in general.

WWW HOMEPAGE

The World Wide Web is very close to linking even the most remote areas of the world and certainly provides an important service to states like

Nevada. As more and more stakeholders subscribe to home services (e.g., America OnLine, CompuServe, Prodigy) and access the Web, it is becoming one of the best means of distributing information needed for good community decision making. The Department of Energy has been using the technology for sometime, and in Nevada, for example, the Coordination and Information Center (CIC) operated for DOE by Bechtel gives electronic access to more than 300,000 historical documents related to nuclear weapons testing, human radiation experiment records, and many other documents concerning programs, projects, and activities.a

Universities enjoy rapid access to the Web through their dedicated lines, and projects like the NRAMP can provide physical access for online information very similar to the videoconferencing access described above. Stakeholders who cannot go online from a home computer can use university-based workstations to gather background materials and information on the issues.

At UNLV, the NRAMP has recently implemented a 12-station computer laboratory (see below) in the Harry Reid Center for Environmental Research (HRC), the home of the NRAMP, that could soon provide general stakeholder access to public DOE databases and specific access to the NRAMP HomePage being constructed by team members under the direction of Dr. Muhilan Pandian. Several kinds of information are being considered for possible entry: a photo history of land uses at the Nevada Test Site; team member photos, areas of expertise, and contact numbers; specific NRAMP documents such as the original grant, the Work Plan, and the Preliminary Risk Assessment results; presentation graphics used in the face-to-face Working Group meetings; and Geographic Information System (GIS) maps produced by the NRAMP technical team. Dr. Pandian's staff are also responsible for the development of THERdBASE, a graphical database engine for integrating and viewing risk assessment results, and the NRAMP is considering placing THERdBASE online for stakeholders.

The most important use of the NRAMP HomePage, however, could be as a collection point for comments on the difficult issues facing the Working Group. If such issues were placed on the HomePage, stakeholders could maintain a running discussion that would be available without time and space limitations for those with home access and on a regularly scheduled basis for those who wish to use the HRC laboratory. Access can, of course, be provided from DOE sites as well. The NRAMP expects many "hits" on its HomePage as additions are made to its content and stakeholders become aware of its availability.

A related idea has been explored by one of the team's technical task leaders, Dr. Dennis Weber, who suggested that all handouts and graphic materials presented in Working Group meetings be compiled into an "NRAMP Primer" for individual or class use outside of Working Group meetings. Because some of these materials have been based in electronic presentations, the team has recently begun to consider placing them on the HomePage in an interactive format. Before that occurs, however, the team would like to conduct several experiments to determine stakeholder preferences for presentations.

Several of the team's technical staff, for example, object to the flashy "fade-in, fade-out, build-from-the-right/left" capabilities of presentation applications like PowerPoint. They feel audiences are distracted from the real content of the materials. Other team members feel that audiences expect an "infotainment" format and quickly become bored with text-heavy viewgraphs. The NRAMP hopes to structure several

experiments that would provide input to these arguments and determine the best possible format for storing information on the Web in an electronic NRAMP Primer.

Once again, the idea of a HomePage is an idea that can be exported readily to other projects and DOE sites. Specific recommendations will be made about Internet technologies when the NRAMP has had sufficient time to fully implement them and to study stakeholder reactions, but there is certainly reason to believe that a HomePage is a viable choice for stakeholder groups based on its low cost and its widespread availability.

DECISION SUPPORT LABORATORY (DSL)

The Harry Reid Center's DSL was funded by a cooperative grant from the Department of Energy's Yucca Mountain Project; it is under the direction of NRAMP team member Dr. Pat Jonker, who recently implemented Ventana's GroupSystems for Windows on the laboratory's 12 computer stations. This software is a group decision support system, or GDSS, that uses a networked system of computers and a large display screen to facilitate electronic meetings. Participants use the computers for brainstorming, organizing, prioritizing, and voting on their ideas and opinions. Meetings may be held in a traditional face-to-face setting with all participants present in the electronic meeting room or in a remote access format that allows participants to sign in to the meeting from their offices at any time of the day or night.

A GDSS is ideally suited to NRAMP-type projects in which stakeholders comment on issues that will eventually be subjected to a consensus process or a vote. It overcomes many of the typical meeting problems such as lack of participation, fear of retaliation, dominance by strong personalities, and subjective minutes or records of what occurred. Because participants enter their comments anonymously from the keyboard, no one can determine who contributed a specific comment; because participants do not need to be recognized by the chair to contribute, everyone can "talk" at once; and because all input is automatically recorded, no one has an opportunity to editorialize outcomes. Votes are recorded instantaneously and the result is shown graphically on the screen.b

The DSL was used for the first time in January 1996 as a platform to collect public perceptions of risk related to DOE Risk Data Sheets (RDSs) and budget priorities from 12 participants in the RDS process.

To further enhance anonymity, all workstations in the DSL are recessed into NOVA desks that provide a large conference style table when the computers are not in use. Although only 12 stations have been installed in the lab at this time, ports are available for 12 more persons to join a face-to-face meeting with portable computers and as many as 75 persons can contribute to a remote meeting when users are not signed on concurrently. The Pentium/75 workstations are each equipped with multimedia features (i.e., CD-ROMs, sound and video boards) and, in the near future, will be connected to the Internet. They offer an ideal location for Southern group members to access the NRAMP HomePage or view DOE information from databases across the country. The NRAMP anticipates an interesting series of experiments with lab use during the remainder of the project period.

Such a laboratory and its GDSS are obviously expensive technologies that are not likely to exist at every DOE site or even at every university. But the possibilities are seductive: imagine attending a meeting while sitting in your own office where all the information you need is readily

at hand; consider signing on to read the comments generated by others at a meeting you missed and then casting your vote at 11:00pm before you go to bed; and forget the drudgery of typing up and reproducing the minutes of your meeting. While meetings as we know them will certainly not disappear in the near future, many options are appearing that will change them dramatically. The creators of Ventana's GroupSystems, for example, are already at work on a facility that will feature "floor-to-ceiling high-resolution video display walls," with each wall showing a full-size representation of meeting participants in other locations and creating the illusion that those participants are actually present.c

FUTURE PROJECTIONS

From the four major technologies discussed, the NRAMP will undoubtedly recommend the telephone survey and videoconferencing as part of a stakeholder interaction model for other sites. But with the Program at only the halfway point, time remains to experiment, not just with HomePages and group decision support systems, but a host of other possibilities. One of the great advantages of working within the university environment is the opportunity to try new applications - the most exciting of which has not yet been built or tested.

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THE FERNALD CITIZENS TASK FORCE: SHIFTING THE FOCUS*

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ABSTRACT

In August 1995, the Fernald Citizens Task Force provided the U.S. Department of Energy (DOE), the Environmental Protection Agency (EPA), the Ohio EPA (OEPA), and FERMCO with cost-effective, implementable recommendations for addressing Fernald's most pressing issues. Fernald is now proceeding with an accelerated cleanup plan to make these recommendations reality. With its initial work plan complete, the Task Force confronted a new challenge: How to shift its focus from developing recommendations designed to influence Fernald's Records of Decision to advising project managers during remedial design and remedial action. This paper reports on the experiences of the Task Force, the DOE, Fernald regulators, and FERMCO as the Task Force made this shift. In the process,

the parties encountered issues involving work plan development, membership, organization, and support resource allocation. Lessons learned as these issues were resolved are summarized. The Fernald experience supports the conclusion that "hands-on" citizen involvement in government decision-making at a major environmental remediation site can effectively transition from one area of focus to another.

INTRODUCTION

Under its original charter, the Fernald Citizens Task Force experienced unprecedented success as a U.S. Department of Energy Site-specific Advisory Board by recommending publicly accepted, cost-effective solutions to critical issues facing Fernald Environmental Management Project decision-makers. This success was the outcome of a deliberate strategy pursued by the Task Force Chair of focusing the Task Force's agenda on "big picture" issues. The Task Force incorporated its recommendations in a report delivered to the DOE in August 1995.

With its initial work plan completed, the Task Force confronted a new challenge: How should its members refocus their energies to further advance the ultimate remediation goal at Fernald?

This paper reports on interactions between the Task Force, the DOE, Fernald regulators, and FERMCO as Task Force members shifted their focus from developing recommendations designed to influence Fernald Records of Decision to a new role during remedial design and remedial action. In the process, the parties encountered and resolved work plan development, membership, organization, and support resource allocation issues. Their experiences provide insight into how "hands-on" citizen involvement in government can effectively transition from one area of focus to another.

BACKGROUND

In the 1980s, it was discovered that the Fernald facility had been contaminating local drinking water for many years. Citizen activism in reaction to disclosures of the contamination drew local and national news media attention and significant negative publicity critical of the DOE. In reaction, the DOE issued news releases and held public meetings to inform citizens of Fernald activities. Early public meetings typically provided forums for "Decide, Announce, Defend" style presentations by DOE officials, which often resulted in citizen protests and accusations. Trust between the DOE and area residents was virtually non-existent. Prompted by separate lawsuits by the state of Ohio and area citizens, the DOE began to address contamination issues that had become a matter of public concern.

In 1991, the DOE and the EPA signed an Amended Consent Agreement, establishing key milestones and strengthening the framework for federal regulatory oversight of the Fernald cleanup. As work progressed under this agreement, DOE managers began to realize that direct citizen involvement was an essential ingredient to arriving at sound decisions. In the spring of 1993, the DOE decided a citizens advisory board would be the most cost-effective means of obtaining focused stakeholder input on Fernald's most pressing issues. An independent convener was hired to identify potential Task Force members to represent the wide-ranging interests of Fernald's various stakeholder groups and to develop a draft charter for the Task Force, in conjunction with the DOE, the EPA, and the OEPA. The charter drafted by the convener, and ultimately adopted by the Task Force chair, charged the Task Force with developing recommendations on four far-reaching issues: future land use, remediation levels, waste disposition, and remediation priorities at the Fernald site.

The Task Force officially convened in August 1993. Early on, Task Force members recognized that a recommendation regarding future use of the Fernald property following remediation would give direction to its deliberations and provide a framework for other recommendations. During this time period, the Task Force also determined an independent consultant should be retained to provide the Task Force with technical and facilitation support. The Task Force chair guided the consultant in developing a work plan to promote productive use of time during the 18-month period preceding issuance of Task Force recommendations to coincide with the draft Record of Decision for Fernald's Operable Unit 5. A key to this work plan was the conscious decision of the Task Force not to review and evaluate each decision and piece of information that would be released by the DOE over that period of time, but to focus solely on developing its work plan in the time available.

VALUE ADDED THROUGH PUBLIC INVOLVEMENT

As the work progressed, the Task Force learned how site decisions were being made and how the Task Force might influence those decisions. Through direct participation at Task Force meetings, DOE and FERMCO project managers gained insight into stakeholder values and expectations. Major policy issues were discussed at length, and technical, economic, and political trade-offs were debated. These interactions ultimately yielded agreement between Task Force members and Fernald project managers on the direction the project should take. The Task Force's recommendations provided a blueprint for project managers to follow in remediating the Fernald site.

For DOE and FERMCO, the return on this investment in public involvement in the Fernald decision making process has been significant. By implementing Task Force recommendations, the project duration will be reduced from 25 years to 10 years, and United States taxpayers stand to save an estimated $2.15 billion. These savings are determined by comparing the costs in dollars and time required for off-site disposal of all Fernald contaminated materials with the balanced approach recommended by the Task Force, which calls for disposal of Fernald's most contaminated materials off site and lesser-contaminated materials on site on an accelerated schedule. Equally important is the payback in terms of increased trust and confidence among Task Force members and project personnel. By their willingness to share information and adjust positions once the perspectives of others were heard, Task Force members and project managers set the tone for future cooperation in support of the mutually beneficial goal of a remediated Fernald site.

FERNALD IN TRANSITION

During 1995, the Fernald project was in transition. The signing of the Operable Unit 5 Record of Decision by the EPA in December 1995 symbolized the conclusion of a decade-long study-and-planning phase at Fernald. The project had crossed that important threshold into the remedial design phase.

The delivery of the Task Force's recommendations in August 1995 also symbolized a turning point. With the urgency of meeting deadlines under its initial work plan removed, Task Force members could consider a future course of action. At their August 1995 meeting, a consensus emerged in support of refocusing the Task Force in a manner that would be productive and useful for project managers. Concurrently, DOE officials reiterated their perspective that the relationships between site personnel and

stakeholders established during Task Force interactions were of great value and should be sustained.

At their September 1995 meeting, Task Force members discussed, in greater depth, issues relating to their future areas of focus and methods of conducting business. These issues included work plan development, membership, organization, and support resource allocation. How these issues were approached and resolved is discussed in the following section in a "lessons learned" format.

LESSONS LEARNED

Work Plan Development

Lesson Learned:

A focused work plan enables citizen participation to have the greatest impact on project decisions.

Discussion:

An important component of the original success of the Task Force was the ability of members to stay focused on their ultimate goal and particularly to understand how each activity fit into the process of achieving that goal. This was achieved through the development of a detailed work plan which identified all activities the Task Force would undertake, the information to be evaluated and the decisions required over time. The Task Force activities were clearly linked to ongoing site activities so members could see how their input would impact the real-time decision-making and cleanup of the site. By clearly outlining the role and importance of each meeting in the decision-making process, attendance was rarely an issue.

In discussing the future of the Task Force, members expressed an aversion to becoming "just another oversight group." It was generally considered important that a new work plan should include definite end products for each area of focus. The Task Force consultant was tasked with developing such an approach.

Informal dialogue among the Task Force chair, Task Force consultant, DOE officials, and FERMCO personnel occurred frequently during work plan development. Through these discussions, current and emerging issues were identified as candidates for Task Force focus. The Task Force chair reviewed the draft work plan with the DOE Fernald Area Office director to ensure it would result in relevant recommendations for project decision-makers.

At its December 1995 meeting, the Task Force refined and adopted the work plan. The work plan focused on the following issues: the engineered on-site disposal facility, waste transportation, environmental monitoring, natural resources, waste treatment, waste disposition, recycling, accelerated schedule/priorities, complex-wide issues, radium extraction, and economic development. For each of these issues, expected actions were identified. These actions are oriented toward developing recommendations or providing comments on these important issues. On all issues, Task Force dialogue and development of recommendations or comments have been scheduled in a timely manner to coincide with upcoming project activities and schedules.

Membership

Lesson Learned:

The Task Force should periodically assess its membership to ensure balanced representation and depth of expertise.

Discussion:

Upon submission of recommendations developed under its initial work plan, submitted to the DOE in August 1995, all Task Force members had the option to resign. None chose to do so. Instead, all expressed continuing interest and commitment to involvement on the Task Force.

The Task Force discussed the issue of membership at its September 1995 meeting. During the discussion, members considered the need to ensure balanced representation and depth of expertise among its membership. A determination was made that one additional member should be recruited. The Task Force chose to appoint a search committee to recruit candidate members. More than 850 announcements and applications were mailed to area residents, with ads being placed in the three Cincinnati-area daily newspapers. Eighteen applications were received, from which four finalists were interviewed to assess level of interest, experience, and proximity to the site. The entire Task Force approved the committee's selection at its December 1995 meeting, and the new member was appointed by the DOE later that month.

The membership review and recruitment process led the Task Force to reflect on its relationships with its constituencies and reaffirmed the need for continuing broad public participation activities independent of the DOE. Both diversity and continuity were considered keys to effective membership. Members acknowledged constant monitoring of community interests and positions is important to ensure the Task Force is representative of the entire community. A consistent membership is important to maintain institutional knowledge, continued cooperation and commitment. The addition of a new member as the Task Force shifts its focus has helped broaden the perspective and experience of the Task Force. This continuity was ensured in the original ground rules for the Task Force, which includes a provision for overlapping membership terms. The member search itself also identified the level of interest in the community in serving on the Task Force and helped identify individuals who might work with the task force over time.

Organization

Lessons Learned:

Citizen groups should organize such that they can be flexible and responsive to changing needs.

Discussion:

To influence key decisions in the Fernald project's planning phase, the Task Force had adopted an aggressive schedule under its initial work plan. Extensive monthly meetings involving the entire Task Force was essential during this work because of the strategic nature of the issues being considered and the time constraints under which the Task Force was working. With the delivery of recommendations under the initial work plan complete, the Task Force acknowledged it could "shift gears" and operate with less intensity. Two changes were adopted to enable the Task Force to relax its aggressive schedule while pursuing a broad range of issues under its new work plan.

First, the membership agreed the full Task Force would meet quarterly, rather than monthly. To ensure members were informed on a timely basis, the Task Force consultant was tasked with publishing a monthly newsletter on project status and issues of concern to the Task Force.

Second, the Task Force decided to utilize a committee structure to as great an extent as necessary. This enabled Task Force members to debate and discuss a broad range of issues concurrently and promote timely

development of recommendations. It is also conducive to timeliness and responsiveness by the Task Force to emerging issues.

The next phase of Task Force activities will be focused more on reacting to site proposals and reviewing detailed design documents. A committee structure allows the Task Force to consider multiple issues concurrently and provides flexibility for scheduling meetings and addressing new issues as they emerge. In addition, the committee structure allows more direct interaction with other stakeholders interested in Fernald activities, simply because meetings will be conducted more frequently and less formally.

Technical Support

Lesson Learned:

Continuity in support resources is important to the success of the Task Force.

Discussion:

An important contributor to the Task Force's success has been full access to information relating to decision-making factors provided by the DOE and FERMCO. A side effect of this unconstrained access is the need to ensure the vast amounts of available information are synthesized into an understandable format consistent with Task Force needs. Task Force members realized this early on and requested DOE to provide for retention of an independent consultant support them. As work under the initial work plan progressed, the consultant became expert in meeting the Task Force's information needs.

The Task Force acknowledged that consistency of support would be a key element as it transitioned from one area of focus to the next. Consequently, it requested DOE to continue to provide for the services of the consultant selected to support the initial work plan. The DOE and FERMCO continue to provide access to information and work with the consultant to ensure information needs are met. FERMCO has designated an individual to provide liaison with the Task Force and the consultant. All parties recognize the importance of keeping information channels wide open.

Open, honest and trusted technical support has always been the backbone of the Task Force's success. Much of this support has rested on the work of the Task Force consultant and select FERMCO employees who have earned the trust of Task Force members over time and developed important relationships with the Task Force. The continued involvement of these individuals is key to the continuing confidence the Task Force has in the material it uses.

While the Task Force consultant works to collect, synthesize and distribute most of the information needed by the Task Force, individual members, particularly committee chairs, need to feel comfortable and confident in calling FERMCO and DOE personnel directly to ask questions, discuss issues or request information. It is through the constant give-and-take approach with the Task Force that many of the most important contributions to improving cleanup performance are made.

CLOSING

The performance of the Fernald Citizens Task Force provides compelling evidence that direct public involvement adds tremendous value to solving the difficult challenges confronting managers of environmental remediation projects. Though the main Task Force recommendations are in place, much of the critical action to implementing these decisions will be conducted during remedial design and remedial action at the site. In

its reorganization, the Task Force has positioned itself to monitor progress, identify problem areas and ensure that the spirit of its recommendations are followed. The difficulties of maintaining an effective Task Force over the long term are significant, but in our view this continuity is essential. All parties recognize the importance of building on the success and credibility of the original mission by ensuring the effective implementation of the concepts and spirit embodied by the Task Force recommendations. Focus, teamwork, knowledge and self-discipline -- all of which are important ingredients of the Fernald Citizens Task Force's success -- are all difficult to replicate. The continuation of the Task Force is the most effective approach to ensuring balanced representation of local citizenry in decisions that will impact lives and livelihoods at Fernald for many generations.

42-4

FOCUSING ON THE BIG PICTURE: CITIZEN PARTICIPATION AT ROCKY FLATS

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ABSTRACT

The Rocky Flats Citizens Advisory Board (CAB) is part of the national network of advisory boards formed by the Department of Energy (DOE) to provide advice and recommendations on cleanup and waste management issues at weapons complex sites. Early in its operations CAB was faced with responding to specific proposals from DOE without having answered or addressed more fundamental questions or issues. Specifically, CAB was asked to comment on the design of a landfill cap to permanently dispose of waste material at the Solar Pond area at Rocky Flats. While reviewing this plan, CAB realized it first needed to address a more fundamental question, whether disposal of waste material was acceptable at Rocky Flats. Questions concerning the design of the landfill cap were irrelevant until the more fundamental question was addressed. To resolve this problem, CAB developed a workplan to address what it called the "big picture" issues. By focusing on these issues, CAB was able to develop a foundation of understanding that guided it in providing more specific recommendations on DOE proposals for the cleanup of the Rocky Flats Environmental Technology Site. A key component of CAB's work was the development of a position on waste disposal at Rocky Flats, a concept opposed by CAB. One of the earliest proposals CAB examined using its "big picture' framework was the draft Rocky Flats Conceptual Vision. This vision called for disposal of low-level and low-level mixed wastes at Rocky Flats. Having already laid a foundation of understanding on this issue, CAB was able to develop a consistent recommendation opposing the vision and offering an alternative calling for monitorable and retrievable storage of waste materials while they remained at Rocky Flats.

AN INTRODUCTION AND BRIEF DESCRIPTION OF ROCKY FLATS

The Rocky Flats Environmental Technology Site (RFETS) outside of Denver, Colorado, formerly known as the Rocky Flats Plant, began operation in the early 1950's primarily to produce the plutonium triggers for nuclear weapons. These operations continued until 1989 when Rocky Flats was

forced to shutdown due to concerns with safety at the aging facility and indirectly as a result of a raid by the Federal Bureau of Investigation and the Environmental Protection Agency over illegal environmental and waste handling practices. The 1989 shutdown was thought to be temporary, but due to changes in the world political climate, the end of the cold war, and the declining need for nuclear weapons, the mission at Rocky Flats was changed to one of cleanup in 1993.

Even with a change in mission, the legacy of its past still remains. Namely, 12.9 metric tons of weapons grade plutonium and tens of thousands of cubic meters of left-over production waste, including transuranic, low level and low level mixed forms, still reside at the site. Thousands more cubic meters of waste will be generated during the decontamination, decommissioning and environmental restoration activities slated for the future. The challenge of what to do with these wastes is of keen interest not only to the Department of Energy and the regulators, but to the local communities and citizens of the Denver metropolitan area.

HISTORY OF CITIZEN PARTICIPATION AT ROCKY FLATS

Citizen involvement with Rocky Flats began in 1969 when stories of a fire in one of the production buildings made front page headlines in the Denver papers. Rocky Flats officials claimed that there were no off-site releases from this fire, but a group of scientists at the University of Colorado in Boulder felt otherwise and conducted independent soil sampling in areas off-site from the plant. The results of this sampling confirmed that there was plutonium in the soil that exceeded background levels expected from world-wide atmospheric weapons testing fall-out. In confronting site officials with this information, the researchers demanded an explanation. The response from Rocky Flats was indeed surprising. Site officials continued to deny any contribution from the 1969 fire, but instead disclosed a previous fire in 1957 and waste handling procedures during the past two decades that were more likely the contributors of the off-site contamination.

With these surprising revelations, many community individuals near the site began to question if there were other events that Rocky Flats officials had not disclosed and also whether the site was able to operate safely in the midst of a major metropolitan area. This skepticism, combined with anti-nuclear sentiments, provided the foundation for citizen interest and involvement in Rocky Flats operations. There was, however, no real forum for an open dialogue with site representatives, as citizen participation took the form of staged protests such as rallies, a human blockade of the rail lines running into the plant, and a human encirclement of the site boundary. Political interest in the site began with the creation of a task force formed by the Colorado governor and the member of Congress from the district in which Rocky Flats resides. This task force led to the formation of the Rocky Flats Environmental Monitoring Council, which was the first true forum attended by site officials, the local governments, and interested citizens to discuss what was happening at Rocky Flats.

Formal citizen participation began in earnest in 1989 when Rocky Flats was placed on the National Priorities List of Superfund. The public participation guidelines of Superfund brought an institutionalized approach to citizen involvement and the beginning of formal meetings to seek public input into plans for dealing with the environmental contamination problems. Recognizing the value of regular stakeholder input and involvement, the Department of Energy and the site contractor

expanded the dialogue to include many other documents and programs such as strategic plans, environmental assessments, budget guidelines, and site transition plans. The volume of information generated to support site activities began to fill entire libraries and the number of public meetings associated with Rocky Flats rose substantially. A public that once complained about too little information soon began to complain about too much, with too many meetings and a lack of focus for how all the pieces of public input fit together.

THE FORMATION OF THE ROCKY FLATS CITIZENS ADVISORY BOARD

In 1992, representatives from state governments, national environmental organizations, the regulators, the Departments of Energy, Defense, and Interior, and other interested parties formed a federal advisory committee known as the Federal Facilities Environmental Restoration Dialogue Committee to address cleanup at federal sites such as Rocky Flats. One of the recommendations of this committee was to form Site Specific Advisory Boards at each of the weapons complex sites to foster stakeholder participation in the cleanup decisions.

The Rocky Flats Citizens Advisory Board (CAB) was formed in 1993 in response to this recommendation with the following mission statement: "The Rocky Flats Citizens Advisory Board, a nonpartisan, broadly representative, independent advisory board with concerns related to Rocky Flats activities, is dedicated to providing informed recommendations and advice to the agencies (Department of Energy, Colorado Department of Public Health and Environment and the Environmental Protection Agency), government entities and other interested parties on policy and technical issues and decisions related to cleanup, waste management and associated activities. The Board is dedicated to public involvement, awareness and education on Rocky Flats issues."

CAB was not formed because of a lack of citizen involvement in Rocky Flats issues. To the contrary, the period before its formation saw the rise of numerous organizations whose sole or partial mission was to address Rocky Flats, each with its own particular viewpoint. CAB from its inception was unique, however, in that for the first time, representatives from these various organizations were brought together around a common table. CAB operates by consensus, a sometimes difficult process, but one that brings often disparate views together in a common voice that is seen as a valuable tool for those who receive the advice or recommendation.

A CASE STUDY IN DECISION-MAKING: THE SOLAR PONDS AT ROCKY FLATS

One of the first major issues addressed by CAB was what to do with the solar ponds at Rocky Flats. The solar ponds were used by the site for most of its operating life to receive process waste water which was allowed to evaporate in the open air. The remaining sludge left after evaporation was periodically removed as radioactive waste. Through almost forty years of operation, the ponds leaked, contaminating the surrounding soil and groundwater. In 1994, the Department of Energy (DOE) came to CAB with a proposal to build an engineered soil cap over the ponds as the chosen method of permanently closing them.

DOE's plan was to remove the ponds' liners and surrounding soils, and then redistribute the contaminated material into the center of the solar ponds area, after which an elaborate cap would be built over the area to isolate the materials from the environment. This plan was developed with the cooperation of both the Environmental Protection Agency and the Colorado Department of Public Health and Environment. In coming to CAB

with the proposal, DOE stated that the final plan was not yet developed and would allow CAB plenty of time to review and make changes to the proposal as it saw necessary. CAB has several standing committees that work on issues in order to develop a proposed recommendation for approval by the full membership. After receiving the solar ponds proposal, CAB assigned its Environmental/Waste Management Committee the responsibility of reviewing the plan.

For several months the Environmental/Waste Management (E/WM) Committee met with DOE and the regulators in discussing the proposal. A tour was scheduled for the site, and draft copies of the plan were given to E/WM for review far in advance of the "official" public comment period. As discussions progressed, DOE fine-tuned the plan based on some of the early E/WM comments. One of the troubling aspects of the proposal was DOE's plan to include the sludge that had been removed from the ponds over the years. This material, called pondcrete, had generated controversy several years earlier because the cementation process used to stabilize the sludge had proven ineffective, as the material failed to solidify properly. Further problematic was that one of the findings of the Grand Jury investigation which followed the FBI raid in 1989 was that the site operators had failed to properly store this pondcrete material. DOE now proposed to dispose this same material as part of the solar ponds closure plan.

Fulfilling its directive, the E/WM Committee came to the full board with its proposed recommendation that CAB approve the design of the solar ponds cap with several modifications. The disposition of the pondcrete remained an issue that the Committee could not come to agreement on and asked for the board's advice. The final recommendation by the board was that DOE could build the cap, but that it would have to consider the project as an interim storage measure, and that any materials placed under the cap would need to be in a monitorable and retrievable form so that eventually they could be removed for disposal elsewhere. The full board could not come to consensus on the disposition of the pondcrete materials either. This recommendation was contrary to DOE's view that the cap would be permanent, "designed to provide a level of protection for at least a thousand years."

One of the lessons learned by CAB in its review and deliberation on the solar ponds cap was that it was being asked to make a decision on a very specific proposal without first having come to consensus on a broader issue. Namely, should any material, at any location be disposed at Rocky Flats. It became apparent to CAB that a hierarchy of more general issues and questions existed that must be answered first before other more specific proposals could be addressed. From its experience with the solar ponds, CAB resolved to develop a new approach to providing advice and recommendations to DOE and the regulators. It would develop a work plan, not driven by specific plans or proposals, but more general in nature to lay a broad foundation upon which future recommendations could be built.

LAYING THE FOUNDATION: THE DEVELOPMENT OF A CAB WORKPLAN

From its inception, the solar ponds closure plan was not the only technical proposal on which CAB was being asked to provide comment. CAB faced a fundamental question of whether it would be a reactive organization, responding only when asked, or whether it would become more proactive by compiling a list of what it termed "the big picture" issues and then developing strategies to address them. CAB chose the latter.

To begin its effort, CAB first developed a list of issues that were important to the membership. Next, these issues were combined and organized into logical program areas such as safety and health, cleanup, waste management, land use, and plutonium disposition. Members then determined which issues to address first. The results of this exercise culminated in the development of four issue areas that were assigned to the various CAB committees. These four issues were cleanup levels, waste disposal policy, plutonium disposition, and future use. The Department of Energy concurred with CAB that these issues were the most appropriate to develop, encouraging CAB to first work on the waste disposal issue. CAB also viewed the waste disposal policy as an urgent priority. This development of a waste disposal policy demonstrates how CAB identified broader issues as a prelude to developing strategies and recommendations on more specific issues and proposals.

THE CAB POSITION ON WASTE MANAGEMENT: STORAGE AND DISPOSAL AT ROCKY FLATS

The research and development of a CAB position on waste management at Rocky Flats was assigned to the CAB Site Wide Issues Committee. The committee began its work by eliciting a set of guiding principles that incorporated the group's values and beliefs associated with waste management activities. Although rather straight forward, the guiding principles are viewed by CAB as providing the foundation by which future waste management plans can be evaluated. These guiding principles are as follows:

1) Different categories of waste will remain at Rocky Flats for some period of time.
2) Waste at Rocky Flats shall be stored in the safest possible manner.
3) Waste shall be stored in such a manner that it cannot escape into the surrounding environment during the time that it remains a potential hazard.
4) Waste shall be stored in such a manner that it is fully monitorable and retrievable. In addition, there shall be institutionalized review systems in place ensuring inspection on a regular basis.
5) It is not acceptable for Rocky Flats to send waste to a facility that would not meet CAB criteria for storage at Rocky Flats.
6) Any waste storage or "disposal" facility must be selected through an objective, scientific process and include public participation.
7) The concept of "disposal" of radioactive waste is misleading due to the toxic and long lived nature of the wastes. Because wastes are out of sight, does not mean that the problem is solved.

After developing the list of guiding principles, the Site Wide Issues Committee next turned to the task of providing specific recommendations on waste management policy to the Department of Energy. These recommendations approved by CAB are:

1) Because it is unlikely that a waste "disposal" facility can be guaranteed to contain the contaminants for the life of the waste, CAB opposes the development of such a facility at Rocky Flats.
2) DOE shall develop plans for long-term storage of all wastes currently at Rocky Flats.
3) Any waste facility must be fully monitorable, with regular inspections, and the waste must be fully and easily retrievable.
4) To ensure the safest possible storage, any waste storage facility must be able to be upgraded; and if the facility cannot be upgraded the facility must be replaced.

5) No waste from other facilities shall be accepted at Rocky Flats for treatment or storage.
6) Because radioactive waste is a national issue, and because there is no guarantee that proposed "solutions" will proceed, a national dialogue must be convened that addresses the issue of waste storage and "disposal". CAB agrees to support and participate in such discussions.
7) DOE shall vigorously pursue a research program aimed at developing technologies to make radioactive waste benign (not a potential hazard).
8) The Rocky Flats Citizens Advisory Board and other external stakeholders (including regulators) shall be involved in the development and approval of all waste management plans and activities.
Through this focus on the broader, "big picture" of waste management, CAB was now in a position to evaluate specific proposals as they came forward with a common understanding of the guidelines and principles important to the members.

BUILDING ON THE FOUNDATION: THE UTILITY OF FOCUSING ON THE BIG PICTURE

After approval of its waste management recommendations, it did not take long before CAB was able to put them to use. Late in 1995, a draft Rocky Flats Conceptual Vision developed by DOE, the Colorado Department of Public Health and Environment, and the Environmental Protection Agency was released for public review and comment.
Work on the Conceptual Vision for Rocky Flats Vision began in 1995 through a cooperative effort with DOE, EPA, and the State of Colorado. These agencies had for the past several years been developing a new cleanup agreement to provide the legally enforceable framework for the cleanup of the Rocky Flats site. In the fall of 1995, the Principals from each of these agencies met in a "workout" session to come to closure on the outstanding issues between them. One of the products of their workout was the draft Conceptual Vision. This vision was based on two phases: an intermediate site condition and a final site condition. A major portion of the final site condition called for permanent disposal of numerous forms of low level wastes at Rocky Flats. In order to evaluate and comment on the vision, CAB was able to utilize its previous waste management recommendation.
A major component of the draft vision was the "long-term disposal" of wastes at Rocky Flats using landfills covered by earthen caps. Materials inside buildings, which during demolition would become low-level or low-level mixed wastes, would be added to a special cell in these landfills or would be entombed within the building foundations under a special cap.
Using its previously developed guidelines, CAB provided the following recommendation on the draft Conceptual Vision:
"Radioactive waste disposal is not acceptable at Rocky Flats. DOE should develop plans for long-term storage of waste in a manner that is fully monitored and retrievable."
CAB further added:
"Today's 'solutions' for problems at Rocky Flats should not leave problems for future generations, or preclude future generations from going beyond these 'solutions'."
At the time of this writing, final decisions had not been made on the issue of radioactive waste disposal at Rocky Flats. Preliminary discussions with DOE and the regulators indicated a willingness to examine the concept of retrievability. CAB favored this discussion and viewed it as a means to provide public health and safety benefits to the current generation, while not precluding advancements in technology that

may provide a more permanent or acceptable solution for future generations.

CONCLUSION

Effective citizen participation in cleanup decision-making at a major federal facility like Rocky Flats is very complex. In response to this complexity, the Rocky Flats Citizens Advisory Board undertook a proactive role in developing a foundation for discussing issues in a broad sense, or "big picture." Once developed this big picture would facilitate CAB's understanding of and ability to address more specific technical issues. From its experience in trying to provide a recommendation on the design of a landfill for the Solar Ponds area at Rocky Flats, CAB realized that a more fundamental issue needed to be addressed first. This issue was whether disposal of radioactive waste at Rocky Flats was acceptable, not only from a technical viewpoint, but from a social and moral perspective as well. Fully aware of the NIMBY (Not in My Backyard) criticism often leveled against citizen activists who oppose projects such as waste treatment or storage facilities, CAB developed a policy on waste disposal that opposed the concept of permanent placement of radioactive materials within a major metropolitan area, while leaving open the possibility of monitored and retrievable storage. Wastes may remain at Rocky Flats for an extended period of time, but during that time CAB anticipates technology development will continue with the hope that one day a more acceptable and lasting solution to the waste problem will be developed. CAB realized the utility of its decision to focus on the broader sense when it was asked to comment on the draft Conceptual Vision for Rocky Flats. The arguments had already been made, and the decisions and consensus views were in place. The draft Conceptual Vision called for waste disposal, an issue for which CAB had already developed a position. CAB anticipated that future versions of the Conceptual Vision would incorporate the concept of monitorable and retrievable storage rather than disposal. The reader will note that the full account of CAB's success in bringing its own "vision" for Rocky Flats to fruition is not yet complete. Final decisions on the future of Rocky Flats are not CAB's to make. It is CAB's sincere desire and hope, however, that both the Department of Energy and the regulators will realize the value of a consensus CAB recommendation and develop a final vision for Rocky Flats that meets both regulatory and community acceptance standards.

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FACILITATING THE TECHNICAL:
THE ROLE OF THE TECHNICAL ADVISOR IN A COMMUNITY ADVISORY BOARD

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ABSTRACT

A technical advisor for the Nevada Test Site (NTS) Community Advisory Board (CAB) is challenged to perform four key roles: 1) technical director, 2) administrator, 3) facilitator, and 4) spokesperson. Differences between roles performed by the technical advisor and the traditional facilitator to support an advisory board are described in this paper. Techniques used by the advisor to support the board include: summarize and simplify information; write and implement standard procedures; focus on program goals; and increase stakeholder knowledge and participation.

The primary difference between a technical advisor (TA) and a traditional facilitator is that a TA draws on scientific experience to guide them as the support base for the CAB mission and positions on issues. As an administrator, a TA utilizes scientific organizational skills and procedures to perform this role. The TA acts as a traditional facilitator during CAB monthly meetings between stakeholders and the Department of Energy. As a spokesperson for the CAB, a TA is different from a traditional facilitator because he represents a credible, technical perspective and resource on the issues and mission objectives of the board.

The technical advisor assists in the recommendation process by the simplification of information in documents and on issues that the CAB relies on to develop stakeholder input. Organizational procedures are developed and implemented by the TA to satisfy the need to have written steps guiding the way board activities are conducted. Board focus on broader though complex issues is supported by the TA emphasis on key issues requiring board review. The TA increases the technical base of stakeholder knowledge through opportunities to discuss EM programs and present board issues and mission objectives to stakeholders.

INTRODUCTION

The Community Advisory Board (CAB) for the Nevada Test Site (NTS) is a group of volunteers organized in 1994 to provide advice and recommendations to the Department of Energy (DOE) on the Environmental Restoration and Waste Management (ERWM) program. The CAB is consistent with DOE's "environmental vision," which includes the development of a strong partnership between the DOE and stakeholders. The board's establishment reflects growing opportunities for public participation in the site-specific advisory board (SSAB) process at local sites throughout the DOE complex to help deal with issues affecting local citizens.

In May of 1994 the Office of Management and Budget and the General Services Administration approved a charter that established an overall, national Environmental Management (EM) Site-Specific Advisory Board. This overall board provided authority for the creation of local site-specific boards throughout the DOE complex and it serves as the umbrella board chartered under the Federal Advisory Committee Act. Congress enacted this law in 1972 to provide standards and uniform procedures to govern the establishment, operation, administration and duration of any advisory committee.

NTS ERWM PROGRAMS

The Nevada Test Site is a unique national resource managed by the U.S. DOE Nevada Operations Office. The Site is located 65 miles northwest of Las Vegas. The 1,350 square mile facility is an area comprised of desert and mountainous terrain larger than the state of Rhode Island. The NTS was established under the Atomic Energy Commission, DOE's predecessor, and has seen more than four decades of nuclear weapons testing. In recent years the NTS has diversified into many other programs including hazardous chemical spills testing and cleanup, waste management, and development of new environmental technologies.

The Nevada Test Site Environmental Management (EM) program is part of DOE's 30-year program to address contamination from nuclear weapons programs at DOE facilities and sites. Nuclear testing and related support operations, nuclear rocket experiments, and non-nuclear experiments resulted in some contamination. Contaminants include radioactive materials, gasoline, oils, solvents, and heavy metals such as lead. EM

program objectives are to identify the nature of contamination, decide its potential risk to the public and to the environment, and to act to protect or restore the natural resources adversely affected by releases of hazardous materials.

The Waste Management (WM) program is responsible for the safe disposal of low-level radioactive and mixed waste from the NTS and other defense-related DOE and U.S. Department of Defense facilities. Waste is disposed of in shallow landfills and craters at two locations on the NTS. In addition, under an agreement with the state of Nevada, mixed transuranic waste is stored at the NTS until it can be shipped to the Waste Isolation Pilot Plant in New Mexico for disposal. Hazardous waste is accumulated at the NTS and shipped off site to an EPA-approved treatment, storage, and disposal facility.

CAB ORGANIZATION AND FOCUS

In the summer of 1993, an informal group of interested citizens, worked with DOE officials to develop a draft charter, mission statement, and bylaws. In October 1993, a public notice soliciting nominations for board membership was advertised in newspapers throughout the state. DOE forwarded the 120 applications to representatives of the state of Nevada's Environmental Protection Division (NDEP) for their review and recommendation. In February of 1994, the NDEP recommended a slate of 15 individuals to serve as initial members on the board. DOE accepted the slate of nominees and in June 1994, the Community Advisory Board for the Nevada Test Site Programs was officially approved by the Secretary of Energy.

As of December 1995, the board is comprised of 18 members representing a diversity of views. Current members include residents and workers near the NTS, environmental or public interest groups, labor and civic groups, Native American representatives, academia, and local governments. In addition to regular members, there are four non-voting ex officio members representing the DOE, the U.S. Defense Nuclear Agency, the state of Nevada, and a test site contractor organization. All board members serve a two-year term with at least one third retained for continuity and some membership terms may be staggered.

The board's charter calls for the formation of standing committees to handle specific issues and topics related to the EM program. Standing committees address areas of: budget and priorities; the NTS Environmental Impact Statement (EIS); waste transportation; future land use; issues; and administration. Standing committees are chaired by board members and open to participation from the DOE and stakeholders.

Some of the issues the CAB has reviewed include proposed plans for environmental restoration-generated waste shipments to the site, treatment plans for storage and disposal of mixed waste; funds generated from waste management; NTS EIS process; plans to manage DOE lands and facilities; equity issues relative to environmental restoration and funding; waste transportation; Draft Waste Management Programmatic EIS; and proposed Federal Facility Agreement and Consent Order outlining environmental restoration priorities for DOE sites in Nevada.

ROLE OF THE TECHNICAL ADVISOR

A technical advisor for the NTS CAB was hired through the Harry Reid Center for Environmental Studies (HRC) at the University of Nevada, Las Vegas (UNLV) under a grant provided by the DOE Nevada Operations Office in Las Vegas. The advisor's relationship with UNLV represents a credible, independent, and neutral position within the community. A university-

based advisor to support the CAB is unique among the DOE site specific advisory boards (SSAB) that are usually supported by federal contractors and a traditional facilitator. The TA is challenged to perform four key roles for the NTS CAB: 1) technical director, 2) administrator, 3) facilitator, and (4) spokesperson. In these roles the TA serves as the technical support base and primary resource for board activities, issues, and research.

Technical Director

In the role of technical director the advisor draws on scientific training to guide him as the technical support base for the CAB. A primary difference between the TA and the traditional facilitator or coordinator is the technical degree and training of the advisor. Many issues and documents of the DOE environmental management (EM) program are complex and not easily understood by board members and stakeholders. Technical persons like engineers and scientists are well suited for the study and interpretation of the complex.

A person with a degree in the technical sciences is likely to have some knowledge of or experience with federal environmental regulations and issues that would apply to cleanup activities at a DOE facility. His role is to readily develop a thorough understanding of EM activities and then consult the board in the review of issues on a regular basis. The science background helps him interpret and summarize technical issues and information into understandable and manageable pieces for the board.

Administrator

In the role of an administrator, a TA differs from the traditional facilitator because he utilizes scientific organizational skills and procedures to perform administrative duties for the board. The way a technical person executes administrative duties are likely to be based on scientific methodology for problem solving. Problem solving methods often follow a standard system of information gathering, data analysis, discussion, conclusion, and reporting. This scientific technique lends itself nicely to organizing activities and objectives on behalf of the board.

For example, board activities leading to a well organized, meeting agenda of presentation, discussion, and business are coordinated by the TA each month. He is also tasked with the execution of numerous administrative tasks like newsletters, reports to the DOE and stakeholders, management of CAB the library, attendance on behalf of the CAB at meetings and workshops, and coordination of the student intern program. Execution of administrative tasks is the most time consuming role for a technical advisor.

Facilitator

The technical advisor performs the role of the traditional facilitator during CAB monthly meetings between stakeholders and the DOE. Persons with scientific backgrounds are challenged to implement parliamentary procedures and facilitate a monthly meeting. Some technical persons may have educational training and public speaking skills through teaching science courses, however, they will typically have to learn the role of facilitator. A trained facilitator can be brought in as needed to help the TA learn and perform this role for the board.

Spokesperson

In the role of spokesperson, a TA functions as a technical consultant and clearing house of information on issues and mission objectives of the board. His full-time position requires him to be the point of contact

between members, the DOE, and the public on CAB activities. The TA is responsible for attending all meetings and workshops of interest to the CAB. He may be called upon at a meeting to present a summary of board activities related to a key issue. The TA also looks for opportunities to inform the public and other community organizations about DOE programs and board activities for stakeholder participation. Other SSAB representatives and even the media may contact the TA for information about a variety of topics on cross cutting issues and similar SSAB business.

METHODS TO FACILITATE THE TECHNICAL

Techniques used by the a TA to support the board include: summarize and simplify information; write and implement standard procedures; focus on program goals; and increase stakeholder knowledge and participation. The TA assists in the recommendation process by the simplification of information in documents and on issues that the CAB relies on to develop stakeholder input. Organizational procedures are developed and implemented by the TA to satisfy the need to have written steps guiding the way board activities are conducted. Board focus on broader though complex issues is supported by the TA emphasis on key issues requiring board review. The TA increases the technical base of stakeholder knowledge through opportunities to discuss EM programs and present board issues and mission objectives to stakeholders.

Summarize and Simplify Information

Board members frequently complain about the overwhelming reading requirements and complexity of the EM programs. Reports are often too numerous and difficult to adequately review and utilize by the average stakeholder. The science background of a TA helps him summarize technical documentation into manageable pieces more acceptable to the board. Important information is extracted and summarized in a memorandum, table, or brief report. For example, the TA summarized information from a future land use panel discussion into notebook form for each CAB member. He reviewed an EM technical strategy document and made a summary of comments in spreadsheet format for the board to use as a reference. The advisor also maintains a document library for stakeholder use.

Write and Implement Standard Procedures

An organization such as an SSAB evolves in time and becomes so active that some level of organization is required to enhance efficiency and utilize resources in a timely manner. The TA's experience of writing scientific papers, reports, and procedural manuals lends itself to the task of organizing information and describing steps in a process. He developed standard operational procedures (SOP) describing all board activities from first hand knowledge and similar information from other SSABs. The SOP describes in writing the history, mission, organization, and activities of the CAB in a single document. It also describes the duties for each part or person associated with the board to help clarify their responsibility in the overall mission objectives.

A description of board procedures helps to identify difficulties with some activities and solutions to overcome them. The SOP provides a format for the CAB and the DOE to work with on a systematic basis. The SOP helps ensure resources are allocated efficiently to plan and organize public participation opportunities in advance. Old and new board members utilize the SOP to understand CAB activities and interests. The SOP will be a resource document that continues to evolve with the CAB and is maintained by the TA with up-to-date information.

Focus on Program Goals

Board focus on broader though still complex issues is assisted by the TA who acts as a consultant to transfer information between the DOE and the CAB. The TA helps focus the activities between both parties toward common goals. With his scientific understanding of environmental processes and a increasing knowledge of site issues, a TA guides the focus of the board on issues where they will have the most impact. For example, the TA discussed with the DOE about which issues were most important for the NTS CAB focus: agreements between the state and DOE; budget; NTS EIS; and future land use. In conjunction with the board and stakeholders, an open discussion and ranking of issues for CAB focus in 1996 was conducted at a monthly meeting. From the ranking of issues for CAB focus in 1996, the monthly calendar and work plan are drafted for the board. The TA emphasizes the monthly meeting agenda to focus board review primarily on these key issues.

Increase Stakeholder Knowledge and Participation

The TA increases the technical base of stakeholder knowledge through opportunities to discuss EM programs and present board issues and mission objectives to stakeholders. The TA is an excellent position to market and speak on behalf of the board. He should be willing to speak at meetings of other community groups and other stakeholder events. Because of his technical background and affiliation with the university, the TA is able to respond readily to questions and issues under board review. For example, the TA has addressed the Nye County commissioners, a DOE declassification workshop, and a transportation workshop on behalf of the CAB. The TA also helped instruct a university class on risk assessment case study of an NTS plutonium-contaminated soil issue.

SUMMARY AND CONCLUSIONS

The TA for the NTS CAB is challenged to perform four key roles in support of the Board's mission: 1) technical director, 2) administrator, 3) facilitator, and 4) spokesperson. Differences between a TA and a traditional facilitator are based primarily on the scientific background of the advisor who uses a standardized system of problem solving to guide him in the performance of the four roles. The role of administrator demands more time from the TA than the other roles. The TA for the NTS CAB is unique among other SSABs because of his affiliation with the university which represents a credible, independent, and neutral position within the community. A technical person is also educated and experienced in the study of the complex, a skill required to comprehend and manage the large amount information pertaining to the environmental issues at a DOE facility. Four methods are used by the TA to facilitate the technical: 1) summarize and simplify information, 2) write and implement standard procedures, 3) focus on program goals, and 4) increase stakeholder knowledge and participation. These methods help manage information for the board, improve efficiency of activities, focus resources on priority issues, and enhance opportunities for stakeholder participation.

The selection of a technical person from a local university to provide major support to an SSAB is different and unique to most advisory boards. In comparison to SSAB support staff, a TA does have the advantage of being able to comprehend and organize the complex, although a disadvantage of a TA would be training him to facilitate monthly meeting discussions. All SSABs will have similar and different site issues that may or may not require the support of a TA. For the NTS CAB, the

selection of a technical advisor to support the board has helped improve the volunteer organization and focus activities toward greater stakeholder knowledge and participation in the EM advisory board process for the DOE.

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Session 43 -- WASTE MANAGEMENT AND ENVIRONMENTAL RESTORATION SUCCESS STORIES AT NTS

Co-chairs: Carl Gertz, USDOE;
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43-1

RADIOACTIVE WASTE ACCEPTANCE TEAM AND GENERATOR INTERFACE YIELDS SUCCESSFUL IMPLEMENTATION OF WASTE ACCEPTANCE CRITERIA*

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ABSTRACT

The Fernald Environmental Management Project has developed a successful Low Level Waste Shipping Program in compliance with the Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements, NVO-325, Revision 1. This shipping program is responsible for the successful disposal of more than 4 million cubic feet of Low Level Waste over the past decade.

The success of the Fernald Low Level Waste Shipping Program is due to the generator program staff working closely with the DOE-NV Radioactive Waste Acceptance Program Team to achieve win/win situations. The teamwork is the direct result of dedicated, proactive professionals working together toward a common objective: the safe disposition of low level radioactive waste.

The growth and development of this program has many lessons learned to share with the low level waste generating community. The recognition of reciprocal interests enables consistently high annual volumes of Fernald waste disposal at the Nevada Test Site without incident. The large volumes successfully disposed serve testimony to the success of the program which is equally important to all Nevada Test Site and Fernald stakeholders.

The Fernald approach to success is currently being shared with other low-level waste generators through DOE-NV sponsored outreach programs. This paper introduces examples of Fernald Environmental Restoration Management Corporation contributions to the DOE-NV Radioactive Waste Acceptance Program outreach initiatives. These practices are applicable to other low level waste disposal programs whether federal, commercial, domestic or international.

BACKGROUND

The Nevada Test Site encompasses 1,350 square miles of land located about 65 miles northwest of Las Vegas, Nevada. Since 1951, the Nevada Test Site has been the primary location for nuclear weapons testing in the United States. Between 1951 and 1992, the Nevada Test Site was the location for 925 nuclear weapons tests. Since the late 1960s, the Nevada Test Site has been actively cleaning up the contaminated areas resulting from nuclear weapons testing and support operations. While weapons testing remains the primary mission of the Nevada Test Site, the current treaty banning testing has altered the mission to maintaining readiness to resume testing. Expanded radioactive waste disposal is one proposed alternative use of the Nevada Test Site in support of the DOE weapons complex remediation efforts.

Initial disposal of radioactive waste began in the early 1970s. Since the mid 1970s, all radioactive waste disposal at the Nevada Test Site has been restricted to one of two designated waste management areas. The Area 3 radioactive waste disposal facility utilizes subsidence craters from underground nuclear weapons tests for waste disposition. The Area 5 radioactive waste disposal facility uses shallow excavated pits and trenches for waste disposal. The volume of waste accepted and disposed in these two areas is approximately 17 million cubic feet of low level waste.

Disposal of radioactive waste from off site locations began in 1976. Waste acceptance criteria were developed to ensure waste received at the Nevada Test Site was in a form that was protective of the workers and the environment. The first waste management criteria was developed in the late 1970s titled, "Operational Radioactive Defense Waste Management Plan for the Nevada Test Site," NVO-185. Between 1978 and 1985, NVO-185 was revised four times to incorporate criteria that were evolving with radioactive waste disposal experience and regulatory changes.

In 1988, DOE Order 5820.2A was issued resulting in the development of the "Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements," NVO-325. This document established waste acceptance criteria which incorporated the requirements of the new order and applicable RCRA regulations. The current waste acceptance criteria, "Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements," NVO-325, Revision 1 was issued in June 1992 to incorporate additional regulatory changes. The existing criteria were developed to provide increased oversight of radioactive waste disposal. All generators seeking approval for radioactive waste disposal at the Nevada Test Site must comply with the requirements of NVO-325, Revision 1.

The Fernald Environmental Management Project, formerly known as the Feed Materials Production Center, is located about 20 miles northwest of Cincinnati, Ohio. Uranium metal products for the nation's defense programs were produced at the facility between 1953 and 1989. During those years the facility produced slightly enriched or depleted products for use in production reactors to make plutonium and tritium at other DOE sites. Uranium production was suspended in July 1989 to focus on environmental restoration. The Fernald workforce has been dedicated entirely to environmental restoration since 1989. Also in 1989, the site was added to the Environmental Protection Agency's National Priority List of federal facilities in need of remediation. In 1991, the DOE officially announced that production at the facility was ended and the program

management responsibility within the DOE was transferred to the Office of Environmental Restoration and Waste Management. In February 1992, an Ohio Field Office was established to oversee all aspects of a full-scale environmental restoration and waste management effort which is expected to last several years. Fernald disposed of radioactive waste in on site disposal pits until 1985, when regulations pertaining to disposal cell design forced Fernald to suspend disposal of radioactive waste in the existing cells.

THE FERNALD DISPOSAL PROGRAM

Fernald initiated radioactive waste disposal at the Nevada Test Site primarily to support ongoing production and construction waste management requirements. The volumes of Fernald waste disposed at the Nevada Test Site increased steadily from 1985 until 1990 (Fig. 1). During this time, Fernald waste volumes increased to contribute fifty percent of the total annual volumes of waste received at the Nevada Test Site. The Fernald waste volumes combined with other waste disposal volumes to increase the Nevada Test Site disposal to more than one million cubic feet of off site waste in fiscal year 1989.

Fig. 1

In 1990, a suspension of waste receipts at the Nevada Test Site resulted from a Tiger Team investigation identifying concern for verification of waste acceptance criteria compliance. This suspension caused the withdraw of all existing disposal approvals for generators shipping waste to the Nevada Test Site. Suspensions remained in effect until generators demonstrated compliance with the current waste acceptance criteria. Recognizing the importance of the Nevada Test Site disposal option to the Fernald environmental mission, the facility waste management staff immediately initiated program improvements. Fernald was the first generator to successfully complete the redesigned approval process by demonstrating compliance to the waste acceptance criteria. Fernald resumed shipments in 1990 and remained the only approved facility shipping to the Nevada Test Site through fiscal year 1991.

As a result of the 1991 DOE announcement that production activities at Fernald was officially ended, Fernald concluded that waste volumes disposed at the Nevada Test Site would have to be increased to support the new environmental restoration mission. A backlog of material originally intended for recycle in the Fernald process accumulated on site between 1985 and 1991. A substantial portion of this material would consequently be designated as waste requiring disposal. The first priority of the environmental restoration project was to address this stockpile of legacy containers. Many of these containers were stored outside and were deteriorating. Escalating costs for maintaining safe storage and repackaging legacy material for off site disposal was the primary motivation for increasing alternative disposal options. With a clearly designated environmental restoration mission, Fernald commenced the DOE-NV approval process for additional waste streams.

The DOE announcement confirming Fernald as the first DOE facility to be in full environmental remediation coincided with increased DOE-NV compliance assessment. The criteria issued in NVO-325, Revision 1, June 1992, required generators to provide more compliance documentation than previously requested. The revised waste acceptance criteria increased the level of documentation required to demonstrate compliance. The process for generators seeking Nevada Test Site disposal approval became more

rigorous. Fernald was among the first generators to obtain application approval in compliance with the revised waste acceptance criteria.

The Fernald strategy for securing approval under the revised criteria was the same approach used in 1990. Fernald focused on the largest waste streams with the least impact resulting from the revised waste acceptance criteria. The first streams submitted were the bulk contaminated trash, maintenance, and construction rubble waste streams. These waste streams are primarily characterized by process knowledge which does not require sampling and analysis. With these approvals, Fernald was able to continue remediation of large piles of scrap metal and maintain disposal outlets for waste generated by ongoing construction at the facility. The effort directed toward gaining this approval enabled Fernald to continue shipments to the Nevada Test Site which in turn enabled the facility to maintain progress on the environmental remediation mission. Having established these waste approvals, Fernald then began to seek approval of the more difficult waste streams including uranium and thorium process residues. The approval process for these waste steams required further Fernald program development because this material was amenable to sampling. This aspect required more extensive compliance documentation. Fernald demonstrated compliance by developing sampling and analysis plans and procedures, laboratory acquisition protocols, and providing analytical results for DOE-NV review and approval. This process required nearly one year to complete during which time the waste streams characterized by process knowledge continued shipments to the Nevada Test Site. Fernald and DOE-NV also developed an innovative review process which provided for conditional approval of wastes amenable to sampling. The conditional approval was pending DOE-NV review of final data packages prior to approval for shipment to the Nevada Test Site. The interim approvals allowed Fernald to ship waste for which the sampling and analysis review was completed. This development allowed Fernald to make progress toward the remediation of the facility while the full approval process continued on course.

FERNALD INITIATIVES

The process of gaining approval for individual waste streams requires at a minimum three to six months of review and response cycles for Fernald with an established program. For new generators without an established program, the approval process may be considerably longer. To accelerate radioactive waste disposal, Fernald adopted a programmatic waste stream approach to streamline the approval process. Although this approach does not match the NVO-325, Revision 1 waste acceptance criteria expectations, Fernald successfully demonstrated how the approach met the NVO-325, Revision 1 objectives. The NVO-325, Revision 1 perspective that waste streams were discrete groupings of waste containers did not match the Fernald definition of waste stream. Fernald realized that the discrete waste stream approach would require many waste stream reviews and approvals. This would unnecessarily delay the Fernald mission and or increase disposal costs. To bridge the gap in waste stream definitions, Fernald established programs that provide sufficient process controls to demonstrate compliance with NVO-325 which affords Fernald more influence over the fate of their program. The programmatic approval is based on the assumption that following routine tasks, relatively similar waste can be certified in similar processes. The fundamental requirement of the programmatic approach is to design waste streams with broad descriptions and then manage individual waste container compliance with the waste

stream approval through the characterization process. The programmatic waste stream approach reduces the time and cost required to get DOE-NV waste stream approval each time a population of waste is characterized. As Fernald worked with DOE-NV to develop this approach, the benefits became clear. Now, DOE-NV suggests to other generators to consider this approach.

To support this programmatic approval process, Fernald developed a complimentary sampling and analysis program in compliance with NVO-325, Revision 1. The NVO-325, Rev 1 requirement to review individual Sampling and Analysis Plans is minimized by a programmatic approach to waste characterization. The Fernald program provides consistency of documentation which aides the review and approval process. The consistency of characterization packages allows DOE-NV to select a representative number of sampling plans for compliance reviews. The reviews provide the level of confidence needed to overview the entire sampling and analysis program. This programmatic approach reduced the time and cost of DOE-NV compliance reviews and reduces the effort required to secure frequent individual waste stream approvals.

By reducing the approval efforts, Fernald realized that the volume of waste being approved for disposal could potentially exceed the capacity of the existing workforce and facilities. In anticipation of this impact, Fernald solicited vendor supplied waste processing services. This initiative forced Fernald and DOE-NV to think "outside of their box" and consider more than one way to manage NVO-325, Revision 1 compliance. Fernald secured DOE-NV approval for two remote vendor services contracts which enabled shipments of processed waste directly from the vendors facility to the Nevada Test Site. This was the first approval of this type ever awarded by the DOE-NV Radioactive Waste Acceptance Program team. This initiative allowed Fernald to increase waste disposal remediation by utilizing existing technology without the procurement, construction, and start-up cost of a DOE owned process. This approval required Fernald to establish a remote waste certification program at the vendors facility. Having developed this program, Fernald assumed a more independent oversight responsibility for the vendors waste management activities. This intermediate position resulted in Fernald gaining elevated awareness of their ownership in the DOE-NV Radioactive Waste Acceptance Program.

Successfully developing programmatic approvals and vendor supplied services approvals requires that the generator understands the DOE-NV program. Fernald accomplishes this familiarization by maintaining close contact with the assigned Radioactive Waste Acceptance Program representative. Fernald avoids delays and interpretation differences through open conversation. This communication minimizes the surprises and keeps all participants at the same level of understanding.

Despite the close contact and the effective program established by Fernald, some compliance discrepancies have occurred. Fernald has disposed of more than 26,600 containers requiring 3,817 shipments to the Nevada Test Site over the past decade. Non-conformances have been recorded for less than 1% of all shipments during this time. Over the past five years, Fernald has reduced the non conformance rate to than 0.5% of the shipments. This improvement is the result of maintaining a program in compliance with the DOE/NV waste acceptance criteria. This improvement is significant when you consider that the number of shipments and containers per shipment have increased during this time. Never-the-

less, despite doing the job right, non-conformances do occur. The nature of the business demands that mistakes be kept to a minimum. When non-conformances are discovered, doing the job right means accepting the responsibility of effecting corrective actions. To maintain low non-conformance rates, Fernald conducts self assessments and acts decisively when non-conforming conditions are identified. Corrective actions are completed quickly to restore compliance, improve the process, and prevent recurrence. Fernald sometimes utilizes self imposed shipment suspensions to motivate corrective action implementation. This strategy minimizes delays to the Fernald program while elevating DOE/NV confidence in the program. This dedication to program improvement is fundamental to maintaining a successful program.

INTERFACE INTANGIBLES

In addition to specific program developments, Fernald also contributes to the DOE-NV Radioactive Waste Acceptance Program development in other ways. Fernald willingly participates in information exchange meetings with other Nevada Test Site waste generators. An example is Fernald presentations of lessons learned materials at DOE-NV Waste Generator Workshops. Or when appropriate, DOE-NV refers potential generators to contact Fernald directly to evaluate this program as an example for developing their programs. Sharing information assists with developing a level of consistency across the complex which improves the generator review and approval process. This consistency reduces the time and cost of generator program development and DOE-NV approval reviews as compliance becomes more routine rather than trial and error.

Fernald further supports program consistency through generator supported audits. As a result of the limited Radioactive Waste Acceptance Program Team resources, other generators expressed concerns that Fernald was monopolizing the team time. This was perceived to be affecting other program reviews. In response, Fernald proposed and supports the DOE-NV audit team by supplementing the team with FERMCO staff. This initiative yields many dividends. One benefit is generator support makes it possible for DOE-NV to maintain the size of the audit staff while freeing permanent team personnel to address other duties. Audit team participation also enables generator personnel to visit other sites and review their programs and establish working relationships with their counterparts. Audit team participation also enables generators an opportunity to evaluate program compliance with waste acceptance criteria from the DOE-NV perspective. This interaction contributes to developing consistency of criteria compliance among the generators which reduces the time and effort required to review and approve individual programs. This program continues even after the Radioactive Waste Acceptance Program Team has been expanded because of the benefits derived from the experience.

Additional generator staff interaction is achieved through work group participation. Fernald supports generator work groups which allows generators an opportunity to provide input to Radioactive Waste Acceptance Program development. Work groups promote communication among generators and encourages an exchange of information and methodology. Fernald supports DOE-NV by participation in several work groups. The first work group supported by Fernald was tasked with RCRA characterization data package development. The work group produced a data package format guidance which has been incorporated into the Radioactive Waste Acceptance Program approval process. One recently concluded work

group activity supported by Fernald involved development of a sealed source disposal position paper. Longer term work group activities currently in progress is the NVO-325, Revision 2 rewrite and standardized waste disposal containers. Fernald continues to support work group opportunities evaluating common issues and recommending solutions to DOE-NV. When appropriate, DOE-NV supports work group deliverable and elevates proposals to higher approval authorities for concurrence. Accepted positions become available to all Nevada Test Site generators for application in their programs.

Supporting Radioactive Waste Acceptance Program and other Nevada Test Site generators is only one piece of the success formula. Keeping the stakeholders informed and addressing their concerns is equally important to the overall success of every DOE facility. Fernald has a successful community relations program with the Fernald stakeholders. The lessons learned from the Fernald facility experience demonstrates the importance of good community awareness. Fernald promotes stakeholder interaction and lends experience and support to the Nevada Test Site stakeholders programs. Fernald supports the DOE-NV stakeholder interests by attending and providing input to Las Vegas area community meetings. Fernald often sends representatives to community meetings to represent the facility and provide prompt responses to Fernald related issues. An example of the value of this service was demonstrated in 1994, when Fernald recorded the first significant shipping accident. A Fernald representative was in attendance at the Nevada Test Site Citizens Advisory Board meeting one week after the incident providing details of the accident and recovery operation. Two weeks after the event, the same representative attended a Nevada Test Site Citizens Advisory Board meeting with photographs and updated information. One month after the incident, a second individual attended a Citizens Advisory Board meeting related to transportation safety to provide first hand accounts of the accident recovery operation. Each meeting attended allowed Fernald the opportunity to personally present the Fernald side of issues and proactively address Nevada resident's concerns for LLW shipment and disposal safety. The result is an informed and accommodating attitude by Nevada Test Site stakeholders toward disposal of Fernald waste.

In addition to attending citizens meetings, Fernald supports DOE-NV studies intended to address other community concerns. Fernald voluntary provides data in support of Nevada Test Site issues such as the Environmental Impact Statement, Transportation Safety Studies, and the Performance Assessment for the disposal facilities. Fernald representatives often review Nevada Test Site documents and provide comments to assist development and implementation of new program policies.

CONCLUSION

As radioactive waste disposal costs continue to increase, maintaining efficient disposal programs becomes more important. The Fernald Environmental Management Project Mission statement... "Together DOE and FERMCO are committed to the safe, least cost, earliest, final cleanup of the Fernald Site, with in applicable DOE Orders, regulations, and commitments in a manner which addresses stakeholder concerns"

...requires that Fernald support the DOE-NV Radioactive Waste Acceptance Program if the mission goals are to be achieved. The Fernald radioactive waste disposal program consistently achieves goals established to demonstrate significant progress toward restoration of the Fernald site.

Fernald realized early in the remediation effort that success can influence future funding considerations. In recognition of this relationship, Fernald dedicates the support required to maintain an effective disposal program with the Nevada Test Site.

Fernald realizes that the Fernald mission is related to the Nevada Test Site mission and that the Nevada Test Site must remain open to off site disposal if the Fernald mission is to succeed. To achieve this objective, Fernald supports the Nevada Test Site program development and maintenance as necessary to address stakeholders concerns. Meeting local community expectations is critical to the success of both sites. Fernald accepts the obligation to do the job right and is dedicated to achieving program results that support this commitment.

Accepting a share of the responsibility for the Nevada Test Site radioactive waste disposal program success cultivates a sense of program ownership at Fernald. This ownership is the result of the recognition that the two sites destinies are closely related. Ownership motivates Fernald to maintaining a high level of performance that is in compliance with the Nevada Test Site waste acceptance criteria. Through consistent compliance with the waste acceptance criteria, a level of confidence and respect develops. The Fernald program maintains a mutually supportive relationship with DOE-NV. This relationship is difficult to establish and even more difficult to sustain. To avoid damaging this relationship, both programs must remain flexible and willing to evolve with changing conditions. Fernald dedicates considerable effort to maintain compliance with NVO-325, Revision 1 and avoid program conflicts.

Establishing effective customer relations is the primary message to be shared by promoting the Fernald program. Success is achieved through learning your customer expectations and then working to meet them. Provided your customer has an equal understanding of your expectations, developing a successful working relationship is natural. Fernald and the DOE-NV Radioactive Waste Acceptance Program Team has developed this relationship. By sharing this information, the message bridges facility and program boundaries. The key point being that by utilizing an innovative mix of dedication and interpersonal skills, Fernald and DOE-NV have implemented a mutually supportive, ergo successful, waste acceptance program in compliance with NVO-325, Revision 1 waste acceptance criteria.

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RISK BASED CONCENTRATION LIMITS FOR DISPOSAL OF TRANSURANIC RADIONUCLIDES AT THE NEVADA TEST SITE

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ABSTRACT

The Area 5 Radioactive Waste Management Site (RWMS) is a Department of Energy (DOE) operated low-level waste disposal site on the Nevada Test Site (NTS). The DOE requires site operators to set concentration limits for disposal of low-level waste based on the results of site-specific performance assessments. Most site operators have followed a procedure similar to that used by the Nuclear Regulatory Commission (NRC) to set generic concentration limits for facilities licensed under 10 CFR 61. A typical approach consists of developing a set of scenarios and

determining the relationship between dose and waste concentration for each scenario. The concentration limits are selected as the lowest derived concentrations that meet the performance objectives. A waste classification system can be developed based on the application of different scenarios to different waste forms or disposal options.

A performance assessment that evaluates five exposure scenarios has been prepared for the Area 5 RWMS. The intruder scenarios were the limiting scenarios for most nuclides, including all the transuranics. The intruder scenarios can be used to support two classes of low-level waste. The most limiting scenario was the intruder-agriculture scenario. This scenario can be used to support a waste class allowing very near surface disposal of unstablized wastes containing transuranic radionuclides up to an activity concentration of 1 105 Bq kg-1 (3 nCi g-1). Higher concentration limits can be justified by taking credit for greater periods of radioactive decay, waste stability or facility design features. For waste disposed below the depth of common construction excavations, concentration limits can be set based on the results of a post-drilling intruder scenario. Using this scenario, concentration limits up to 3.7 105 Bq kg-1 (100 nCi g-1) can be derived for most transuranic radionuclides.

INTRODUCTION

Safe disposal of low-level radioactive waste has been assured in the U.S. by controlling the location of disposal site, the activity concentration of the waste and the waste stability. Transuranic radionuclides have always been a concern because of their long half-lives and high dose conversion factors. Prior to issuance of 10 CFR 61, disposal of transuranic-bearing waste in the near surface was commonly limited to a concentration of 3.7 104 Bq kg-1 (10 nCi g-1), based on the comparative risks of 226Ra in soil (1). With the issuance of 10 CFR 61, the NRC and Agreement States set generic limits for the concentration of transuranic radionuclides in low-level wastes that apply to all disposal sites. For transuranic radionuclides, the limits are 3.7 104 Bq kg-1 (10 nCi g-1) for Class A wastes and 3.7 105 Bq kg-1 (100 nCi g-1) for Class C waste. In contrast, the DOE has allowed disposal site operators to set concentration limits based on the results of site-specific performance assessments. Consequently, various concentration limits are in use at DOE facilities. No DOE sites accept low-level waste with transuranic concentrations exceeding 3.7 105 Bq kg-1 (100 nCi g-1).

WASTE CONCENTRATION LIMITS AND WASTE CLASSIFICATION SYSTEMS

The NRC concentration limits and waste classification system are based on the results of an analysis of the dose received by an inadvertent intruder (2). The inadvertent intruder is defined as an individual that enters the site 100 to 500 years after closure and is unaware that buried radioactive material is present. Most methods of analysis assume that the doses received by an intruder are directly proportional to waste concentration. The NRC did not consider the exposure of members of the general public through pathways such as groundwater when developing the concentration limits. This issue was left open as a site-specific analysis.

The NRC evaluated three intruder scenarios to set the concentration limits. In these scenarios, the intruder is assumed to construct a residence with a basement and reside permanently on the contaminated site. During excavation of the basement, the intruder is assumed to exhume and contact buried waste. The intruder-construction scenario is a

short-term scenario describing the exposure of an intruder during excavation of a basement and construction of a house. The intruder-discovery scenario is a shortened version of the intruder-construction scenario that assumes that the intruder identifies the hazardous nature of the waste and the exposure stops after discovery. The intruder-agriculture scenario covers the chronic exposure that occurs after construction of the house. During excavation of the basement, buried waste is mixed into the excavated soil. The waste is assumed to be indistinguishable from soil. The soil-waste mixture is used as backfill around the foundation and distributed around the residence. The intruder resides continuously on the contaminated site. The intruder is exposed by external irradiation, inhalation and ingestion of food grown in contaminated soil.

The NRC Class A concentration limit is the lowest derived concentration that yields a dose equivalent of 0.005 Sv yr-1 (500 mrem yr-1) in the intruder-construction or intruder-agriculture scenario for an intrusion event occurring at 100 years after closure (1). The intruder-agriculture scenario is limiting for most radionuclides (3). Although this waste class has no explicit depth of burial requirement, the analysis was performed with a 2 m cover (2). Class A waste carries no stabilization requirements. The 3.7 104 Bq kg-1 (10 nCi g-1) limit adopted by NRC was based on the observation that the derived limits for several important transuranic radionuclides were near this concentration and that it had been demonstrated that industry could segregate and manage waste at this concentration level (4).

Higher concentrations could be allowed if facility design could be modified to reduce the potential for intrusion, delay its occurrence or mitigate its impacts. NRC assumed that this could be accomplished by increasing cover thickness, by layering waste, by using engineered barriers or by stabilizing the waste (1). Increasing cover thickness reduces the potential for an intruder to contact waste while digging a construction excavation. Placing lower activity concentration waste on top of high activity concentration waste increases the likelihood that an intruder will recognize that waste is present and stop digging. Engineered barriers perform a similar function of warning the intruder that something unexpected is present. Engineered barriers are useful when shallow groundwater does not allow a greater depth of burial. Waste stabilization was expected to affect site performance and the consequences of inadvertent intrusion. Stabilization was assumed to be beneficial by maintaining cap integrity, thereby minimizing the potential for migration of contaminants out of the unit. Stabilization also reduces the intruder's ability to disperse exhumed waste and makes it more likely that the waste will be recognizable for a longer time.

For the transuranic radionuclides, it was realized that intruder barriers might not remain effective as long as the waste remained hazardous. The NRC noted that not all waste would contained long-lived radionuclides, and that, over time, the average concentration of waste in the disposal unit would decrease with radioactive decay. It was proposed that radioactive decay would allow disposal of individual packages exceeding the concentration limit, since the decay of short-lived waste in surrounding packages would dilute the average concentration in the disposal unit over time. The NRC eventually settled on allowing a factor of 10 higher concentration for Class C transuranic waste, based on the assumption that dilution by lower activity waste would reduce the average

concentration to acceptable levels (4). Therefore, the Class C concentration limit for transuranics was set at 3.7 105 Bq kg-1 (100 nCi g-1). All Class C waste requires a depth of burial of at least 5 m and stabilization. Both requirements are intended to reduce the potential for intrusion and its impacts.

With the issuance of DOE Order 5820.2A (5), DOE allowed site operators to set concentration limits based on the results of site-specific performance assessments. As guidance, DOE issued an example of an approach for developing site-specific waste concentration limits (3). The approach was similar to the NRC approach. A notable addition was that new scenarios were developed to account for site-specific disposal practices. These scenarios, called the drilling and post-drilling scenarios, were developed as intruder scenarios applicable to waste disposed of at depths greater than 5 m. The scenarios assume that a water well is drilled through the waste disposal unit and the drill cuttings are dispersed over the ground around the well head. The drilling scenario is a short-term scenario dealing with the exposure of the drillers. The post-drilling scenario is a chronic exposure scenario dealing with a resident that lives within the area contaminated with drill cuttings. In addition to considering intruder scenarios, a DOE site operator must also consider exposure of the general public when setting waste concentration limits. Doses from the groundwater pathway are usually the most important exposure route for the general public.

One notable difference between concentration limits derived for DOE and NRC facilities is that the limits are based on different times of compliance, dose limits and dose conversion factors. The NRC limits its intruder analyses to a period from 100 to 500 years after disposal, whereas DOE analyses typically evaluate an interval out to 10,000 years. The DOE uses the International Commission on Radiation Protection (ICRP) recommendations in Publications 26 and 30 (6,7). With adoption of the new recommendations, the DOE dose limit for the intruder was set at 0.001 Sv yr-1 (100 mrem yr-1) rather than the 0.005 Sv yr-1 (500 mrem yr-1) used by NRC. The dose conversion factors used by each Agency were developed using different methodologies and some differences exist.

DEVELOPMENT OF NTS WASTE CONCENTRATION LIMITS

The Area 5 RWMS is a shallow land burial waste disposal facility for DOE generated low-level radioactive waste. The Area 5 RWMS has disposed of NTS-generated waste since 1961 and DOE low-level waste generated nation-wide since 1978. A performance assessment has recently been prepared for the Area 5 RWMS (8). It evaluates two scenarios for exposure of the general public and three intruder scenarios. Results from these analyses can be used to develop site-specific waste concentration limits.

Site-specific issues affecting the derivation of waste concentration limits may include waste form, disposal facility design and environmental conditions. At the Area 5 RWMS, environmental conditions are probably the most important site specific issue. Waste forms received at the Area 5 RWMS are usually not stabilized. The Area 5 RWMS performance assessment takes no credit for waste stability and assumes that the waste is immediately available for transport. In addition, the facility uses no liners or engineered barriers. Waste is landfilled in shallow unlined trenches and covered with 2.4 m of native alluvium. A closure cap will be placed on top of the 2.4 m alluvium cover.

Many environmental conditions at the Area 5 RWMS are advantageous for waste disposal. Located in an area that is transitional between the

Mohave and the Great Basin Deserts, the site receives little rainfall and is subject to high evapotranspiration. Potential evapotranspiration is estimated to be 14 times greater than the annual precipitation of 12 cm (8). Under the current climatic conditions, downward flow of infiltrating water through the waste disposal cells to the aquifer is not believed to occur. In the unlikely event that recharge were to occur, the thick vadose zone (235 m) and low water contents would lead to unretarded solute travel times approaching 1 106 years (9).

The arid nature of the site also reduces the potential land uses. Permanent surface water does not occur near the site. Runoff occasionally occurs in ephemeral stream channels after intense or prolonged rainfall. The infertile soils, arid climate and extreme temperatures limit the potential agricultural uses of the site. The great depth to groundwater, 240 m, makes irrigation economically unfeasible. The difficulty of obtaining water also reduces the potential for residential or industrial development. There are no known mineral or petroleum resources near the site. Sites with similar resources in the region remain mostly undeveloped.

Scenarios for exposure of the general public were selected based on the most common land use patterns observed at similar locations in southern Nevada. Two scenarios were identified, a transient visitor scenario and an off-site ranch scenario. The transient visitor scenario assumes that the general public is exposed while visiting or occupying the site on a temporary basis. Exposure occurs through external irradiation and inhalation of suspended soil. The off-site ranch scenario assumes that a ranch has been established at the closest site with surface water. Range-fed cattle from the ranch are assumed to have access to the site. Residents at the ranch are exposed through contaminated soil transported from the site to the ranch by atmospheric dispersion and by ingestion of beef and dairy products from cattle grazing at the site. All scenarios for exposure of the general public were coupled to a release model for the intact disposal site. This model estimates the release of contamination to surface soils based on uptake by the native flora and the burrowing activities of invertebrates.

A different approach was taken for development of intruder scenarios. DOE guidance is to develop site-specific intruder scenarios (10). However, developing scenarios for low probability events, such as intrusion, based human behavior 10,000 years into the future is probably beyond what can be defended by science (11). The common land uses current observed do not include activities leading to inadvertent intrusion. Therefore, the approach used to develop scenarios for the public, selecting of the most common land uses observed, leads to the conclusion that no intruder scenarios are appropriate for analysis. Rather than attempt to develop and defend site-specific intruder scenarios, the selected scenarios were those widely used in performance assessments (1,2,3). These scenarios were assumed to be hypothetical events analyzed to set conservative waste concentration limits. The scenarios considered were the intruder-construction, intruder-discovery and intruder-agriculture scenarios used by the NRC (1,2) to develop 10 CFR 61 and the drilling and post-drilling scenarios proposed by Kennedy and Peloquin (3) for DOE facilities using deep burial. The scenarios were made site-specific by eliminating any features that were physically impossible. Physically impossible features such as aquatic food pathways or groundwater pathways were eliminated. Models describing the remaining features were parameterized with best

estimate values for a Mohave Desert site. Two scenarios, the intruder-agriculture and the post-drilling scenario were found to be limiting. The intruder-agriculture scenario was limiting for wastes disposed of in the near surface. The post-drilling scenario was limiting for waste disposed of below the depth of common construction excavations.

Waste Concentration Limits

The results from each scenario can be used to set concentration limits. The waste concentration limit is the lowest concentration at the time of disposal that causes one of the performance objectives to just be exceeded. A waste classification system can be developed from the analysis of scenarios with different waste forms or disposal methods. All transuranic radionuclide concentrations were limited by the intruder-agriculture scenario or the post-drilling scenario, depending on the depth of burial.

The analysis result required to set the concentration limit is the scenario dose conversion factor. The scenario dose conversion factor is the total effective dose equivalent at the time of intrusion divided by the activity concentration of the radionuclide at the time of intrusion. The scenario dose conversion factor includes the dose from short-lived progeny than can be assumed to be in equilibrium before the end of institutional control. The concentration limit should also include the dose from any progeny produced after disposal that can not be assumed to be in equilibrium. This requires defining a decay factor that is the maximum activity concentration of the progeny within the compliance interval resulting from a unit activity concentration of the first member of the chain. The decay factor accounts for the effects of radioactive decay and ingrowth only. After (12), the decay factor for the jth member of the chain, DFj, is given by

Eq. 1

where li is the decay constant of the ith radionuclide, t is the time and where

Eq. 2

and

Eq. 3

The time, t, is the time within the compliance interval, 100 to 10,000 years, when DFj reaches a maximum.

An expression for the concentration limit that includes the dose from all progeny that are produced after disposal can be written. The concentration limit is

Eq. 4

where CL,i is the concentration limit of nuclide i (Bq m-3), HL the dose limit (Sv yr-1), DCFj the scenario dose conversion factor for nuclide j (Sv m3 Bq -1 yr-1) and DFj the decay factor of radionuclide j (dimensionless).

The scenario dose conversion factors for transuranic radionuclides range over three orders of magnitude (Table I). Neptunium-237 had the highest factor because of its high plant-soil concentration factor and relatively high ingestion dose conversion factor. The lowest factor was predicted for 241Pu because of its limited bioavailability and relatively low internal dose conversion factor. Many of the factors are similar because of the dominance of the inhalation pathway and similar inhalation dose conversion factors. The intruder-agriculture scenario yields the highest factors, approximately two hundred times greater than the post-drilling

scenario. This is mostly attributable to the higher surface soil concentrations estimated for the intruder-agriculture scenario.

Table I

Concentration limits derived from analysis of an intruder-agriculture scenario are compared in Table II. The NRC limits were developed by analysis of an intruder-agriculture scenario at 100 years using a dose equivalent limit of 0.005 Sv yr-1 (500 mrem yr-1) and NRC dose conversion factors (1,13). The waste concentration limits of Kennedy and Farris (14) were developed as a draft DOE waste classification system. These concentration limits are based on the NRC methodology and parameters, but use a dose limit of 0.001 Sv yr-1 (100 mrem yr-1) and DOE dose conversion factors based on ICRP methodology (6,7). The Kennedy and Ferris analysis and the NTS analysis for Area 5 are similar and the concentration limits are similar for most nuclides. The Kennedy and Farris limits are derived for intrusion at 100 years and the NTS limits are derived for intrusion at the time within 200 to 10,000 years when the maximum dose will occur. The different times of compliance partially explain the large differences for the short-lived nuclides 244Cm (t = 18.1 yr) and 241Pu (t = 14.3 yr). The differences between the Area 5 RWMS and the NRC intruder-agriculture limits are largely explained by the factor of five difference in the dose limits, with the exception of 237Np.

Table II

The upper limit of transuranic waste concentrations is compared in Table III. The upper limit for NRC-licensed facilities is the Class C limit, 3.7 105 Bq kg-1 (100 nCi g-1). In Table III, the NRC limit in units of activity per unit mass have been converted to a range of values in units of Bq m-3 by assuming that waste stream densities can vary between 1.0 and 2.3 g cm-3. The NRC Class C limit was developed for an intruder-agriculture scenario and assumed intrusion into a stabilized waste form at 500 years. The concentration limits were subsequently increased by a factor of ten to obtain the final limit (4). For transuranic radonuclides, this factor of ten increase accounts for dilution with lower activity concentration waste streams. The Kennedy and Farris limits were derived by the same method used by the NRC, but again use the lower dose equivalent limit and different dose conversion factors. The NTS Area 5 RWMS limits are derived from analysis of a post-drilling scenario occurring between 200 and 10,000 years. The NRC Class C limits and Kennedy and Farris limits are very similar since the same method was used. The post-drilling scenario used at NTS supports disposal of higher activity concentrations of transuranics radionuclides. With the exception of 237Np, the post-drilling scenario supports disposal of transuranic radionuclides up to the 3.7 105 Bq kg-1 (100 nCi g-1).

WASTE FORM AND DISPOSAL CONFIGURATION

In addition to the waste concentration limit, a waste classification system should stipulate the waste form and waste disposal configuration. These should match the assumptions made in the analysis. The concentration limits derived for the Area 5 RWMS assume no special waste form. Considering factors relevant to intruder scenarios only, these waste concentration limits are potentially for unstabilized waste classes. However, waste stabilization may have benefits that are not apparent in an intruder analysis. Waste stabilization is expected to enhance performance by reducing infiltration and by reducing the potential for dispersion after intrusion. The NRC considered stabilization to be necessary because caps engineered to reduce

infiltration will become less effective if subsidence of unstabilized waste causes disruption of the cap's impermeable layers (1). However, at an arid site such as the NTS, the natural hydrologic conditions may be sufficient minimize infiltration. Evapotranspiration is so much greater than precipitation that impermeable or low permeability layers are thought to be unnecessary. Therefore, stabilization to prevent subsidence may be unnecessary if maintaining the integrity of impermeable layers in the only criterion. Large scale subsidence, however, may allow the collection of run-on and ponding of water. This may allow infiltration to significant depths even at an arid desert site. Studies are planned to evaluate this potential issue. Stabilization may still be desirable to reduce the dispersability of the waste in the near term. This is not as important for the transuranic radionuclides because the half-lives are usually longer than the life time of the waste form. Therefore, the importance of stabilization for transuranic waste streams remains uncertain until hydrologic modeling of run-on into a subsided cap is completed.

Two classes of waste are being considered for the Area 5 RWMS. Class I waste is based on the intruder-agriculture limits in Table II. The NTS Class I limits are applicable to the disposal of unstabilized waste below a depth of 2.4 m in a trench of any thickness. The Class II concentration limits are the lesser of the post-drilling limits in Table III or 3.7 105 Bq kg-1 (100 nCi g-1). The NTS Class II waste concentration limits are applicable to the disposal of wastes in a 5 m thick layer at least 4 m below the surface.

The outcome of hydrologic studies of infiltration in subsided trenches will determine if the two waste classes require stabilization. If both waste classes have the same stabilization requirements, then they could be layered in the same trench. For a typical Area 5 RWMS disposal cell, Class I waste could be disposed of at any level in the trench and Class II waste would have to be restricted to the lower 2 m of the trench.

Table III

CONCLUSION

Waste concentration limits and a waste classification system has been proposed for the Area 5 RWMS. The derived limits are based on intruder scenario analyses as widely practiced in low-level waste management. These analyses assure that transuranic waste containing up to 3.7 105 Bq kg-1 (100 nCi g-1) can be disposed of in the near surface of the Area 5 RWMS and that the performance objectives of DOE Order 5820.2A will be meet.

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THE NEVADA TEST SITE (NTS) AS A WASTE DISPOSAL SITE: A HISTORICAL PERSPECTIVE

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ABSTRACT

The NTS is the location of two low-level waste (LLW) disposal sites which serve both the NTS and the U.S. Department of Energy (DOE) complex. The Area 5 Radioactive Waste Management Site (RWMS) is the location of a standard shallow land disposal operation, where packaged containers of waste are stacked neatly in excavated pits and trenches. LLW, mixed waste (MW), transuranic (TRU) waste, and classified LLW are handled at Area 5. The Area 3 RWMS is the location of the atypical disposal facility, where underground nuclear test subsidence craters are used for the disposal of containerized bulk waste. These two waste sites have received over 500,000 m3 (17,000,000 ft3) of LLW since 1978, when a managed waste program was instituted.

Annual waste volumes have fluctuated throughout the 18 year history from 9,600 m3 (340,000 ft3) to 81,400 m3 (2,875,000 ft3). In FY 1995, a total of 25,063 m3 (885,000 ft3) of LLW (917 shipments) was received. This accounted for approximately 85 percent of all waste that was shipped in FY 1995 from all the DOE facilities disposing waste offsite. To accommodate future waste, an additional new cell was excavated in FY 1995. This pit measures 305 m (1,000 ft) long, 46 m (150 ft) wide, and 6.7 m (22 ft) deep, having a disposal capacity of approximately 70,800 m3 (2,500,000 ft3). The cost of excavation was approximately $445,000. This paper presents a review of the history and evolution of the NTS waste sites as major contributors for handling between 30 and 40 percent of the DOE complex LLW.

BACKGROUND AND INTRODUCTION

In 1978, DOE/Nevada Operations Office (NV) designated two areas on the NTS as RWMSs: the Area 5 RWMS and the Area 3 RWMS. The Area 5 RWMS is located on an alluvial fan within a dedicated 296 hectare (732 acre) site in the southeast section of the 3,500 km2 (1350 mi2) NTS. The disposal cells are standard, excavated, shallow-land burial pits and trenches. The Area 3 RWMS, situated approximately 24 km (15 mi) north of the Area 5 RWMS, utilizes subsidence craters which were the result of underground nuclear tests. Adjacent craters are combined by excavating out a common wall to form the cell. The structure is then completed by shaping the cell, configuring access roads into the craters, and maintaining a level stacking surface. Fill material is obtained from the area between the nested pair. Currently, seven craters over a 20 hectare (50 acre) area have been dedicated for use as disposal cells in the Area 3 RWMS.

The two NTS sites are situated in remote locations within two different closed basins, have deep groundwater tables (thick unsaturated zones), and experience an arid climate. The Area 5 RWMS is located 238 m (780 ft) above the water table; Area 3 is approximately 488 m (1600 ft) above the water table. The locations receive between four and six inches of precipitation a year. Recharge to the aquifer has been found to be virtually nonexistent (1). Waste practices have also been improved from those of earlier waste disposal sites. For example, 2.4 m (8 ft) of soil

cover is emplaced over the waste in active disposal cells to bring it to 1.2 m (4 ft) above natural grade, as compared to the typical standard 1.8 m (6 ft) cover. This additional soil cover keeps moisture from reaching the waste. Other similar management improvements have been made to ensure the maximization of safety to both the employee, as well as the general public.

Assistance is provided to Area 3 and Area 5 RWMS operations personnel by a technical support team who ensure Federal, state, and DOE regulatory requirements are met. This support includes planning, budgeting, engineering, technical support, reporting, and computer assistance. Technical support includes maintaining the Safety Analysis Report, performance assessments, Part B permitting, waste management plan, program management plans, budget documents, design reports, etc. A key support effort is provided by the DOE/Radioactive Waste Acceptance Program team who review waste certification programs of NTS generators. Waste acceptance criteria for the site are outlined in the "Nevada Test Site Defense Waste Acceptance Criteria (WAC), Certification, and Transfer Requirements," NVO-325, Rev. 1, June 1992. This document outlines the specific criteria and application process for disposal of waste at the NTS. Approval to dispose of waste at the NTS is granted only after the waste has been certified as meeting the WAC. In FY 1995, there were 14 offsite waste generators and two onsite generators disposing over 50 approved waste streams.

The operations unit is comprised of approximately 36 professional, clerical, radiological, and craft personnel. This staff is responsible for waste receipt, inspection of waste packages, reviewing shipping papers, off-loading trucks, stacking waste, recording locations using barcode readers, and covering the waste. Additional duties include preparation of radioactive waste permits, billing, and record keeping. Personnel also maintain the sites, excavate new pits and trenches, maintain equipment, support site monitoring, provide placement of concrete monuments to designate pit locations, and any other activities which ensure the smooth operation of the sites.

HISTORY

The decision to use two sites evolved from the early use of the NTS as a nuclear testing facility. As early as 1961, when radioactive waste was accumulated from testing work on the NTS, certain areas were designated as "Radioactive Waste Storage/Management Locations." Two of those areas became the present RWMSs (2).

In 1974, the NTS accepted for storage, TRU waste from Lawrence Livermore National Laboratory. In 1976, a high-specific-activity (HSA) waste shipment was received from Mound Laboratory for disposal. These events lead to the decision that DOE could use the NTS as a waste disposal location for facilities that could not dispose waste at their own sites. In 1978, detailed studies of the NTS environment and the use of this site for waste disposal became official. Fiscal support was obtained from DOE to fund a small staff and to initiate the construction of early pits and trenches. A chargeback fee was established for all waste generators.

Waste activities have increased over the years. In the 1980s, a major cleanup of the atmospheric test locations began on the NTS. Waste from tower shot locations, which included contaminated cable and equipment, soil, and other contaminated debris, was removed and disposed in the Area 3 RWMS. This task was called the Waste Consolidation Project (3). By 1988, this waste had completely filled the first nested pair of

subsidence craters in Area 3 (Disposal Unit U3ax/bl) and the unit was ready for final closure. This single disposal unit contains 219,914 m3 (7,766,197 ft3) of waste contaminated with 1211 curies of activity. However, one shipment of this waste, received in 1980, contained 1024 curies of tritium. Since 1989, all waste in the Area 3 RWMS has been placed in a second set of disposal craters, the U3ah/at unit. Although the waste consolidation project is no longer in progress, the current unit is used to dispose containerized bulk waste, typically received in transportainers. Large pieces of unpackaged waste, such as machinery, are considered for disposal there. At the end of FY 1995, over 311,500 m3 (11,000,000 ft3) of LLW had been disposed in the Area 3 RWMS. A calculation of the waste received being decayed to the year 2100 shows that only 166 curies of waste would exist in the Area 3 RWMS (this calculation omits daughter-product formation).

The Area 5 RWMS has also served the DOE complex as a waste disposal site using conventional shallow land burial practices. LLW and MW (under interim disposal authority) are disposed of at the site; TRU and MW are also stored at the site. The Area 5 RWMS has a 4.4 hectare (11 acre) portion dedicated to classified LLW and a number of Greater Confinement Disposal (GCD) boreholes. These GCD cells measure 3 to 3.7 m (10 to 12 ft) in diameter and are 36.6 m (120 ft) deep. This disposal methodology was used from 1984 to 1989 for the disposal of HSA waste. In 1989, the state of Nevada declared the GCD boreholes to be injection wells; they have not been used since. The lower 15.2 m (50 ft) of the GCD borehole contained the waste topped with a 21.3 m (70 ft) soil cover. Area 5 has the capability to dispose waste in a deeper pit, Pit 6, to a depth of 14.6 m (48 ft). This deeper cell was specifically designed to handle a thorium waste stream which requires deeper burial due to the radon flux. As the large volumes of waste being received at the NTS facilities filled the major disposal cells in use, the construction of a new LLW pit was necessary during FY 1995. The excavation of Pit 5 started on January 23, 1995, and was completed February 23, 1995, removing 166,683 m3 (218,000 loose yd3) of dirt. The bottom of the pit was plated with 2,600 m3 (3,400 yd3) of Type II soil. The floor of the pit measures 305 m (1,000 ft) long, 46 m (150 ft) wide, and 6.7 m (22 ft) deep, with all walls having a 1:1 slope, or a 45 degree angle. Included in the excavation was an access ramp built on the south end of the pit for a truck entrance into the pit. The pit has an estimated disposal capacity of approximately 70,800 m3 (2,500,000 ft3). At the current rate of waste receipt, this pit should provide disposal capacity for approximately five to six years. The cost of excavating this facility was $445,000 or $2.18 per cubic yard of soil removed from the cell. The pit was put into service on May 15, 1995.

The Area 5 RWMS receives solid waste containerized in wooden boxes, metal boxes, and 208 or 322 liter (55- or 85-gallon) steel drums. The boxes are stacked to form walls which are used to make nests for the stacking of drums. Waste is stacked in a tiered staircase configuration to within 1.2 m (4 ft) of the natural grade of the surface. A cover of soil 2.4 m (8 ft) thick is emplaced over the waste prior to developing a final closure cap.

At the end of FY 1995, over 190,000 m3 (over 6,750,000 ft3) of LLW had been disposed in the Area 5 RWMS. This volume includes waste in GCD boreholes and in the classified area. The total curie count amounts to 7,311,972 curies, which, decayed to the year 2100, amounts to

approximately 42,870 curies. Most of those curies are fission products and uranium/thorium isotopes.

There is approximately 612 m3 (21,610 ft3) of mixed TRU, received from 1974 through 1990 stored on the TRU storage pad. Since the TRU waste storage pad has not been granted interim status, the waste is being stored under a settlement agreement with the state of Nevada.

Between September 1987 and May 1990, over 5,664 m3 (200,000 ft3) of MW was disposed in Pit 3 under interim status authority. MW currently being generated on the NTS is being stored at the Area 5 RWMS on the TRU waste storage pad, also under an agreement with the state of Nevada.

STATISTICS

The comparative waste statistics presented in the following section have been taken from the latest available version (December 1995), "Integrated Data Base Report-1994: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics." This document is updated annually and provides the DOE complex with information, as well as information from commercial disposal sites.

At the end of 1994, a review of the NTS and DOE waste disposal records indicated that the two NTS waste facilities have received over 16 percent, by volume, of all DOE complex waste ever generated, and over 10.7 percent of all waste, considering both commercial and DOE (Fig. 1). Currently 95 percent of all curies received at the NTS have been tritium (Fig. 2). When analyzed by waste volume, however, nearly 75 percent of the waste is fission products and uranium/thorium. This is due to the large volume of NTS and Fernald waste, both of which contain fission products and uranium/thorium. At the end of 1994, over 9.8 million curies of waste had been accepted at the NTS.

Fig. 1

Fig. 2

Figure 3 is a plot of total waste volume, by year, with a curve of offsite-generated waste volume imposed upon this figure. This curve shows the major NTS cleanup activity noted earlier, which occurred through 1989 (the "All Waste" volume area less the "Offsite Waste" curve). The curve also indicates a break from trend in 1990 and 1991. This decrease in waste volume was a result of the 1990 Tiger Team assessments which impacted the NTS. Disposal capability was suspended to implement the waste acceptance criteria. The annual volume of waste disposed at the NTS, as compared to that of the entire DOE Complex shows that the NTS is currently handling about 30-40 percent of all DOE waste (Fig. 4). In 1994, the NTS accepted nearly 30 percent of all radioactive waste which was disposed from both commercial and DOE sites.

Fig. 3

Fig. 4

In FY 1995, a total of 25,063 m3 (885,000 ft3) of LLW (917 shipments) was received. This accounts for approximately 85 percent of all waste shipped offsite during the fiscal year from all DOE facilities that dispose waste offsite. Similar results were noted in FY 1993 and 1994.

CONCLUSIONS

When considering where to site a waste disposal facility or which disposal methodology to use, the primary focus should be to prevent the waste from affecting the environment and exposing the public. Both the Area 3 and Area 5 RWMSs meet that criteria. The sites are located in arid regions, with an average annual precipitation between 10 and 15 centimeters (4 and 6 in) a year. Depth to the water table is 488 m (1600

ft) and 238 m (780 ft), respectively. Studies have shown that there is virtually no recharge from the surface to the water table at the location of the RWMS's. A strict waste acceptance criteria prevents the introduction of free liquids into the waste, and requires strong waste packaging. Each one of these features alone would enhance any facilities ability to keep the hazardous material from reaching the environment, but together, as they are found at the NTS, these physical and administrative barriers provide a waste disposal condition that is quite possibly second to none anywhere in the world.

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43-4

COVER DESIGN FOR DISPOSAL UNITS IN ARID CLIMATES: AN ALTERNATIVE TO THE RCRA GUIDANCE

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ABSTRACT

Landfill cover designs described in the Resource Conservation and Recovery Act (RCRA) technical guidance are primarily intended for humid environments. These covers are not necessarily optimal, or even appropriate, for use in arid regions such as the Nevada Test Site (NTS). The Department of Energy, Nevada Operations Office initiated the Integrated Closure Program to satisfy the need for final closure of waste disposal units at the NTS. The program will achieve the following goals:

develop and implement a cover design appropriate for use in an arid environment while meeting the intent of RCRA guidance,

permanently close existing and future waste disposal units at the NTS Area 3 and Area 5 Radioactive Waste Management Sites,

and protect public health and the environment using the best available technology.

In an Alternative Evaluation Study a cover system was developed for an inactive disposal unit named U3ax/bl. This cover system consists of two major components: a cover design consisting of a rooting and storage

layer, a geosynthetic clay liner (GCL), and a soil cement base; and a perimeter dike design consisting of soil cement (up-gradient) and rock riprap (down-gradient).

A typical RCRA cover design consists of a multi-layered system sloped at 3 to 5%. This slope is intended to promote water drainage from the cover surface. The U3ax/bl cover system differs from the typical RCRA design in several ways. An uppermost rooting and storage layer (native alluvium) provides a suitable habitat for native plants and stores moisture from infrequent precipitation events until evapotranspiration can remove it from the cover. Under the rooting and storage layer is a hard layer consisting of a GCL and soil cement intended to control water infiltration in extremely wet years as well as prevent biointrusion. The soil cement layer will also resist minor subsidence. A soil cement perimeter dike placed on the up-gradient side of the cover provides protection from run-on (flow draining onto the disposal unit). The down-gradient perimeter dike incorporates rock riprap to prevent erosion from precipitation run-off.

The U3ax/bl cover system, with a minimum design life of 500 years, will control erosion, limit infiltration, enhance evapotranspiration, function with minimum maintenance, limit upward vapor transport, discourage biointrusion, and accommodate minor subsidence. Thus, this cover system is better suited for use in an arid environment than a typical RCRA cover.

BACKGROUND INFORMATION

The Department of Energy, Nevada Operations Office (DOE/NV) has made the closure of U3ax/bl a high priority. U3ax/bl is an inactive mixed waste disposal unit located at the Area 3 Radioactive Waste Management Site (RWMS) on the Nevada Test Site (NTS). Because several closures are planned at both the Area 3 and Area 5 RWMSs, the Integrated Closure Program was initiated to develop a conceptual cover design which is transferable to similar sites saving time and money. Under the Integrated Closure Program, an Alternative Evaluation Study (AES) was conducted (July, 1994) to select a preferred cover design for U3ax/bl, to build consensus among various DOE/NV contractors, and to leave a record justifying major decisions in conceptual design. An AES is a modified form of value engineering designed to reach a technical consensus among the participants.

The goal of the Integrated Closure Program is to permanently close existing and future waste disposal units at the NTS Area 3 and Area 5 RWMSs, using the best available technology to protect public health and the environment. Because the disposal units fall under multiple regulations, an integrated approach is needed to ensure that similar wastes in similar environments receive similar closures. The closure philosophy consists of enhancing favorable natural processes, avoiding unjustifiable design inconsistencies between units, complying with applicable regulations, following standard industry practice, and applying a level of effort that is commensurate with other DOE facilities yet does not set unreasonable precedent.

WHY RCRA TECHNICAL GUIDANCE IS NOT APPROPRIATE FOR ARID ENVIRONMENTS

Conceptually, ground water is protected under Title 40 Code of Federal Regulations (CFR) Part 264 by the dual-pronged approach of prevention and correction. Under the less stringent Part 265, prevention is the primary method of protection (1). In support of these regulations, the Environmental Protection Agency (EPA) developed guidance documents that

present design parameters that comply with the requirements of 40 CFR Parts 264 and 265 (2). Covers described in the RCRA technical guidance are primarily intended for environments where precipitation exceeds evapotranspiration rates. These covers are not necessarily optimal for use in arid regions such as the NTS where evapotranspiration rates greatly exceed precipitation. The main premise in this guidance is prevention of contaminant migration through the control of run-off and placement of low permeability barriers limiting infiltration.

Recommended cover components from RCRA technical guidance include:

vegetation or surface armor on a soil layer (minimum 60 cm thick),

filter layer, drainage layer (minimum 30 cm thick), and flexible membrane liner (FML) and low permeability soil layer (minimum 60 cm thick) (2).

Optional cover components from RCRA technical guidance include:

biointrusion barrier (cobbles),

gas vent layer, and

geosynthetic filter materials (2).

RCRA requires a 30 year monitoring and maintenance period under the post-closure care and use of property section in the technical guidance (40 CFR Part 265.117). The U3ax/bl cover system has a minimum design life of 500 years which exceeds the RCRA time period. The 500 year minimum design life meets the requirements for cover longevity prescribed in DOE orders (DOE 5820.2A). Several DOE and commercial low-level waste disposal facilities in the arid western United States use a cover design life on the order of 500 years (3).

Typical conditions at disposal sites in humid regions include:

precipitation exceeds evapotranspiration (the potential for deep recharge exists),

net water movement is downward (leachate migration is toward the water table),

the water table is shallow (the distance a contaminant must travel to reach the uppermost aquifer is relatively small), and

the water table has a significant gradient that results in lateral flow (should contaminants reach the uppermost aquifer they will continue to migrate down-gradient) (4).

Conditions at the Area 3 and Area 5 RWMSs on the Nevada Test Site include:

evapotranspiration exceeds precipitation (the potential for deep recharge is small),

net water movement is upward (leachate migration is away from the water table) (4),

the water table is deep (approximately 490 m in Area 3 and 250 m in Area 5), therefore the distance a contaminant must travel through the vadose zone to reach the uppermost aquifer is large (5,6),

the upper aquifer is contained within a closed basin which does not have a gradient that results in significant lateral flow (should contaminants reach the uppermost aquifer they will continue to migrate down-gradient very slowly) (4).

Over the long-term, erosion is a bigger threat to cover integrity and performance than infiltration based on the arid environment and storm characteristics at the U3ax/bl disposal unit. U3ax/bl is located in Yucca Flat and receives an annual mean rainfall of 15.9 cm with an annual evaporation rate of 180 cm (7). This indicates that evapotranspiration is the strongest influence on the moisture distribution within the upper

alluvium. Precipitation is highly variable in Yucca Flat. Intense, isolated storms occur in the summer and long duration, low intensity storms occur in the winter (8). Between storms there are long dry periods with high evaporation rates. Relating this weather sequence to long-term closure cap performance, precipitation run-on and run-off from the high intensity storms would have the largest deleterious effects on a cover due to the erosion potential. RCRA technical guidance recommends a final cover slope of 3 to 5% to promote run-off (2). This approach, while reducing infiltration, increases the cover erosion potential thereby increasing maintenance costs unnecessarily.

THE ALTERNATIVE COVER DESIGN

In recognition that conditions vary from site to site, RCRA technical guidance and 40 CFR Part 265 allow for alternative cover designs provided long-term performance equivalent to the recommended designs is met. In meeting these performance objectives, the cover design for U3ax/bl uses an alternate approach to control erosion, limit infiltration, limit vapor transport, enhance evapotranspiration, prevent biointrusion, and accommodate minor subsidence (9). Figures 1 and 2 show the U3ax/bl cover system conceptual model.

Fig. 1

Fig. 2

The principal differences between this cover alternative and RCRA recommended designs are:

- greater thickness (1.5 m verses 60 cm) of the upper soil layer (rooting and storage layer) which provides storage capacity for water,
- no drainage layer or low permeability FML,
- low permeability layer provided by a geosynthetic clay liner (GCL) and soil cement, and
- final slope of 0.5% instead of the RCRA recommended 3 to 5%.

The key adaptation of the cover to arid areas is the uppermost 1.5 m thick rooting and storage layer consisting of native alluvium. This layer provides a suitable habitat for plants and acts as a "sponge" to store infiltrating water until the plants and evaporation can recycle the water back to the atmosphere. The relatively flat cover slope (0.5%) will reduce erosion from cover run-off. A lateral drainage layer, as seen in many RCRA cover designs, is not needed since the rooting and storage layer provides sufficient storage for all infiltrating precipitation. The lower cover layers, consisting of a GCL and a 1 m thick soil cement layer, collectively function as a low permeability barrier to downward infiltration and upward vapor transport, as well as deter biointrusion. The soil cement layer will also resist minor subsidence. The rooting and storage layer will control infiltration in most years, except in extremely wet years when the GCL and soil cement will become the primary infiltration barrier. In general, rigid layers such as soil cement may be susceptible to cracking. The GCL is included primarily to bridge over small cracks, but also as a redundant low permeability barrier. Tables I and II present a functional analysis of the cover layers and perimeter system, respectively. The use of durable, natural materials which are available on site will promote longevity, reduce maintenance, simplify construction, and lower costs.

Table I

Table II

HOW THE ALTERNATIVE COVER MEETS RCRA REQUIREMENTS

The requirements from RCRA 40 CFR Part 265.310 are:

Provide long-term minimization of migration of liquids through the closed landfill.

Function with minimum maintenance.

Promote drainage and minimize erosion or abrasion of the cover.

Accommodate settling and subsidence so that the cover's integrity is maintained.

Have a permeability less than or equal to the permeability of any bottom liner system or natural subsoils present (10).

Although the proposed cover design does not follow the specific design guidelines for a typical RCRA cover, the U3ax/bl cover design will meet the intent of RCRA 40 CFR Part 265.310. The proposed cover design meets RCRA requirements by the following:

Migration of liquids to the water table is prevented by the GCL and soil cement layers.

Maintenance is minimized by reducing the cover slope to 0.5%. This will decrease erosion of the cover caused by run-off.

The soil cement layer accommodates minor settling and subsidence.

The soil cement up-gradient perimeter dike resists run-on erosion while plants and a relatively flat cover (0.5%) resist run-off erosion.

Permeability requirements are met because the hydraulic conductivity of the GCL and the soil cement are greater, by at least several orders of magnitude, than that of the native alluvium below U3ax/bl.

ALTERNATIVE COVER DESIGN AND ENGINEERING STUDIES

A series of engineering activities have been or are currently planned to be conducted in support of the U3ax/bl conceptual cover design. These efforts will provide an engineering basis for the design of a closure cap that will meet or exceed RCRA requirements. Currently a Conceptual Design Document (CDD) is being prepared based on the preferred cover alternative from the AES. Formal value engineering studies under the direction of a certified value specialist will follow the CDD. Cost estimates will be prepared for improving functional value of the currently planned design features. Both up-front and life-cycle considerations will be addressed. Detailed cover design is the final step in the process before construction.

CONCLUSION

A RCRA alternative cover design was developed at an Alternative Evaluation Study and consists of vegetation, a rooting and storage soil layer, a GCL, a soil cement hard layer, and a soil cement (up-gradient) and rock riprap (down-gradient) perimeter dike. The cover will provide an increased level of protection against run-off erosion, run-on erosion, biointrusion, fluid migration, and minor subsidence. The cover design was developed specifically for an arid environment where evapotranspiration rates exceed that of precipitation. This design, while differing from the RCRA technical guidance, meets the requirements of 40 CFR Part 265.310 by utilizing techniques which are appropriate to arid site conditions, while enhancing long-term performance.

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RECYCLING AND WASTE MINIMIZATION ALTERNATIVES ASSOCIATED WITH UNEXPLODED ORDNANCE REMEDIATION AT THE TONOPAH TEST RANGE

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ABSTRACT
Six unexploded ordnance (UXO) disposal sites were remediated at the Tonopah Test Range. Each site contained various types of UXO and miscellaneous debris associated with range cleanup operations. Several

methods of waste minimization and recycling were used to assist in the clean up of the six ordnance disposal sites. Clean up operations at all six sites were performed at the same time to save time and money. Several process changes and material substitutions were made during the course of operations to minimize waste volume.

Using process knowledge, Explosive Ordnance Disposal (EOD) specialists knowledge, the RCRA scrap metal exclusion, and on-site radiological screening, items removed from the ordnance sites were segregated into recyclable, non-recyclable debris, and potentially hazardous waste piles. The above techniques greatly minimized the potential for cross contamination and mixed waste generation especially at two of the nine sites where depleted uranium was detected within some of the scrap being removed at one and soil bags of unknown origin were found at another.

All UXO were detonated to render them safe, demilitarize their appearance, and to remove inert filler so that the scrap ordnance could be recycled. Volume reduction, in the form of compacting, reduced trucking requirements substantially as well as the final disposition of the compacted material (landfill disposal vs. recycling). Based on the types of metals being recycled (i.e., aluminum, steel), transportation and disposal costs were recovered.

By the use of onsite contractors excess explosives (i.e., Composition C-4), EOD specialists were able to assist them in disposal of their explosives and cut back on the purchasing of additional explosives for demilitarization operations. Due to the various waste minimization and recycling practices, ordnance clean up and scrap removal from the six ordnance disposal sites were completed within five months. The overall project cost savings was approximately $200,000.

INTRODUCTION

The Tonopah Test Range (TTR) is a research facility with the mission to test the mechanical operation and delivery systems for nuclear ordnance and other defense-related projects. Nuclear detonations do not occur during testing because only mock nuclear ordnance is used. Sandia National Laboratories (SNL), under the authority of the U.S. Department of Energy, Albuquerque Operations Office (DOE/AL), has had historical responsibility for the operation of the facility. Through a Memorandum of Agreement with DOE/AL, primary responsibility for environmental restoration activities associated with TTR has been transferred to the DOE Nevada Operations Office (DOE/NV); however, DOE/AL maintains an oversight role of range activities (DOE/NV, 1995).

The TTR is located in Nye County, Nevada, on the northern portion of the Nellis Air Force Range. It is approximately 255 km (140 mi) northwest of Las Vegas by air (ERDA, 1975). It occupies about 1,616 square kilometers (624 square miles). The TTR is bordered on the south, east, and west by the Nellis Air Force Range and on the north by sparsely populated public land administered by the Bureau of Land Management and the U.S. Forest Service (DOE/AL, 1992). The remoteness of the TTR and its restricted airspace ensures that tests can be conducted with a high degree of safety and security (ERDA, 1975).

RECYCLING AND WASTE MINIMIZATION ALTERNATIVES ASSOCIATED WITH UNEXPLODED ORDNANCE REMEDIATION

The U.S. Department of Energy, the U.S. Air Force, and various DOE contractors have been utilizing TTR to field test conventional and special weapons, research rockets, and artillery since the late 1950's. During range clean up operations, Unexploded Ordnance (UXO) and

associated debris not retrieved as part of the test, were deposited in six locations across the range. The DOE/NV Environmental Restoration Division proposed cleaning up the six UXO disposal sites. Several methods of waste minimization and recycling were used to assist in the cleanup of the six ordnance disposal sites. Several process changes and material substitutions were made during the course of the project to minimize waste volume and type. Each disposal site was unique in the type and quantity of UXO and debris it contained. The largest site contained over 22,000 bomblets (baseball and softball sized ordnance) as well as several 500-lb and 2,000-lb bombs. Another site contained one thousand 55-gallon drums which had been used as targets, along with rocket motors and associated debris, and copious amounts of non-recyclable debris. At one site, approximately 150 rocket motors ranging in size from 5-inches in diameter and 6-feet in length to 30 to 40-inches in diameter and 12-feet in length were found.

Using process knowledge, the RCRA scrap metal exclusion, and on-site radiological screening, the Explosive Ordnance Disposal (EOD) specialists, segregated the items removed from the ordnance sites into discrete piles. The piles consisted of UXO requiring processing, recyclable scrap (aluminum and non-aluminum), non-recyclable scrap (i.e., wood, concrete), and potential hazardous waste. These techniques greatly minimized the potential for cross contamination and mixed waste generation.

All UXO were explosively detonated/processed to render them safe as recyclable scrap. To be accepted by the recycler, the scrap could not be recognizable as ordnance. In addition, it had to be less than 5-ft in length and 24 to 30-in. in diameter for the best recycling return. Due to these criteria, the majority of the scrap ordnance required further demilitarization through the additional use of explosives and/or a cutting torch. To reduce the cost of purchasing the additional explosives for demilitarization activities, an agreement was made with onsite contractors to use up their excess explosives (i.e., Composition C-4, shaped charges, detonating cord). The explosives transferred to DOE/NV had a combined Net Explosives Weight (N.E.W.) of approximately 410 lbs. This was a double cost savings for DOE since additional explosives did not have to be purchased for the project and DOE did not have to dispose of the excess explosives as hazardous waste. The use of onsite DOE subcontractors for cutting activities also reduced the overall project costs.

It was originally intended that each disposal site would have one UXO processing area associated with it. Through EOD specialists expertise and knowledge it was determined that the UXO were safe to transport. This made it possible to establish one central processing area for all six sites. Having one processing area reduced the time and money needed to mobilize and demobilize equipment and EOD personnel to each site. It also decreased the number of samples to be collected during closure activities and reduced the amount of soil potentially contaminated by UXO processing activities.

Processing techniques were refined over the duration of the project to become more efficient. This resulted in changing the type of waste generated, reducing the amount of explosives used, and increasing the number of UXO processed at each detonation. For example, the original planned method for processing bomblets included the use of PVC tubes as containers for the bomblets. This was changed to cardboard tubes prior to

the project's start since the PVC waste would not be biodegradable. During field activities it was decided to discontinue using the cardboard tubes. Experience found that filling the tubes was an unnecessary and time consuming step and that a large amount of cardboard waste was being generated. Also, without the size restriction due to the diameter of the tubes, more bomblets could be detonated during a single detonation, which in turn decreased the duration of bomblet processing.

Volume reduction was implemented at the site containing the empty target drums. Some of the drums were used for scrap containment and those left over were compacted by a drum compactor. By reusing and compacting the drums, the drums became salvageable, thus saving landfill space and reducing disposal costs. The number of truck loads required to remove the drums was decreased from ten to three which saved labor hours and transportation costs.

A total of 120 tons of steel and approximately ten tons of aluminum were shipped offsite for recycling. Transportation costs were recovered with the profits generated by recycling the scrap metal. An agreement was made between DOE/NV and one of the occupants of TTR to accept the 20 tons of non-recyclable debris removed from the six sites at their TTR landfill. This saved DOE/NV the cost of shipping the debris to either the Nevada Test Site or to an offsite landfill for disposal.

The overall project cost savings was approximately $200,000. This savings was the direct result of the waste minimization and recycling techniques discussed above, as well as the coordination of onsite contractors with project needs.

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43-6

THE STATE OF NEVADA'S OVERSIGHT OF THE DEPARTMENT OF ENERGY'S NEVADA OPERATIONS OFFICE ENVIRONMENT MANAGEMENT PROGRAM

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ABSTRACT

The Nevada Test Site is located 65 miles north of Las Vegas, Nevada and is the principle location where the United States Department of Energy conducted nuclear testing activities. Although all DOE facilities carried on various types of activities which created different waste management concerns, the NTS is a unique site. All the other sites in the DOE complex are dealing with waste problems which are principally the result of the poor or mismanagement of wastes generated from the production of nuclear weapons. The Nevada Test Site was, for the most part, intentionally contaminated, i.e. was identified specifically for nuclear testing. However, wastes which should and could have been managed in a fashion to minimize their environmental impacts, for the most part, were not. As a result of the activities that have been conducted, there are thousands of acres of surface and hundreds of square miles of subsurface contamination. Since these conditions have been acknowledged, both the State of Nevada and the Department of Energy, Nevada Operations Office have been attempting to correct the national perception that the Nevada Test Site, in its entirety, should be written off as a national sacrifice zone and therefore, could be the designated disposal site for all defense related waste and require only limited further evaluation.

This presentation will discuss the Department of Energy Nevada Operations Office historic and present day environmental restoration, waste management and disposal practices, how the State of Nevada's involvement has impacted both environmental restoration and waste management practices to date and concerns and issues the State feels are still inadequately addressed and remain unresolved. The management of the principal types of wastes, (solid, hazardous, mixed, low level and greater than class C) will be reviewed in relationship to State regulatory authority and how the State has or has not driven waste management practices at the Nevada Test Site. The extent to which Nuclear Regulatory Commission requirements and the Defense Nuclear Safety Board recommendations are implemented within the context of DOE's implementation of the self imposed ORDER 5820.2A "Radioactive Waste Management" will also be discussed.

HISTORY OF THE SITE AND WASTE MANAGEMENT PRACTICES

The Nevada Test Site (NTS) is a United States Department of Energy, Nevada Operations Office (DOE/NVOO), nuclear testing and research facility occupying 1350 square miles of federally owned land in southeastern Nye County, Nevada approximately 65 miles northwest of Las Vegas (1).

The NTS has been the primary continental location for the United States nuclear testing program since 1951. Programmatic functions, as defined by DOE/NVOO, include the management and disposal of wastes generated by NTS activities and by other USDOE defense-related facilities across the United States. On-going Environmental Management activities conducted at the NTS include the storage and disposal of low-level radioactive waste (LLW), storage of LLW and transuranic (TRU) mixed waste, environmental restoration efforts, and technology development projects (1). DOE/NVOO defense programs still conduct high explosive testing, subcritical nuclear tests, as well as other research and in addition, the Defense Nuclear Agency (DNA) is still conducting research at the NTS as well as other Department of Defense (DOD) entities which utilize the site for various types of testing and training exercises. All of these activities have the potential to generate wastes.

Although DOE/NVOO was not directly responsible for the activities that caused contamination that was done by all of the research labs and other tenants during testing activities, the NTS is managed by DOE/NVOO and therefore, as the landlord, they become the responsible entity. Historically, the entities associated with testing did not consider environmental or waste management issues that did not directly impact the testing mission. The disposal practices at the NTS for hazardous and radioactive material as well as solid waste had been for, the most part, unmanaged. Wastes were not adequately characterized and were inappropriately disposed (i.e. dumped onto the ground or abandoned at the point of generation, placed in subsidence craters or buried in sites that were not characterized), without concern for the contaminants fate in the environment. This practice meant documentation for a waste stream was not recorded or recorded randomly therefore, now it is very difficult to utilize process knowledge as a basis to characterize a waste stream.

As a result of the infamous Rocky Flats Raid, in 1987 DOE/HQ (Washington, D.C.) initiated what was referred to as "Tiger Team Audits", individual site audits were conducted by groups of DOE personnel from other sites. In response to the prospect of these audits, DOE/NVOO conducted their own audit and prepared an Environmental Survey Action Plan (ESAP). This document provided information on 105 sites requiring some form of action which indicated that DOE/NVOO was in noncompliance with environmental regulations and was or had potentially illegally discharged wastes. A copy of this draft Plan was made available to the Nevada Division of Environmental Protection (NDEP) for review in 1988. In addition to the ESAP, DOE/NVOO also submitted a RCRA Part B application for a Hazardous (Mixed) Waste Disposal Facility in which the Part A also identified a number of existing facilities which would be regulated under RCRA. Submittal of this information resulted in the State of Nevada formally beginning to interact with DOE/NVOO through NDEP.

After reviewing the ESAP document, NDEP began drafting a Finding of Alleged Violation (FOAV). Concurrent with the drafting of the FOAV, a Consent Order was prepared which would mandate DOE/NVOO take specific action relative to the non-compliance issues and begin to provide additional information about the NTS operations related to environmental regulations. DOE/NVOO proposed that NDEP set aside the FOAV & Consent Order and instead enter into an "Agreement in Principal" (AIP) which USDOE facilities throughout the country were negotiating with other States.

NDEP held in abeyance the issuance of the FOAV & Consent Order choosing instead to enter into the AIP that was signed by both parties on September 7, 1990. This document detailed the known environmental issues of concern and established the basis for how the two groups would interface with each other.

DOE/NVOO had always been reluctant to supply information or allow access to activities at the NTS by State personnel including NDEP employees. The stated reasons for this were the requirements which mandated secrecy and silence related to any activities associated with the development and evaluation of nuclear weapons. As a result of the AIP, access and clearance were made available to State personnel for the NTS. Beginning in January 1991, the State's presence changed from 1 or 2 days a year to 10 full time employees whose job responsibilities are totally dedicated to environmental evaluation and oversight of DOE/NVOO activities.

STATE OVERSIGHT ACTIVITIES

It became apparent that if the State was going to be effective, it would be necessary to become familiar with not only the needed environmental restoration and waste management activities but also the activities and associated management practices that created the problems. This presentation will not elaborate on the State's concerns related to program management but does acknowledge that DOE/NVOO is making significant progress in mandating that tenants address environmental and waste management issues related to proposed activities prior to them being initiated.

ENVIRONMENTAL RESTORATION

Forty plus years of nuclear testing and associated activities has left its mark on approximately 30% of the NTS. In addition to the approximately 1000 nuclear tests, there were numerous other sites, presently over 2000 identified, containing abandoned materials and wastes that needed to be characterized and corrective actions initiated. Types of contaminants at sites varied from solid and hazardous, including mixed, to low level and possibly some TRU wastes. NDEP, DOE/NVOO and DNA, the major DoD tenant, are in the process of finalizing an Federal Facility Agreement & Consent Order defining the process by which all contaminated sites will be evaluated and appropriate corrective actions implemented. These corrective actions have the potential to generate large quantities of wastes of all types.

With respect to characterization of contaminated sites, the State is particularly interested in the studies being conducted at Oak Ridge, Tennessee which question the colloidal theory of subsurface radionuclide transport. Based on a perceived migratory process which has historically indicated that most radionuclides would not travel far, contamination which resulted from the underground testing activities has been basically unmanaged. The studies suggest what was once considered to be a non-migratory radionuclide contaminant, even in groundwater could, in fact, be transient as a resultant of minute colloidal migration. This issue is of significant concern as approximately 30% of the nuclear tests, representing up to 90% of the source term, were conducted at or within the ground water table at the site. Prior to being able to manage these large contaminated areas, which potentially include hundreds of square miles, the bounds of the contaminated areas must be defined. A large portion of DOE/NVOO's efforts are presently directed toward trying to address the extent of the subsurface contamination and its potential movement.

WASTE MANAGEMENT

The State of Nevada enacted hazardous waste laws in 1981 and began to adopt the Federal RCRA regulations in 1984. NDEP has had EPA delegated authority for the federal RCRA base hazardous waste program since November 1, 1985 and was delegated federal Hazardous and Solid Waste Amendments (HSWA) authority on July 29, 1992 for mixed wastes and corrective action activities. As was noted previously, DOE/NVOO submitted a Part B Application to the state for a mixed waste disposal facility in 1988 subsequent to mixed waste being declared a regulated waste under RCRA and indicated its present facility was eligible to continue operations under interim status. DOE/NVOO later contended that the State didn't have the authority to regulate mixed wastes until it received that authority from the federal EPA. This issue has always been a point of controversy in that the USDOE, as well as other federal entities, have frequently contended that the State has no authority to act relative to a

federal facility unless so authorized by the Federal government. Presently the NDEP has both State and delegated federal jurisdiction and regulates the treatment, storage, and disposal of hazardous waste, including mixed waste, as well as air and water pollution and solid waste issues on the NTS.

HAZARDOUS (MIXED) WASTE vs LOW LEVEL WASTE ISSUES

The DOE/NVOO is currently storing, under a Consent Agreement, transuranic (TRU) mixed wastes on the NTS that are proposed to be sent to the Waste Isolation Pilot Plant (WIPP) in New Mexico. The actions and time lines to complete the characterization and have the wastes available for shipment along with the disposition of other low level mixed wastes are defined in the consent order issued in accordance with the Federal Facility Compliance Act (FFCA).

The Federal Facilities Compliance Act authorizes the States to evaluate USDOE proposed alternatives and determine how USDOE will treat and manage their mixed wastes. It is difficult to assess the treatment decisions associated with these wastes without also addressing their subsequent disposal alternatives. The State has concerns regarding USDOE's management of low level radioactive waste and to what extent it may effect acceptance of technologies for treatment of mixed wastes which result in a pure low level waste. These forms of treatment would result in the subsequent management and disposition of these wastes to be construed to no longer be under State regulatory authorities.

DOE/NVOO is performing the mixed waste disposal site evaluations utilizing the same criteria and standards that would be applicable for a Low Level Waste assessment. These evaluations need to ensure that the criteria used are also valid to assess the potential migration concerns associated with the hazardous constituents in the waste. This issue could be a significant concern for sites which could be found satisfactory only for containment of certain radioactive isotopes with short half lives. Any assessment of a site must address migration for any of the contaminants that could potentially be disposed at a site and not be limited by just those associated with radionuclides.

The U.S. EPA's proposed Hazardous Waste Identification Rule (HWIR) presents options for excluding hazardous waste from Subtitle C requirements based primarily on how the wastes are managed and on the level of hazard posed by the wastes. As proposed, once a defense low level mixed waste is excluded from the Subtitle C regulations, there will be no clear State regulatory authority to confirm that DOE/NV appropriately manages the waste. Most of USDOE's existing mixed waste could fit into this excluded category.

The management of defense related LLW is presently perceived to be under the sole control of the USDOE although Nevada's State water pollution control laws could be applied. These State authorities would be in conflict with USDOE's interpretation of the Atomic Energy Act (AEA) which allows USDOE to accumulate, store, transport and dispose of these wastes as they alone determine. Neither the EPA nor the Nuclear Regulatory Commission (NRC) has any direct authority over the waste characterized as defense low level waste. DOE/HQ continues to contend that the mixed wastes could and should be more suitably managed under the AEA, (it should be noted that the AEA contains no enforceable environmental requirements). DOE/NVOO has adopted USDOE ORDERS which are equivalent to internal policy directing themselves to perform in a certain manner, however failure to perform as directed has no negative consequences.

Allowing USDOE to remove these wastes from regulation and manage mixed waste under the AEA would basically place USDOE back into self regulation. The primary driver mandating USDOE to adequately characterize and manage their waste and to obtain funding for cleaning up environmentally contaminated sites has come from the State and federal environmental regulatory agencies who presently have this authority.

LOW LEVEL WASTE

The USDOE routinely ships Defense Low-Level Radioactive waste from its other installations around the country to the NTS to be disposed of. At the end of 1994, more than 17,158,689 cubic feet of defense low-level radioactive wastes had been disposed of on the NTS. The prospects for additional wastes to be shipped to the NTS by the USDOE from its other installations throughout the country is a certainty. To put these figures in perspective for the State, the total accumulation of radioactive waste at the US Ecology site at Beatty, Nevada, regulated under NRC agreement status until closed in 1993, totals 4,000,000 cubic feet for the 30 years in which it was in operation.

The State has had three significant concerns related to DOE/NV's accepting, for disposal, very large quantities of low level waste from off-site facilities. These issues and the current status are discussed below.

1. It is the State's position that the existing National Environmental Policy Act (NEPA) documentation under which the NTS was withdrawn, clearly did not authorize the site to function as a major disposal facility for off site wastes.

The State ultimately sued USDOE on this issue and the matter is presently within the court system. However, after the suit was filed, DOE/NVOO almost immediately initiated work on an EIS which has presently been issued in Draft for comment. To DOE/NVOO's credit, they actively sought stakeholder input into the EIS development which has resulted in a much more complete assessment of alternatives upon which acceptable decisions related to total site use, not just low level waste management, should be able to be made.

2. These wastes, classified as low level, are originating from CERCLA and/or RCRA facilities and have historically had very little characterization data to support the determination that they were not mixed wastes.

The issue revolved around the adequacy, accessibility and evaluation of waste characterization information upon which other USDOE sites made a determination that the wastes being shipped were not RCRA wastes. DOE/NVOO's Waste Acceptance Criteria, NVO-325, historically was very deficient in defining these requirements to which the State took exception. DOE/NVOO originally questioned NDEP's authority over what it asserted to be solely a LLW matter and beyond NDEP jurisdiction. NDEP indicated if necessary it would issue a Finding of Violation alleging that the wastes being disposed of were mixed wastes and there had been an inadequate determination made to demonstrate otherwise. After an initial, but brief resistance to the States position, DOE/NVOO accepted and has since welcomed NDEP's input.

Although defense LLW is not specifically regulated, all waste that is proposed for shipment to the NTS is now presumed to be mixed until adequately documented otherwise. Historically, shipments that were identified as only LLW were not required to provide as much documentation as those which were identified as mixed wastes. Information which

supported either position was maintained at the shipping site and frequently not in a readily auditable manner. Subsequent to State involvement significant improvements have occurred. The generators must now more completely characterize these wastes with clearly defined QA/QC requirements and organize their waste characterization information in a manner that is easily auditable. DOE/NV with NDEP assistance and input has defined a records management and submittal protocol for all shippers. Although NVO-325, as it exists today, is significantly improved there is still room for further improvement. One of the areas still needing development is the information supporting process knowledge determinations. There has never been a definitive requirement, until the presently proposed REVISION 2, to identify each isotope and activity level of the radionuclides in the waste, identify limits and to do so in acceptable units. Historically, the source term information that was provided with a waste shipment was a Curies per weight value for the entire waste stream which did not correlate to NRC standards of Curies per volume. This information was originally only provided for the entire shipment and not individual containers.

3. The NTS has never completed Performance Assessments for the sites where wastes were/are being disposed.

DOE/NVOO has not completed a Performance Assessment for the existing disposal sites, therefore the documentation to support that any disposal activity is environmentally sound is lacking. Although the State has significant concerns related to the ability of the facilities to provide containment, it is recognized that the actual existing waste management practices at the facilities are very acceptable. The concerns associated with the facilities ability to demonstrate adequate containment are expanded in depth in the following sections.

The NTS is different from other USDOE facilities in the fact that most of the waste it handles comes from other USDOE and DOD defense related sites. At the present time, 98.5% of the waste disposal that is managed by DOE/NVOO at the NTS is from off-site generators (2). It should also be noted that 85% of waste that has moved off-site from all other USDOE facilities around the country was shipped to the State of Nevada (2). It is the clearly stated intention of the USDOE to use the NTS for the continued disposal of large volumes of defense low-level radioactive wastes into the foreseeable future. USDOE has proposed that the NTS become the largest disposal facility in the country for low level wastes. Proposed shipments are for 1 million cubic feet per year for the next 70 plus years.

USDOE has had an internal policy, since 1988, requiring all USDOE disposal facilities to prepare a site Performance Assessment which would define the capabilities of their low level waste disposal facilities to contain the wastes which are being disposed. Standard practice for private sector facilities would have required a facility to conduct an evaluation prior to operation. To date, there has only been limited compliance with the policy directive to conduct this study on the NTS, even though it has been identified as a deficiency by the Defense Nuclear Facilities Safety Board (DNFSB), which is an oversight advisory board to the USDOE.

Part of the problem is that although DOE/NVOO has routinely requested funding to conduct site evaluation work, it has not received monies to do so because the budgets are driven by what legal requirements must be met.

Those that do not have such drivers become a low priority and therefore, routinely are below the available funding authorized.

There has been some work performed to date at two NTS locations. This site characterization work was initiated only because a portion of these areas were, or had been, utilized for hazardous waste activities and site characterization work and was required to be initiated under the regulatory requirements imposed by NDEP under its hazardous waste authorities.

DOE/NVOO is receiving and disposing of defense low level wastes at sites without a performance assessment. The State has concerns that because this assessment has not been a priority at USDOE, when the assessment is finally completed, DOE/NVOO may not be able to demonstrate the assessment is adequate due to non-existent or insufficient records for the historic waste disposed. It is also questionable if DOE/NVOO can prepare the Performance Assessment in accordance with the guidelines provided by the DNFSB.

In 1994, the DNFSB evaluated USDOE's low-level waste management program at various sites across the USDOE weapons complex. That review resulted in the DNFSB recommendation that in establishing low-level waste sites, USDOE guidance for meeting established performance assessment criteria had not been correctly applied. Reference dose criteria had not been applied to disposal facilities to determine the composite effects at contiguous burial facilities.

What this means is that on the NTS, DOE/NVOO has established low-level disposal sites in a way that significantly complicates performance assessment approaches for determining cumulative radionuclide migration to the biosphere in order to meet prescribed release standards for either USDOE or NRC disposal facilities. In the case of the two low-level waste disposal sites on the NTS, it could be argued that neither site could be adequately evaluated for meeting specific performance assessment objectives. This is not only due to limited records but also the type of historic activities conducted which encompass disposal of unique waste types like USDOE Special Case Waste (SCW) which could include NRC regulated Greater Than Class C (GTCC) waste or the equivalent and utilization of areas impacted by testing.

Because of USDOE's historic disposal practices along with a general failure to follow the agency's disposal evaluation procedures, as defined under USDOE order 5820.2A, it is argued that DOE/NVOO would be hard-pressed to meet even the minimum NRC licensing standards for commingling disposal of LLW with USDOE SCW at the NTS which could include wastes that meet the classification of GTCC.

USDOE Strategy for Management and Disposal of GTCC Low-Level Radioactive Waste and SCW

USDOE's Waste Management Programmatic Environmental Impact Statement specifically excluded the evaluation of these categories of waste. Since the NTS has disposed of this waste and is one of USDOE facilities undergoing performance evaluations for Greater Confinement Disposal of Special Case Waste, the Test Site is a plausible alternative for GTCC waste disposal. In addition, the NTS site wide EIS considers disposal of these wastes as a future possible mission of the NTS and the present version of NVO-325 makes provisions for, on a case by case basis, receipt of these wastes.

Although officially USDOE has taken very limited actions related to the GTCC-DOE\SCW, there are some ongoing activities and issues that have raised the following concerns for the State.

a. Most GTCC LLW is associated with nuclear utility waste (the largest volume being activated metals associated with spent fuel assemblies and reactor core components), and lacks concise estimates of the volume of these wastes.

b. Questions remain on whether fuel assemblies and reactor core components should be considered directly as either spent fuel or high level waste (HLW) rather than GTCC waste. Because of these uncertainties, waste treatment alternatives and storage options for reactor GTCC waste must be resolved first, before disposal.

c. Commingling of NRC regulated wastes with non NRC regulated wastes at a non NRC site raises additional regulatory issues.

d. The only public hearings held on the disposition of GTCC wastes were conducted in Washington, D.C. and Oregon. Neither of these locations have potential sites under consideration which leads the State to believe USDOE may not be interested in potentially impacted stakeholder input.

CONCLUSIONS AND THE FUTURE

This paper is being presented under the general category of success stories and one may view the contents as critical and therefore not as successes. However, the State of Nevada and DOE Nevada Operations Office relations have improved through the past years. We have worked together and realized some significant milestones such as State participation in revisions to NVO-325, participating in the off site audit process, the drafting of the Federal Facility Agreement and Consent Order for corrective actions and the FFCAct Consent Order for mixed waste management. Although we have not overcome all of the obstacles, the State is of the opinion that there has been success in obtaining recognition that there are some significant problems. Being able to provide some definition to the extent of the problems and therefore begin to address these situations must be considered as success.

We will be involved in a number of interesting interactions over the next few years: first, to expedite the process for implementing corrective actions and gaining control over the historically contaminated sites and wastes in an environmentally acceptable and cost effective manner; second, to continue the process of bringing DOE/NV activities under full and fair regulation; third, to develop the acceptable mix between State and federal authority over environmental concerns; and fourth, to accomplish the previous three in a manner that takes into consideration the broad spectrum of stakeholder concerns.

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Session 44 -- SITE REMEDIATION

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44-1

DECOMMISSIONING COSTS OF URANIUM PRODUCTION FACILITIES - AN INTERNATIONAL COMPARISON

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ABSTRACT

The development of the world uranium production since World War II was strongly influenced by strategic demand. While this continued within the former Soviet Union until the fall of the "Iron Curtain" uranium production within the WOCA (World Outside Centrally Planned Economics Areas) countries went through two "boom" and "bust" cycles which caused the opening and unsystematic closure of many uranium mines and mills. During the initial period of uranium production, environmental protection was of no importance and legal provisions setting the terms for proper rehabilitation, decommissioning and institutional control were missing. This situation led to uncontrolled abandonment of production sites. In the former Eastern Block countries this situation prevailed and the massive closure of uranium production capacities in recent years results in major environmental legacies due to lack of technologies and experience in environmental management, lack of financial resources and lack of a legislative and regulatory frame for decommissioning. In the Western World countries since the early seventies uranium mining is strictly regulated regarding the environmental impacts of the projects. Companies are expected to take the costs of decommissioning and restoration into account from the very beginning of the projects and to adjust them continuously during the production phase.

This paper summarizes results of an international study on decommissioning and reclamation costs of uranium production facilities commissioned by the German Federal Ministry of Economics. Within the frame of this study 14 countries with more than 100 projects were evaluated representing a cumulative uranium production of approx. three billion pounds of U3O8 and estimated rehabilitation costs of approx. 3.700 million US dollars.

Due to numerous factors which influence decommissioning costs, the variance of specific costs is quite broad ranging from a minimum of 0.12 US$/lb U3O8 to more than 40 US$/lb U3O8. The weighted average is 1.24 US$/lb U3O8. The specific unit costs per tonne of tailings range between 0.12 and 43 US$.

Factors influencing the costs can be classified as either inherent to the project and site-specific or of more general nature. One dominant inherent factor is the ore grade and ore mineralogy and the geological and hydrogeological setting of the deposit. One of the more important general factors is whether the decommissioning of the projects takes place immediately after mine closure and is financed and carried out by the mining company or is publicly funded and carried out some time after closing-down by public institutions.

During the last twenty years major technology advancements have been made in waste management from uranium mining and milling. As a result the environmental impact of production operations has been mitigated and

sustainable solutions for closing and rehabilitating these operations are available. This development has caused a shift of environmental cost (which substantially increased in absolute terms) from the decommissioning stage of a project to the planning and production stage.

INTRODUCTION AND BACKGROUND

The post World War II uranium production, including 1992, is estimated at approximately 1.8 million metric tonnes of U. WOCA countries contributed more than 1 million mt U to this total (1). Figure 1 shows the major uranium producing countries in order of their shares in the total production. Due to the fall of the 'lron Curtain', the extent, distribution and conditions under which uranium production took place in the Soviet Union and COMECON countries has been revealed in recent years. The following now independent countries were the main contributors to the Eastern Bloc total: East Germany, Czech Republic, Russia, Kazakhstan, Uzbekistan, Ukraine and Bulgaria.

Fig. 1

The development of the world uranium production since World War II (Fig. 2) was initially exclusively determined by the military demand of the superpowers. The very sharp increase until the year 1959, when total annual production went up to approx. 48,000 t U is the result of a less steep but earlier starting expansion of uranium production in the former Eastern Bloc countries due to the immediate exploitation of uranium deposits in the Soviet-occupied countries in Eastern Europe, in particular East Germany, Czechoslovakia and Bulgaria, and a dramatic increase between 1954 and 1959 by WOCA countries, mainly carried by the United States and its close allies Canada, South Africa and Australia.

Fig. 2

While uranium production in the Soviet Union and the former COMECON countries continued to be treated strictly strategically with an almost continuous increase until the middle eighties, the production development of the WOCA countries is characterized by two "boom" and "bust" cycles. After termination of highly profitable supply contracts with the US Department of Energy to cover the military demand, numerous uranium production projects particularly in the United States were given up. At that time legal provisions setting the terms for proper rehabilitation, decommissioning and institutional control were still missing. Environmental protection was not an important issue at that time. Therefore, many sites were abandoned by their owners without appropriate security and recultivation measures being taken. Some of these sites were remediated later within the frame of government financed programs. An outstanding example is the UMTRA project in the United States established under the 1978 Uranium Mill Tailings Radiation Control Act. Since 1980 twenty-four orphan mine and mill tailing sites are being remediated under this program (2). In Australia the Rum Jungle project, which ceased production in 1971, was remediated by the Federal Government during the period 1982 to 1991.

Since the early seventies, the growing environmental awareness in the mining industry of the Western industrial countries has resulted in legislation regulating the operation of new mining projects. The objective of the regulations is to ensure an adequate consideration of the overall economic and socio-economic interests in view of the impact on man and environment caused by mining. This development affects uranium production in particular because mining and processing of natural uranium entails the release of radioactive and chemotoxic substances. Their

potentially harmful effects are countered by imposing special regulations. Today, the basic legal conditions of modern uranium production in the Western World expect the producers of uranium to take the subsequent costs of decommissioning and rehabilitation into account right from the start at the feasibility stage and to adjust them continuously during the production phase.

Towards the end of the "Cold War", major uranium stocks, the decline in reactor demand, a drastic decline of uranium prices and the development of low cost and high grade deposits in Canada caused the closure of many projects in Western and Eastern countries. While in the Western industrial countries the decommissioning of these projects is ongoing and firmly regulated and mostly financed by the owner of the projects, the closure of the state-owned projects in the former COMECON countries results in extensive legacies of environmental devastation since economic and environmental aspects were of minor importance during production. Financial reserves for restoration of the mostly large production sites and plants have not been put in place. Therefore, only minor rehabilitation work is carried out at this stage. The uranium mining and milling plants of the former Soviet-German WISMUT Corporation in Thuringia and Saxony, where production ceased at the end of 1990, represent a typical example of this legacy, though unique in size (3), (4). The German Federal Government has assumed a financial obligation of 13 billion German marks (approx. 8 billion US$) for rehabilitation and decommissioning of these sites of which approx. 3.5 billion marks have been spent during the period 1990 to 1994.

STUDY ON DECOMMISSIONING COSTS

This study summarizes results of a research contract commissioned by the German Federal Ministry of Economics to Uranerzbergbau-GmbH (5). The major objectives of this research were to compile the available cost data of rehabilitation and decommissioning of uranium mining and milling sites, to analyze critical factors influencing the costs, and to draw comparisons. More than one hundred uranium projects were selected to form the basis of these investigations. They include decommissioned plants as well as projects which are still producing or which have been shut down and for which decommissioning plans, including cost estimates, have been drawn up. A separate study within the frame of the above referenced research contract included an analysis of costs of aquifer rehabilitation of in situ leaching projects in the United States.

As characteristic values for the comparison of rehabilitation costs between projects, districts and countries, specific rehabilitation costs were defined as US$ per pound U3O8 of produced or recoverable reserves and US$ per tonne of tailings. All costs were converted to US$ and adjusted to the year 1993.

RESULTS

On the basis of the time of rehabilitation in relation to production, the origin of funds and the party in charge of the rehabilitation, four types of projects can be defined (Fig. 3). The highest costs are related to the "UMTRA Title I-Type" and the "WISMUT-Type" of remediation projects. The "UMTRA Title I-Type" (Type 1) is characterized by production activities at the early stages of uranium mining when environmental impacts of the projects were neither assessed at the planning stage nor mitigated during and after production. The projects were abandoned by their former owners and a long lagtime occurred before restoration work started. The work is publicly funded and mostly carried out by public institutions which

generally have no internal in-sight knowledge of the project development and history. Therefore, a lengthy process of taking inventories, characterization of the wastes, risk assessment, planning and setting up an infrastructure, takes place before the remedial action starts. Furthermore, programmatic documents about policies and procedures have to be developed to satisfy the various stakeholders in the project to justify proposed expenditures and to meet political initiatives. The work is mostly carried out very cautiously in a step-by-step approach without an expertise based on practical experience and often in a pilot-type manner guided by R&D-work. Organizationally and operationally, the Type 1 remediation projects are more expensive than projects carried out by private mining companies. A distinct example are the very considerable differences in the costs of the US UMTRA Title I- and Title II-Programs. Fig. 3

The "WISMUT-Type" (Type 2) stands for remediation projects of former Eastern Bloc countries. Many of the above mentioned characteristics of Type 1 projects are valid. Generally the production processes have an even stronger impact on the environment and the sizes of plants are larger compared to Western standards. Furthermore, an abundance of auxiliary plants are integrated or related to the proper uranium production plant. The WISMUT case itself (6,7), due to the German reunification, takes a special position in this class. The high specific costs estimated for the WISMUT project are caused mainly by the location of the project within a densely populated area, the low ore grade, a major acid generation potential, partly employed unique exploitation methods, lacking experience and routine of regulating and permitting agencies, a dual permitting system, comparatively high personnel and particularly earth moving costs and high social costs.

The "Classical Type" (Type 3) of remediation projects is the most widely distributed one in Western countries. The mining company carries out the remediation work immediately after production ceases and the work is financed from previous earnings. The cost variation of projects within this class is largely due to project-specific factors of influence such as type of deposit, mining method and environment related factors.

Finally, the "ISL-Type" (Type 4) remediation projects represent modern projects which have gone through an extensive environmental assessment already at the planning stage. They have implemented the concept of environmental management throughout the life cycle of the project and rehabilitation is carried out in most cases progressively with the production. In situ leach uranium mines in the United States represent this type of project most distinctly.

Type 3- and particularly Type 4-projects often do not allow a clear separation between rehabilitation costs and production concurrent expenses for environmental protection.

The 14 uranium producing countries investigated in the referenced study represent a cumulative uranium production including 1993 of approx. 2 billion pounds of U_3O_8 This corresponds to approx. 63% of the cumulative world production. This production is related to 1.681 million tonnes of residues from ore processing (tailings) of which 780 million tons refer to plants producing uranium as by-product. The accumulated and estimated costs for decommissioning and rehabilitation of the investigated projects amount to 3.7 billion US$ (excluding the WISMUT project). The resulting specific rehabilitation costs are US$ 1.25 per lb U_3O_8 and US$ 2.20 per tonne of tailings. Leaving those plants out of account which

produce/produced uranium as by-product of primary production of gold and copper the specific costs per tonne of tailings goes up to approx. US$ 4.

FACTORS

The specific costs for the individual projects and countries differ widely (cf. Table I). The large variation is indicative of the dependency of the costs on numerous factors which are either project-specific, or regional or more general and location- and project-transgressing in nature.

Table I

The dominating project-specific factor influencing the specific rehabilitation costs is determined by deposit parameters such as ore grade, ore distribution, mineralogy of ore and surrounding strata and hydrogeological conditions. Production from high grade deposits like the nonconformity-type deposits in the Athabasca region of Northern Saskatchewan for example results in much less mill residues than production from comparatively low grade sandstone-type and conglomerate-type deposits. Furthermore, the nature of tailings determines the amount of work and costs for long-term stabilization. Fine grained argillaceous tailings, for instance those which resulted from the treatment of ores from the black shale deposits of the Ronneburg area in Thuringia take a high expense to dewater and stabilize while the sandy tailings of the sandstone ores of the Western United States dewater much faster and the expense for stabilization and encapsulation is substantially smaller, though long-term stabilization against erosion is a major cost factor. The rehabilitation of open pit mines is generally more expensive than of underground mines, in particular if, due to the depth of the deposit, large quantities of waste rock had to be removed and stored in dumps. Sulfitic acid generating waste rock needs particular attention in open pit reclamation and is best relocated below the groundwater level. Besides the remediation of tailings, the rehabilitation of open pit mines and related mine dumps are the most expensive restoration measures since they involve large-scale earthmoving. On the other hand, under humid climatic conditions and with acid generation in mines, mine dumps and tailings water treatment costs can become a substantial portion of the total remediation cost.

New developments in waste management, for instance the in-pit deposition of tailings that is now recognized as the best practical technology, has a cost saving effect on final restoration.

A further quite important factor influencing the costs of decommissioning is the permitting process and the permitting practice of the regulating agencies under the existing regulatory regimes.

INTERNATIONAL COMPARISON

The uranium producing countries investigated with regards to decommissioning costs and major factors of influence can be assigned to the following three groups:

1) Western industrial and other established uranium producing countries with a mining tradition, such as: Australia, Canada, USA and France, Gabon, Namibia, Niger and South Africa.

2) Former COMECON countries: Bulgaria, Czech Republic and Hungary

3) Western industrial countries with a very small production and major nuclear energy programs: Germany (West), Sweden and Spain.

For the group 1 countries, an extensive adjustment of the regulatory conditions, the prime objectives of rehabilitation and the applied restoration technologies can be stated, which in part is due to the fact

that uranium production in these countries is carried out mainly by only a few large international mining companies. The specific costs of rehabilitation in these countries lie within a relatively narrow range below or at approx. 0.7 US$/lb U3O8. The UMTRA Title I project with specific costs of 14.7 US$/lb U3O8 is to be seen as an exceptional case (cf. above).

The special conditions which apply to the former Eastern bloc countries (cf. above) are reflected in high specific rehabilitation costs of 1.5 to 3.0 US$/lb U3O8. The WISMUT project with specific costs of 13,9US$/lb U3O8 belongs into this group and is presently being analyzed in more detail.

The countries of the third group are characterized by small projects and small production. The restored sites can be considered as R&D projects. They were state-financed. The range of specific costs varies between 5.0 and 40.4 US$/lb U3O8.

CONCLUSIONS

During the last decades, uranium production was treated with secrecy in many producer countries of the world. This was mainly due to the military-strategic significance, attributed to uranium production. For the time being, a first comprehensive worldwide inventory of uranium mining projects with respect to decommissioning and rehabilitation has been established.

The decommissioning cost analysis reveals a wide variation of specific costs for rehabilitation work. It can be used for the evaluation of the cost structure of ongoing decommissioning projects, e. g. the WISMUT project in Germany. The data may be used for the estimation of future decommissioning costs of mining projects and as a baseline for the specification of financial reserves. Furthermore, the analysis gives valuable background information for optimization measures during mining in order to minimize later decommissioning costs.

Thus, the presented results of the research study can be regarded as an important contribution to the development of future uranium mines as ecologically sound and sustainable mining projects.

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44-2

INTEGRATING REMOVAL ACTIONS AND REMEDIAL ACTIONS - SOIL AND DEBRIS MANAGEMENT AT THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT*

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ABSTRACT

Since 1991, excess soil and debris generated at the Fernald Environmental Management Project (FEMP) have been managed in accordance with the principles contained in a programmatic Removal Action (RvA) Work Plan (WP). This plan provides a sitewide management concept and implementation strategy for improved storage and management of excess soil and debris over the period required to design and construct improved storage facilities. These management principles, however, are no longer consistent with the directions in approved and draft Records of Decision (RODs) and anticipated in draft RODs other decision documents. A new approach has been taken to foster improved management techniques for soil and debris that can be readily incorporated into remedial design/remedial action plans.

In accordance with proposed and selected remedies, the Removal Action Work Plan has been revised to update the soil and debris management approach to recognize recent decisions under the FEMP's Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) process. This paper describes the methods that were applied to address the issues associated with keeping the components of the new work plan field implementable and flexible; this is especially important as remedial design is either in its initial stages or has not been started and final remediation options could not be precluded. A sitewide interim policy that will allow each of the five operable units (OUs) to conduct remedial actions consistent with the respective RODs, yet achieve a consistent methodology for soil and debris management, has been developed in the Revised Work Plan. Under the revised work plan, remedial activities can proceed in advance of the remedial design under the auspices of the removal action; the removal action work plan can subsequently be integrated into the remedial design.

This paper finally identifies applications and lessons learned that evolved from the process of developing the revised removal action work plan, and provides general examples of how other facilities can benefit from this approach.

BACKGROUND

Since production operations at the Fernald site were halted in 1989, removal actions have successfully been used to address threats from the facilities, structures, and contaminants that remain. These actions have been implemented as interim measures until the final remedial actions can fully mitigate the impacts to human health and the environment associated with site contaminants. As an example, Removal Action No. 17 (RvA 17), which is programmatic in nature, was initiated to provide controlled storage of excess contaminated soils and debris generated during maintenance, construction, and removal actions at the FEMP. Specifically, it established procedures for the management and storage of soil and debris.

In establishing the procedures, thresholds were established for managing the soil in distinct piles. These initial thresholds were:

Category I Radiologically Contaminated Soils

- <100 pCi/g total uranium (U)
- <5 pCi/g total radium (Ra)
- <50 pCi/g total thorium (Th)
- No Resource Conservation and Recovery Act (RCRA) hazardous waste or polychlorinated biphenyl (PCBs) contamination

Category II Radiologically Contaminated Soils

- -> 100 pCi/g total U
- -> 5 pCi/g total Ra
- -> 50 pCi/g total Th

Category I soils could be stored in stockpiles that were covered with tarpaulins. Category II soils could be stored in stockpiles that were placed on tarpaulins and also covered with tarpaulins. Soils containing hazardous waste or PCBs were to be containerized and placed in approved storage. The area of contiguous contamination (AOC) concept that is employed under CERCLA was incorporated into the work plan such that each operable unit was to maintain its own stockpiles.

RvA 17 was to be conducted in two phases. In Phase I, the stockpiles were to be established in accordance with the above thresholds and were to remain in existence until new storage facilities could be constructed under Phase II. Once the new storage facilities were in place, the soil would be moved into them until final disposition decisions were made.

WHY CHANGE?

Since the implementation of RvA 17, several events have occurred. First, with the concurrence of the regulatory agencies, the FEMP determined that new storage facilities were not needed. The ability to dispose of some debris under the Record of Decision for Interim Remedial Action for OU3 (former production area decontamination and dismantling [D&D]) coupled with projected increases in storage capacity resulting from completion of facility D&D projects and accelerated shipments of existing and legacy wastes resulted in a reduction of scope for RvA 17 activities. Second, the RODs for each of the OUs have either been issued or are to be issued within the near term. Most notably, the ROD for OU2 (other waste units, including flyash piles, lime sludge ponds, and solid waste landfill) included the construction of an on-site disposal facility (OSDF) and associated waste acceptance criteria (WAC); it also adopted final

remediation levels (FRLs) for contaminated soil located within OU2. The OU5 (environmental media) ROD expanded the OSDF to accept additional materials, and expanded the waste acceptance criteria for the soil; additional FRLs were developed for contaminated soils within the OU5 area. The WAC were not consistent with the thresholds that existed in RvA 17. The OU5 ROD also adopted a sitewide Corrective Action Management Unit for soil remediation which includes the OSDF to replace the AOC concept that was previously utilized. Finally, the draft OU3 Remedial Investigation/Feasibility Study for final disposition of D&D debris was issued, which recommends disposal of the D&D debris in the OSDF and establishes appropriate WAC for the debris.

NEW APPROACH

As the RODs were moving towards approval, it was recognized that management of soil and debris under RvA 17 needed to be consistent with the anticipated remedial actions. Additionally, it was necessary to integrate the implementation of approved or anticipated RODs and individual remedial action plans with the management approach for soil and debris. Therefore, a team representing each of the key projects, the environmental compliance program, the environmental monitoring program, construction, and the waste management program was assembled to develop a revised removal action work plan that attained that consistency.

The key objectives established by the team included the need for the work plan to be practical (i.e., field-implementable) and the need for an easy transition for the existing soil and debris storage into the new concept. This involved integrated planning in recognition of the final disposition options for soil and debris. A flow chart was developed to aid in the development of the strategy (see Fig. 1).

Fig. 1

Information from the selected remedies or preferred alternatives for each operable unit will determine the potential to combine and reduce soil staging or storage areas. The needed information includes knowledge of:

Planned final disposition (e.g., on-property or off-site disposal);

Location and mode of transport to off-site disposal facility(ies) as applicable;

Total number of soil staging/storage areas projected during remediation;

Projection of on-site treatment requirements for on-property/off-site disposal; and

Types of staging areas required (e.g., stockpiles, container storage areas, construction of new facilities, use of existing facilities/structures).

The strategy for segregating or combining soil within an operable unit or from several operable units creates a commitment to manage each staging area according to the common planned disposition of the soil in that staging area. Generally, these criteria are based on the WAC for the designated disposal facility (see Table I); further, location criteria are based on general environmental protection requirements (e.g., floodplains/wetlands standards) and the designated FRLs for that location. Additionally, several criteria will be required for management practices for all soil stockpiles, such as run-on and run-off controls.

Table I

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The key criteria for debris management revolve around the final disposition and the category of debris. Several categories of debris were

identified under the RI/FS for OU3 (see Table II); these will be used by other OUs which may generate debris also. Some of these categories were administratively precluded from on-site disposal based on process knowledge of the levels of contamination; the other categories are to be designated for storage based on final disposition and characterization data. Debris to be dispositioned off-site will be containerized at the point of generation and shipped off-site as soon as practical. For debris proposed to be disposed on-property, some debris will be bulk-staged to permit the most effective handling of these media. In cases where bulk staging is desired, the debris will be managed to assure minimization of airborne emissions, and staging will occur to assure control of run-off. These debris will be staged in a manner to minimize double handling, minimize costs by optimizing container use, and minimize labor associated with maintenance. Use of bulk storage will not preclude any disposition options, including reuse/recycling. Material will be stockpiled on an existing storage pad or on the foundations of dismantled buildings.

Table II

Another important consideration in the development of the RvA 17 WP was the duration of the plan. Since the FEMP was rapidly approaching the completion of all RODs and would soon be moving fully into remedial action, there was a recognized need to limit the duration of the removal actions consistent with NCP criteria. The RvA 17 WP will remain in effect until the OSDF is in operation and the appropriate remedial action plans are implemented. It is anticipated that the remedial action plans could utilize the work plan as the basis for soil and debris management actions.

One final aspect of the RvA 17 WP was the incorporation of a sitewide non-aqueous investigative derived waste (IDW) policy. The IDW include drilling muds and cuttings from soil borings and well installation, soil, debris, and other materials from the collection of samples; residues (e.g., ash, spent carbon) from testing of treatment technologies and treatment systems; and contaminated personal protective equipment used during investigations. The IDW policy was included in this document because the material is similar to that addressed in the soil and debris management plans. Additionally, two separate policies existed for non-aqueous IDW, and this provided an opportunity to merge them and assure consistency. The established policy states that the preferred management options for non-aqueous IDW are to return the IDW to or near its source, if possible, or to manage it in accordance with the principles delineated in the soil and debris management plans.

APPLICATION/CONCLUSIONS/LESSONS LEARNED

The RvA 17 WP was conditionally approved with comment by the U.S. Environmental Protection Agency - Region 5 on first review. The Ohio Environmental Protection Agency had several comments that focused on clarification of responsibilities and integrated planning. There are several lessons learned that can be applied from the process of developing this revised removal action work plan:

Early discussions with the regulators on the approach to be used is very useful. By addressing the potentially controversial issues (primarily open bulk storage), acceptance of the concept was obtained early and led to expedited approval.

Teaming and integration is important. This work plan addressed all key issues because the important internal stakeholders participated in the development of the work plan. Additionally, a workshop approach with

participation from all affected parties within the FEMP was used to resolve all comments that were received during the initial internal review process.

Maintain flexibility. Since the site is just commencing the remedial action process, it was important to assure that remedial action options that would be defined in the remedial design/remedial action plans would not be precluded; at the same time, the approach taken in the removal action work plan had to be consistent with the direction being taken in the various RODs.

Maintain protectiveness, but use the easiest, least cost method for the short term. This was especially true in the storage of debris. Storage containers are not inexpensive, and need to be used judiciously. By demonstrating that outside bulk storage is protective, significant costs could be saved.

Use characterization methods that are field-implementable and appropriate for the waste acceptance criteria. Rely on process knowledge to the extent practicable. Where process knowledge is not sufficient, use field screening techniques; however, these field screening methodologies must be capable of measuring at the levels of the WAC and FRLs, as appropriate.

APPLICABILITY TO OTHER SITES

The CERCLA concepts that were incorporated into the RvA 17 Work Plan can be used by other sites in trying to accomplish expedited response actions using a removal action while awaiting remedial action RODs. For example, interim actions could be used to expedite remediation by excavating contaminated soil from a surface impoundment or managing it for final disposition prior to the final ROD. Additionally, the number of interim storage areas for debris and soil could be minimized and consolidated by identifying early on the final disposition options and then characterizing the material accordingly. Although this paper addresses the CERCLA arena, the same concepts are applicable to the RCRA process - interim measures (analogous to CERCLA removal actions) and corrective actions (analogous to CERCLA remedial actions).

44-3

EFFECTIVE USE OF RISK ASSESSMENTS AND THE PUBLIC COMMENT PROCESS TO ACHIEVE ACCEPTABLE REMEDIATION GOALS FOR MERCURY-CONTAMINATED SITES*

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ABSTRACT

The effective use of scientific research data and the willingness to hear and involve the public in a manner that is in keeping with regulatory requirements are two important challenges affecting the cleanup of the Lower East Fork Poplar Creek floodplainsa mercury-contaminated site in Oak Ridge, Tennessee. As a result of recalculating risk levels using new information, the remediation goals and the cleanup strategy for the Lower East Fork Poplar Creek floodplains have been significantly changed. These changes reflect an important reduction in cleanup costs while ensuring protection of human health and the environment.

Use of the risk assessment process should yield an effective evaluation of information on the adverse health effects to human receptors exposed to hazardous and radioactive materials. The keys to the risk assessment's effectiveness are having an accurate understanding of the contaminant and correctly applying the remediation goal methodology. Determining effectiveness is also related future land use of the contaminated site, and future land uses will depend heavily upon the local community. Involving the public in the evaluations of the contamination and risk methodology has resulted in successfully achieving an acceptable remediation goal for all stakeholders. This project and its stakeholders have made the risk assessment process more effective by better defining the contaminant and adjusting assessment parameters in a way that affords adequate protection for human health and the environment while building stakeholder acceptance.

As a result of this synergism, the remediation goal initially set at 50 parts per million (ppm) of mercury has been changed to 400 ppm, resulting in significant reductions in both the destruction of the floodplain landscape and project costs. The volume of soils to be excavated has been decreased from 1 million cubic yards to 25 thousand cubic yards, and the cost has been reduced from approximately $1 billion to less than $20 million. The Record of Decision for Lower East Fork Poplar Creek was approved in August 1995.

BACKGROUND

The contamination of Lower East Fork Poplar Creek (LEFPC) occurred during the 1950s and 1960s when mercury was used in the manufacturing process of nuclear weapons components at the Oak Ridge Y-12 Plant in Oak Ridge, Tennessee. The mercury-contaminated waste water was initially discharged into the upper portion of East Fork Poplar Creek. Figure 1 shows historic mercury releases in the upper portion of East Fork Poplar Creek from 1955 to 1994. The contamination extends along the floodplain of this 14-mile-long creek that flows through U.S. Department of Energy (DOE), residential, and commercial property areas. The remediation of LEFPC has posed many challenges to the technical participants, regulators, and public.

Fig. 1

The LEFPC site includes two distinct but overlapping areas: LEFPC and the Sewer Line Beltway (Fig. 2). LEFPC flows 14 miles from Lake Reality at the Y-12 Plant to its confluence with Poplar Creek near the Oak Ridge K-25 Site. LEFPC flows through an area that hosts a range of human activities and land uses. The site includes creek sediment and soils, which make up the creek's 100-year floodplain. The Sewer Line Beltway consists of 10 miles of sewer lines, with one portion within the floodplain of LEFPC and two branches in the city of Oak Ridge. The Oak Ridge Sewer Line Beltway was studied because soils from the creek bank were used as excavation fill for the sewer line during the early 1980s. Because the soils around the Sewer Line Beltway present no significant risk, the beltway is not discussed further.

Fig. 2

The project began with characterization studies that charted the effects of mercury contamination on the environment. Contamination of the LEFPC floodplain can be understood by using a conceptual model for contaminant transport. The initial premise is that soil contamination in the floodplain is closely linked to hydrologic events. Contaminants from the Y-12 Plant were washed down LEFPC during high-flow conditions following

rainstorms. At least some contaminants were adsorbed onto sediment particles and were transported downstream in a suspended phase. Other contaminants were transported in a dissolved phase. During flooding, the creek overflows its banks and spreads out across its floodplain, depositing contaminated sediments on vegetation and the land. Based on this conceptual model, the remedial investigation focused on the evaluation of surface water, creek sediments, floodplain soils, and groundwater as potentially affected media. Because of the continued releases from the Y-12 Plant, the mercury in the surface water was not addressed in this remediation effort. However, the mercury contamination eventually settled into the creek sediment. During past years when the contamination originally occurred, naturally occurring events, such as flooding and erosion, created a change in the locations of the contamination. Based on the results of the investigations, the floodplain soils pose the greatest risk to human health and the environment and are the basis for the remedial action.

DEVELOPMENT OF REMEDIATION GOALS

The critical human health exposure pathway at the site was determined to be soil ingestion by children, and the initial cleanup goal was calculated at 50 ppm of mercury (soils with greater than 50 ppm mercury would be removed; soils with less than 50 ppm would remain). However, the initial risk assessment assumed all the mercury in the soils was mercuric chloride (the mercury species on which the toxicity reference dose was based), a soluble mercury compound not expected to be present in a floodplain frequently saturated with water. Previous investigations had suggested that mercury in the LEFPC floodplain was less soluble and therefore less bioavailable than mercuric chloride, possibly making the results of the risk assessment unduly conservative. Using a sequential extraction procedure for determining the species of mercury in soil, Revis et al. (6) estimated that approximately 85% of the mercury in the soils had been converted to mercuric sulfide as consequence of microbial activity in the reducing conditions of the soil. Subsequently, Revis et al. (7) also showed mercuric sulfide absorbed gastrointestinally to a lesser extent than mercuric chloride in mice feeding studies. An examination of 20 soils using optical and scanning electron microscopy also revealed a consistent elemental association between mercury and sulfur, further bolstering the idea that the mercury in the soils had been converted to mercuric sulfide. However, using a different sequential extraction procedure (5), elemental mercury was detected to a larger extent than mercuric sulfide (4). Barnett et al. (1) showed the use of the two previously used sequential extraction procedures (as well as a third) on the same set of soils produced inconsistent results, prompting questions about the utility of current sequential extraction procedures to unequivocally provide information about the mercury species in soil. For this reason, a subsequent effort was initiated to determine whether or not crystalline mercuric sulfide was present in soil. Using x-ray and electron beam methods, Barnett et al. (1) confirmed the presence of submicron crystals of mercuric sulfide in the form of metacinnabar. Because of the difficulty in extrapolating particle- and soil-specific speciation information and the lack of a reference dose for mercuric sulfide, a bioavailability study designed to measure the amount of mercury available for absorption in a child's digestive tract was conducted to provide direct (independent of speciation) information for input to the risk assessment (2). The bioavailability study used soil

leaching tests to simulate the human digestive system and determine the fraction of mercury in LEFPC soils potentially available for absorption in the human digestive tract. The study was conducted on a suite of soils from the floodplain as well as some pure mercury compounds for comparison purposes. Soil samples were collected at 2 depths from 10 sites (for a total of 20 samples) to represent a range of environmental conditions in the floodplain. Surface samples were collected within 3 inches of the surface. Deeper samples were collected in layers historically associated with the highest concentration of mercury in the floodplain. A 740-mg-portion of each air-dried soil was added to 1 liter of distilled water adjusted to pH 2.5 and allowed to leach for 4 hours, after which a filtered subsample was taken for mercury analysis. This treatment simulated the stomach. The subsample volume was replaced with distilled water, and the remaining leachate was pH adjusted to 6.5 and allowed to leach for 4 hours, after which a second subsample was filtered and collected for mercury analysis. This treatment simulated the remainder of the gastrointestinal tract.

The results of the leaching study for the 20 soil samples are shown in Table I. The total percent soluble mercury is simply the sum of the percent soluble at both pH values and represents a worst-case scenario, since some of the mercury in the soil is undoubtedly soluble at both pH values and is double counted in the simple summation. In 19 of the 20 samples, the mercury was not significantly (as a percentage) extracted at either pH. Less than 5% of the mercury was solubilized in 15 of the samples. Total soluble mercury in the 19 samples ranged from 0.3 to 14.2%, with an average of 3.2%. For one sample, however, there was a total of 45.9% soluble mercury. The high leachate concentrations for this sample are thought to reflect differences in the speciation of mercury in the sample, as this was the only sample among 20 which exhibited detectable mercury vapor in the sample headspace, which is not characteristic of mercuric sulfide (8). This sample was also the only deep, mercury-concentrated sample where total mercury and total sulfur were not well correlated. Including this sample, the average percent of mercury leached from the 20 samples was 5.3%. Attempts at more realistically simulating the human digestive system (e.g., by raising the leaching temperature to body temperature and using deoxycholic acid, a common constituent of the human digestive system) did not significantly affect the results.

Table I

Mercuric chloride and mercuric sulfide (cinnabar and metacinnabar) were also subjected to the leaching procedure. Less than 1% of the mercuric sulfide samples, both cinnabar and metacinnabar, leached in the procedure. In contrast, all of the mercuric chloride sample dissolved, producing leachate concentrations almost 1,000 times higher than the highest soil leachate. Although the procedure is a simple representation of a complex system such as the human digestive system, the solubility and hence bioavailability of mercury in LEFPC soils is obviously substantially less than pure mercuric chloride. As a result, use of a reference dose for mercuric chloride in the LEFPC risk assessment without incorporating a corresponding bioavailability factor would be unduly conservative.

The solubility information from this study was then modeled using Monte Carlo techniques, which produced a probability distribution of mercury bioavailability (3). After reviewing this information with regulators and

the general public, a bioavailability of 10% was selected. Using a bioavailability of 10%, the cleanup goal was increased from 50 ppm (based on the default value of 100%bioavailability) to 400 ppm of mercury.
The data on bioaccumulation of mercury from LEFPC mercury-contaminated soils and other soil/body burden relationships were integrated and developed into an ecological evaluation. The most sensitive group for protection is the mid-level predator. A risk management decision was made to provide the necessary level of protection, which took into account the habitat disturbance and mortality which would result from cleanup activities, balanced against the contaminant exposure risk. Mercury contamination of greater than 400 ppm provides the necessary level of protection to the environment.

IMPACTS

As the country became more environmentally conscious, federal laws were enacted to direct cleanup of contaminated sites. Priority was given to the LEFPC site because the contamination exceeded federal facility boundaries and posed a risk of uncontrolled conditions and exposure to the public. In December 1989, the Oak Ridge Reservation was placed on the U.S. Environmental Protection Agency's (EPA's) National Priorities List. This meant the cleanup of sites contaminated by DOE operations on and off the reservation would be conducted under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), which requires public participation in the cleanup process. DOE recognized the high visibility of this project and decided to involve the public early. The Record of Decision describes the selected remedy to include excavation of the contaminated media, backfilling the evacuated areas, and disposing the contaminated media in a permitted landfill at the Y-12 Plant.
Several public meetings were held to provide information to area citizens and hear their concerns about LEFPC. As the work continued, workshops were held to explain the process to the public and encourage participation by community members in selecting a preferred alternative. Also, several workshops were held with citizens who own property along the creek. The public's involvement at Superfund (CERCLA) sites typically consists of public meetings and public comment periods; however, these come late in the remedial decision process. To gain more meaningful participation early on, DOE decided to form a voluntary, informal group in May 1993 called the Citizen's Working Group (CWG). Because of its visibility and timing, the LEFPC environmental restoration project was chosen to launch a CWG as a pilot program. Volunteers for the group were solicited through advertisements and announcements to form a project-specific, broadly representative group of concerned citizens. Their task was to give DOE feedback about the proposed remedial alternatives for the creek's floodplain soils. The LEFPC CWG was not formed to be a decision-making or consensus-building group.
The CWG reviewed proposed remedial alternatives, identified concerns, and suggested new options. The group's activities were representative of the CERCLA process; therefore, they were able to maximize their ability to help DOE gain new perspectives about potential remedial alternatives under consideration. In addition, they were in a position to introduce new alternatives. Input from the group helped DOE better understand citizens' concerns related to remediation of the floodplain soils and the potential impacts to the community.
The high cost of performing environmental restoration activities resurfaced throughout the characterization and decision phase for the

LEFPC project. The public strongly emphasized the cost to remediate versus the protection of human health and the environment. Figure 3 shows a cost comparison from the feasibility study, baseline, and value engineering study. The costs reflect the estimates to perform remedial design and remedial action activities. As a result of public concern, a value engineering study was performed to give a critical examination of the preferred alternative. Each activity in the preferred alternative was examined for innovative ideas and potential performance improvements. Cost estimates were generated from a proposed list of new and modified approaches. The majority of the new approaches proved to be cost effective.

Fig. 3

CONCLUSIONS

The reason for understanding and assessing the risks for human health and environment is to derive a remediation goal. The remediation goal, 400 ppm, has been determined to be the desired in-point of concentration of mercury contaminant.. Mercury can exist in many forms, some more toxic than others. Public input, required by law, ensures that communities and people affected by past environmental practices will have a role in selecting the remedies for the problem. All stakeholders have a significant role in balancing the goal of protecting human health and the environment with remediation costs. The difficulty comes into play when a data set does not exist on the effects to the human organs and therefore must be extrapolated from animal/laboratory experimentation. At LEFPC, the effects of mercury in a soil matrix would be of greatest risk to children, based on the contaminant (mercury) and the exposure pathway (ingestion). To significantly reduce the risk of ingestion of mercury-contaminated soils, soils with greater than 400 ppm mercury will be removed and handled in an environmentally acceptable manner by excavating the contaminated soil, disposing it in a permitted landfill on the Oak Ridge Reservation, backfilling the excavated areas with clean soil, and restoring the disturbed site to its condition before remediation work.

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44-5

PILOT SCALE TESTING OF SOIL WASHING ON LOW LEVEL RADIOACTIVELY CONTAMINATED SOILS

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ABSTRACT

Several sites in the Formerly Utilized Sites Remedial Action Program (FUSRAP) have shown, in laboratory studies, to be amenable to volume reduction treatment by particle size separation. This treatment using water as a carrier media is often called soil washing. The current practice, at FUSRAP sites, of excavation and hauling to a disposal facility is both expensive and produces a large volume of soil that is potentially clean but is handled with the contaminated soil. Bechtel National Inc. (BNI) has performed pilot-scale soil washing on select FUSRAP soil to demonstrate treatment/volume reduction.

The soils examined for soil washing were from the New Jersey sites, and each had undergone a laboratory characterization study that had shown that radiological contaminants were concentrated in the fine silt and clay fractions of the soil. The characterization treatment goal was 5 pCi/g for residential areas, and 15 pCi/g for industrial area. Removal of those fine fractions could produce coarse fractions of soil well below the 5/15 criteria with volume reductions of 50-80%.

To demonstrate scale-up and potential full-scale remediation, a pilot-plant study was performed using a soil washing pilot-plant originally designed by the Environmental Protection Agency (EPA) for testing on Montclair and Glenridge, New Jersey soils. This pilot plant named VORCE (Volume Reduction/Chemical Extraction) was modified to meet the specific requirements for treatment of the FUSRAP soils. After a series of tests on clean soils to develop operating parameters and system performance the machine was used to treat Maywood, New Jersey Interim Storage Site (MISS) pile soils. All tests performed to date indicate soil washing is very promising for treatment. Further tests are continuing on soils from other sites and locations.

INTRODUCTION

The Department of Energy's (DOE) Formerly Utilized Remedial Action Program (FUSRAP) consists of 46 sites in 14 states. These sites were originally used for manufacture or processing of ores and metals for support of the atomic bomb development. FUSRAP sites contain large volumes of soils (and other media) contaminated by low level radioactive contaminants particularly, thorium, radium and uranium, and in some instances metals and other hazardous contaminants.

Treatment options for these contaminated soils are limited. Options are: 1) leave in place with either institutional controls and/or engineered controls, 2) remove and dispose in an approved disposal facility or 3) treat to reduce the volume. The first option has potential for community

resistance; option two is expensive; and option three has only limited data to support its selection. From characterization studies of site soils several of the large volume sites have been identified as potential candidate sites amenable to treatment using soil washing technology.

The original concepts of soil washing have their history in the mining and metals industries. Mining processes have been used for years to concentrate metal particles for extraction as metal concentrates. Much of this equipment, i.e., trommels, screw classifiers, attrition mills, hydrocyclones, etc., originated in the mining industry.

Soil is comprised of various particle size fractions and most often the contaminates are concentrated in one size fraction. Most often this fraction is the finer silts and clays (50 mm or less). Removal of this fraction facilitates the potential for volume reduction and cheaper disposal costs. Increased surface area, cationic exchange potential, and the innate shape of these particles are important in the attraction of contaminants to the fine fraction.

Soil washing separates soils by particle sizes producing clean and contaminated fractions, using water as the fluidizing media. The clean fractions can be recombined, augmented if required and used as backfill. The contaminated fraction concentrated in a smaller volume can be shipped to an approved disposal facility for final disposition. Clean streams by FUSRAP standards are normally site dependent but 5/15 pCi/g above background is normally accepted, with 5 pCi/g for residential area soils and 15 pCi/g for industrial area soils. Results obtained from two test campaigns are discussed in this paper.

HISTORY

Two FUSRAP sites, Maywood and Wayne, located in New Jersey, have shown from characterization studies to have soil distributions with potential volume reductions of 50-80 percent. These soils follow the traditional theory of the contaminants (radionuclides) attract to the fines. Removal of this fraction, which is between 20-50 percent of the total soil, dependent upon size cut, provides a significant volume of soil that will meet clean-up criteria.

In order to verify the results of the characterization study, a pilot test was to be planned. This test would provide valuable data in determining the volume reduction and the associated cost if the soils from the Wayne and Maywood sites were treated. In order to perform this study DOE needed a pilot plant. At this time the United States Environmental Protection Agency (EPA) National Air and Radiation Engineering Laboratory (NAREL) had a machine that was available to DOE . This machine, called the Volume Reduction/Chemical Extraction (VORCE) machine, was used by NAREL for a soil washing treatment demonstration for the radium contaminated Montclair and Glenridge, New Jersey sites. The machine was transferred from EPA to DOE for use. Bechtel National, under recommendations by the United States Bureau of Mines (BOM) modified the system to meet the anticipated treatment goals. The system is a nominal 0.5 to 1 ton per hour system incorporating a series of unit operations to force separation of fine particles from the coarser oversize fractions. Significant modifications were made to feed system, attrition mills, and the fines circuit to better provide the operations required to achieve the treatment goals.

Originally, the VORCE plant was intended for mobilization to the Wayne, New Jersey site for testing and demonstration, during the fall of 1994, but community pressures precluded performing the study in New Jersey.

After evaluating several DOE locations, the Oak Ridge, K-25 site was chosen as the location for the tests. Soil was loaded and transported by intermodal container to Tennessee for the soil washing studies. Initially, equipment evaluation of the machine using clean soil was performed in the fall of 1994. Operations were suspended during winter and restarted in the spring of 1995 when equipment evaluation was continued. The original site for the machine check out was not suitable for performing radiological tests. Therefore, the system was mobilized to another area of the facility in the summer of 1995. The current location where tests on radiological soils are being conducted are constructed with a lined engineering treatment pad.

SYSTEM DESCRIPTION

The final VORCE configuration is as follows with a schematic shown in Fig. 1. The soil is fed through a 2 or 4 inch static grizzly into a feed hopper. The soil in the hopper is conveyed to the first unit process, the trommel screen, by means of a drag flite conveyor. Fine fraction soil from the trommel is collected in a sump, while the coarse fraction is discharged to a screw classifier. The screw classifier overflow is collected in the same sump as the trommel overflow. The coarse fraction from the screw classifier is discharged into an attrition mill. The attrition mill discharges into another screw classifier. Again the overflow from the second screw classifier is collected in a sump. The sump contents are fed to a hydrocyclone. The underflow from the cyclone is discharged into a hydraulic classifier. Overflow from the hydraulic classifier is discharged to a clarifier and the sludge sent to a filter press. The underflow from the hydraulic classifier is sent to a vibratory screen for dewatering. Product streams from the process are trommel oversize, secondary classifier oversize, vibratory screen oversize, and filter cake.

Fig. 1

SOIL WASHING PROCESS AND DESCRIPTION

The term soil washing is most often used to describe a series of treatment operations that either separate soils into their various particle size fractions using water (or chemicals as the carrier medium) or chemically extract the contaminant from the soil matrix. The VORCE soil washing pilot plant is strictly physical particle size separation using water as the carrier medium. The VORCE plant uses a series of unit operations to physically deagglomerate the soil and separate it into specific size fractions

A description of the process is as follows: soil is loaded onto either a 2-inch or 4-inch grizzly. The oversize material is removed by hand; the undersize passes through the grizzly into a feed hopper.

The material in the feed hopper is fed, by a drag flite conveyor, to a trommel. In the trommel deagglomeration of the soil particles occurs, and soil containing contaminants is washed from the large oversize particles. Soil particles (gravel) between 1/4-inch and 2- or 4-inches are washed clean using high pressure sprays, and report to a discharge chute as a clean fraction. The undersize material reports to a screw classifier. The screw classifier separates soil particles based on Stokes Law which states that a particle in a fluid settles at a rate that is the function of the particle's diameter. The smaller particles (with lower settling rates) overflow and are collected in an sump. A size cut of approximately 60 mesh (250 m) is performed at the screw classifier. The larger particles are removed by a screw and dewatered. The "coarse" sand is sent

to an attrition mill where the action of the mill causes interparticle abrasion to remove adhered fines from the coarse sand particles. The soil discharges from the attrition mill to another screw classifier where another 60 mesh particle size cut is made. The overflow from the second screw classifier reports to the sump, and the underflow reports to a discharge bin as a clean fraction.

The fine products from the coarse size operations collected in the sump are pumped to a hydrocyclone. The hydrocyclone is used for two purposes: 1) make a fine cut of 200 mesh (75 mm); and 2) to dewater the feed to the next unit operation. The fines overflow is collected in another sump, and the coarse underflow is fed to the hydraulic classifier. The hydraulic classifier is used for performing a fine cut between 200-325 mesh (75-45 mm). The hydrocyclone performs separations based upon centrifugal force, while the hydraulic classifier uses a combination of Stokes Law and hindered settling to make the cuts.

The fines collected from the hydraulic classifier are collected in the same sump as those from the hydrocyclone. The coarse underflow fraction is fed to a dewatering screen. The screen oversize is collected as a clean product. The fines and water from the screen are collected in the same sump as the other fines fractions. These fines are flocculated using polymer and fed to a clarifier. The flocculated fines are collected in the bottom of the clarifier and the clear water overflows into the process make-up water tank, thus providing recycling of the water.

The sludge in the bottom of the clarifier is them pumped to a holding tank prior to feeding to a filter press for final dewatering. The filter cake contains the contaminated fines, and the filtrate is recycled back to the clarifier for reuse.

SITE PREPARATION

With community pressures limiting soil washing or treatment in New Jersey, it was necessary to find a site that was acceptable for the soil washing tests. As noted earlier the K-25 in Oak Ridge was deemed best for this application. Reasons were: 1) close proximity to Bechtel, 2) could be operated under Lockheed Martin's Center for Environmental Technology as a technology demonstration program operating under their licenses or permits as required. In order to minimize uncontrolled releases of contaminated material from the soil washer an engineered and controlled site was necessary.

The radiological site was prepared near the existing K-31 building on the K-25 site. The location was a grassy area that required installation of engineering controls to minimize the potential spread of contamination. These controls consisted of a geomembrane with a double liner under the process area and a single liner under the non-process area. Gravel was placed in the area to provide drainage. Separate sumps were installed to collect rain water and spilled process water, and to keep them separate. Rain water would be sampled and segregated from process water for in-house discharge. Contaminated water was collected in a sump and recycled into the machine. Contingencies were developed so that the entire machine could be dumped and collected in a storage tank if required. Segregation of water provided valuable waste minimization.

Since the machine lacked performance testing, especially with the new modifications, it was necessary to evaluate the machine with clean soil, which could be done concurrently to the rad site preparation. The machine was set up to do clean soil performance testing in a storage area near the old powerhouse area of K-25. With no contamination potential from the

clean soil this was acceptable until contaminated soil processing. Several campaigns were performed at this location using various clean soil types to evaluate process efficiency, and to develop process parameters. Once the radiological site was completed the machine was demobilized and remobilized to the new site.

SOIL CHARACTERIZATION STUDIES

As noted above, two soils were characterized to show volume reduction by soil washing was viable. These soils were obtained from the Maywood Interim Storage Site pile and the Wayne Interim Storage Site pile. The data, shown in Fig. 2 for Maywood and Wayne show that volume reduction by particle size could produce volume reductions in the 60-80 percent range depending upon the size fraction removed.

Further studies performed by Oak Ridge National Laboratory (ORNL) also provided a soil sample from within the Oak Ridge reservation that also provided a soil with properties that would support soil washing as a volume reduction technology. This soil, deemed the CSX soil, was railroad ballast and gravel contaminated with cesium from spills near the rail yards on the Oak Ridge reservation.

Fig. 2

TREATMENT CAMPAIGNS

In order to demonstrate the effectivity of soil washing on Maywood pile soil, approximately 100 tons of soil from the pile was loaded into six intermodal containers and shipped to Tennessee for testing. No intent was made to test all 100 tons, but contingency amounts were built into the quantity ordered. The test consisted of operational settings based upon "real-time" experience, and allowing the machine to reach steady state prior to collecting samples of the feed and four output streams. Field adjustments were made as necessary to ensure proper performance. Four different series of samples were collected and analyzed for radiological content, and particle size. Isotopes of concern were Ra-226, Th-232, and U-238.

Soil was fed to the hopper using a front-end loader (ref. Fig. 1). Oversize material, greater than 6 inches, was remove by hand, as was any material collected on top of the grizzly. The feed soil sample was collected from each bucket for later compositing and mixing.

The feed rate was set at 704 lb./hr. This number appears low based on the design value of 2 tons/hour for VORCE. This lower figure was due to the mismatch in sizes of equipment, which limited the maximum quantity that can be fed to the machine without causing "sanding out" of the primary classifier. When "sanding out" occurs the separation efficiency for that particular unit process is reduced from lack of settling height.

The processed material was collected and volumes recorded, as was feed, for mass balance information. Fractions collected were as follows: 1) >1/4-inch, 2) 1/4-inch to 60 mesh, 3) 60 mesh to 200 mesh, and 4) <200 mesh.

RESULTS

Studies to date have been performed on the Maywood pile soils and the CSX soils. The studies incorporated running the machine at a steady state condition and collecting samples of the feed and all output streams; analyzing for radionuclides of concern and particle size. Data from the feed indicate a variation from the soil characterized in the previous bench scale work. Figure 3 shows the particle size distribution average for the feed soil to the plant during the Maywood study. Figure 4 shows the average rad concentration by isotope for the feed soil and the mass

reduction generated by the machine. The results of 98.6% mass recovery and 63% mass reduction track well with the theoretical results of the feed soil.

The CSX soil results are pending at this time, but preliminary data from the actual tests have shown that a 86+% mass reduction is obtainable. Much of the cesium is water soluble (concentration ranges in the water of 3300 pCi/l) necessitating water treatment prior to final disposition. Cost data was recorded but has not been reduced at this time, but it is expected to be below the costs for removal and disposal in an approved disposal facility. How much cannot be made until the data is evaluated.

CONCLUSIONS AND THE FUTURE

Initial tests indicate that soil washing is promising for certain FUSRAP sites. Sites with characteristics that show particle size separation can produce volume reduction are amenable to this treatment process. The machine is scheduled to test more soil beginning in early spring and concluding in mid-summer. A detailed report will be generated to capture the study and its results.

Fig. 3

Fig. 4

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CHEMICAL EXTRACTION STUDIES OF SOILS CONTAMINATED WITH RADIUM-226 AND THORIUM-230

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ABSTRACT

Soil samples from a Superfund Site contained elevated concentrations of radium-226 and thorium-230. The soil allegedly was contaminated with mill tailings from a chemical processing plant that extracted radium from carnotite ores.

Chemical extraction studies of soil samples were performed using solutions containing nitric acid, hydrochloric acid, hexametaphosphate, or combinations of hydrochloric acid and sodium chloride in single, multiple, and sequential extractions. The extractions were examined under various conditions of time, concentration, liquid-to-solid ratio, temperature, and rotational velocity. The products were analyzed for radium-226 and thorium-230. A full quadratic model of the bench-scale extraction studies for pilot-plant design considerations was examined to

determine any important interactions of temperature, concentration of extractant, and liquid-to-solid ratios.

Chemical extraction studies of soil samples demonstrated that dilute nitric acid solutions are effective in removing over 90 percent of the radium and thorium activity. Dilute hydrochloric acid solutions remove over 90 percent of the radium but less thorium. However, a sequential extraction procedure, employing hydrochloric acid followed by hexametaphosphate solutions, is equally effective in removing both radium and thorium. Hydrochloric acid could be replaced by a hydrochloric acid/sodium chloride solution, reducing the hydrochloric acid requirements for the process. A full quadratic model of the bench-scale extraction studies data revealed several important interactions between temperature, concentration of extractant, and liquid-to-solid ratios. Optimal values for these significant variables were determined using the model.

Physical separation of the whole-soil samples by particle sizing isolated a significant quantity of a soil fraction larger then 50-mesh size (0.30-mm) (+50) with radium-226 and thorium-230 concentrations less than 15 pCi/g. This fraction was extracted under optimum conditions to assess chemical treatment as a remediation technology to reduce the concentration of the contaminants to less than 5 pCi/g above background. Chemical extraction of the +50 soil fraction containing less than 15 pCi/g radium-226 and thorium-230 using the sequential extraction method described above reduces the concentrations to approximately 5 pCi/g. A combination of physical separation of whole soil followed by chemical extraction of a select particle-size fraction may offer a viable method for site remediation of radionuclide-contaminated sites by volume reduction.

INTRODUCTION

There are well over twenty sites in the United States containing soils contaminated with radium-226 and thorium-230. The Superfund site, where the soil samples used in this study were taken, was allegedly contaminated with tailings from a chemical processing plant that extracted radium from carnotite ores early in this century (1). As a result of using these tailings as landfill during construction, approximately 300,000 yd3 of soil over 95 acres were contaminated. Buildings on the site were affected to one degree or another by elevated levels of gamma radiation as well as radon-222 gas. The significant contaminants, producing the gamma radiation and radon gas, were radium-226 and thorium-230.

The contamination was the result of the presence of process tailings containing barium-radium sulfate precipitates, incompletely processed ores, and other radiominerals that are mixed with the native soils (2). The volume of contaminated soil can be reduced approximately forty percent by a process using vigorous washing and subsequent size separation with water, since the major portion of the radionuclide activity was in the smaller-sized soil particles (-50-mesh) (3). The resulting soil product has a specific activity of about 15 pCi/g radium-226. A pilot-scale operational system that incorporates the laboratory process was designed and used to test the remediation effort (4,5). Studies on uranium-mill tailings from other contaminated sites indicated that volume reduction by chemical extraction was feasible (6). A 1989 study (7) of soils contaminated with radium and thorium revealed that extraction with nitric acid removed approximately 90 percent of the

radium and thorium, producing a soil product with about 15 pCi/g radium-226 or thorium-230. Hydrochloric acid was as at least as effective as nitric acid for radium removal but not as effective for thorium. However, a sequential extraction, using hydrochloric acid followed by sodium hexametaphosphate, produced similar results as obtained with nitric acid. These results suggested a follow-up study to examine the sequential extraction protocol and to optimize the variables in the procedure.

Soil samples from the Superfund site were extracted with acid or acid/salt solutions to determine their effectiveness in removing the radium-226 and thorium-230. This paper presents the results of those extraction studies.

EXPERIMENTAL

Radiochemical Analyses

All soil samples and soil fractions were dried at 60C in preparation for analysis for radium-226 (8,9). Samples of soil and selected fractions and samples were also analyzed for thorium-230 (8-12) and, in some cases, for uranium isotopes (8-11).

Radium-226 Analysis

Dry soil samples or soil fractions were weighed in counting containers representing the appropriate standard counting geometry (10, 20, or 40 mL in a 60 mL vial; 100, 200, 300, or 400 mL in a 450 mL container). The samples were analyzed by gamma-ray spectroscopy using high purity germanium detectors. Radium-226 was identified and measured using the 186 KeV photopeak. Since only very small quantities of uranium-235 were found in the samples (less than 0.1 percent), interference by its 185 KeV photopeak was not a significant consideration.

Thorium-230 Analysis

Aliquots of soil samples and fractions were completely solubilized in hot-acid mixtures. Thorium was separated by ion-exchange chromatography and counted by alpha spectroscopy, using thorium-234 as a tracer (12) to determine the chemical yield of the procedure. The yield was determined by counting on a low-background beta counter.

Chemical Extraction -- Sample Preparation

Approximately 450-g samples of soils were analyzed for radium-226. After analysis, the samples were mixed with deionized water (5 mL/g) in a one-gallon Nalgene container fitted with a screw cap and shaken at 350 rpm on a Lab-Line Orbital Shaker at room temperature for 30 minutes. At the end of that time, they were hand screened through a 4-mesh sieve (4.75 mm) then a 16-mesh sieve (1.18 mm), using copious amounts of water to wash the soil through each sieve. As a result, three soil fractions were collected and dried at 60C: +4, -4/+16, and -16. the +4 and -4/+16 fractions were stored in bulk; the -16 fractions were divided into 40-mL portions, weighed, analyzed for radium-226 and used for extraction studies. The -50 and +50 sieve-size fractions were prepared in a similar manner, using a 50-mesh sieve (0.3 mm) only. Whole soil was dried at 60C and analyzed.

Extractions Conditions

Soil samples, prepared as described above, were extracted with acid, salt, or acid/salt solutions. Conditions were selected under the guidelines described in a previous report (7) and after interpretation of the extraction experiments described therein.

Acid Conditions

Samples were extracted with 0.5, 1.5, or 3.0 molar (M) hydrochloric or nitric acid once or twice for three hours at 30, 60, or 90C and a liquid-to-solid ratio of 2, 5 or 10 mL/g.

Salt and Salt/Acid Conditions

Samples were extracted with salt/acid solutions containing sodium chloride and hydrochloric acid. The extraction conditions were limited to 2.0 or 1.0M salt with 3.0, 2.0, 1.0M hydrochloric acid (for at total chloride concentration of 3.0M), 30, 60, or 90C, one- or two-stages, three hours at liquid-to-solid ratio of 2, 5, or 10 mL/g. Sodium hexametaphosphate was used at 0.036, 0.11, or 0.22M.

Single-Stage Extractions

A one- or two-liter spherical reaction flask was closed with a 4-neck (standard taper) cover. The cover was secured with a kettle clamp and fitted with a reflux condenser, centigrade thermometer, precision glass stirrer with Teflon paddle, and thermocouple assembly. The stirrer was rotated by a T-Line high-torque stirrer that could deliver up to 4.7 inch-pounds at 500 rpm. Stirring speed was monitored by a Cole-Palmer optical tachometer, model 8204-20. The thermocouple assembly consisted of a Glas-Col iron-constantan thermocouple, model JS 2048P, sealed in a glass tube to protect it from acid corrosion and packed with glass wool to aid the transfer of heat from the glass tubing to the metal thermocouple. Heat was provided by a mantle powered by a variable autotransformer set at 50 percent output. The temperature was monitored and controlled by a Glas-Col Digitrol, model PS 612, that was connected to the thermocouple and the variable autotransformer. Visual monitoring was available with the centigrade thermometer.

To begin an extraction experiment, an appropriate volume of extractant was added to the reaction vessel through the thermometer neck. Heating and stirring at 350 rpm were begun, and the temperature of the solution was allowed to stabilize at the selected value. The soil sample was transferred by a spatula with the aid of a small amount of water and acid solution. At the end of the extraction period, the mixture was cooled in an ice bath to room temperature. All mixtures were filtered with suction on a Whatman No. 1 filter paper using deionized water to thoroughly wash the sample on the filter paper. The filter cake (R1) was dried at 60C, weighed, and analyzed for radium-226 and thorium-230. The filtrate (F1) was subsequently filtered with suction through a 0.4 micron Nuclepore polycarbonate filter. The resulting filtrate (F2) was weighed and analyzed for radium-226 and thorium-230.

Multi-Stage Extractions

Multi-Stage extractions were performed using the method described above with the following alterations: soil samples were extracted, and the filter cake from the first filtration step of the procedure (R1) was used for the next stage of extraction. The filter cake produced by the second extraction was used for the subsequent extraction and so on until the number of extraction stages were complete.

Sequential Extractions

Sequential extractions were performed on soil samples using the multi-stage procedure described above, except that each extraction in the process was performed with a different reagent. For example, the soil sample was first extracted with hydrochloric acid, and the residue (R1) was then extracted with sodium hexametaphosphate solution.

Parametric and Optimization Studies

In order to investigate the effect of the principal parameters on radium and thorium extraction efficiency when using the most promising extraction procedures, random designs were constructed and a series of experiments following these designs was performed. The types of procedures examined were:

I) Extraction using hydrochloric acidonly radium measurements made.

II) Extraction using nitric acidradium and thorium measurements made.

III) Extraction using a mix of hydrochloric acid and sodium chlorideonly radium measurements made.

IV) Sequential extraction using hydrochloric acid followed by sodium hexametaphosphateradium and thorium measurements made.

All extraction procedures were run for three hours; and, for each procedure type, the following variables were randomized:

1) Experiment number (whether the particular experiment was performed first, second, third, etc.)

2) Liquid/solid ratio (Low = 2, Medium = 5, or High = 10 mL/g)

3) Hydrochloric or nitric acid molarity (Low = 0.5, Medium = 1.5, or High = 3.0M), where applicable

4) Extraction temperature (Low = 30, Medium = 60, or High = 90C)

5) Molarity of hydrogen ion for hydrochloric acid and sodium chloride mixes only (Procedure Type III only) (Low = 0.5, Medium = 1.5, or High = 3.0M; NaCl was included to make the chloride = 3.0M)

6) Molarity of sodium hexametaphosphate (Low = 0.036, Medium = 0.110, or High = 0.220M)for the sequential hydrochloric acid/hexametaphosphate extractions (Procedure Type IV only)

7) Extraction vessel utilized (three different vessels were available for use)

8) Detector used to count the sample (seven different GeLi detectors were available for counting)

RESULTS AND DISCUSSION

Chemical Extraction

In experiments reported earlier (7), 3M hydrochloric acid or nitric acid was found to be effective in reducing the radium-226 specific activity from about 135 pCi/g to 15 pCi/g and thorium-230 from about 127 pCi/g to 15 pCi/g; salt solutions of various types, including sodium chloride, were not effective alone. In this study the effectiveness of the hydrochloric acid mixture with sodium chloride was examined to determine if the concentration and therefore the actual quantity of hydrochloric acid used in the extraction procedure could be reduced. The data in Tables I and II indicate that 2M sodium chloride with 1M hydrochloric acid produced results similar to those obtained with 3M hydrochloric acid alone. Thus, a dilute hydrochloric acid solution, requiring only one-third the amount of acid previously used for radium-226 extraction, is a promising candidate for a chemical processing plant. Notice that the chloride ion concentration in sodium chloride/hydrochloric acid is 3M, the same concentration as in 3M hydrochloric acid. In addition, the use of the salt/acid mixtures, as the first step in a two-step process to effectively remove both radium-226 and thorium-230, is as effective as using 3M hydrochloric acid in the first step. Apparently, as long as the acid concentration is about one molar, a chloride ion concentration of approximately 3M from an alternate chloride-ion source is sufficient for effective radium-226 extraction.

Table I

Table II

The results of sequential extraction of the +50-mesh fraction and -16-mesh soil with hydrochloric acid followed by sodium hexametaphosphate are also summarized in Tables I and II. The data indicate that a sequential extraction, using either 3M hydrochloric acid or 2M sodium chloride/1M hydrochloric acid followed by 0.22M sodium hexametaphosphate, produces results with whole soil that are similar to those found in previous studies (7) of -16-mesh soil. The final specific activity of radium-226 is less than 15 pCi/g, and that of thorium-230 is less than of radium-226.

The -1/4"/+50-mesh fraction produced by the vigorous wash method developed earlier for volume reduction (7), provided surprising results with the radium-226 specific activity of about 5 pCi/g and thorium-230 specific activity slightly less.

These results offer a promising alternative for the treatment of soil and/or the potential for further reduction of the radionuclide content of the vigorously washed soil fraction to about the 5 pCi/g range by a combination of physical-size separation and chemical extraction of the +50-mesh fraction.

Overview of Parametric and Optimization Studies

Space limitations do not permit including the entire database resulting from this series of experiments here. However, Table III summarizes the best four results obtained for each procedure type. In this table and the remaining paragraphs of this section, the notation used is self explanatory except for the following point:

In conducting the sequential extraction experiment (Procedure Type IV), 3M hydrochloric acid was first used with an L/S ratio of 10 mL/g at 90C (all variables "High"); an appropriate experimental design was followed. This first extraction step was then followed by a hexametaphosphate extraction. In this case, each sample was returned to the same vessel used in the acid extraction step for hexametaphosphate processing. All other variables were randomized, and it is the levels of these latter quantities (i.e., those describing the hexametaphosphate extraction step) that are cited in Table III.

Table III

Statistical Comparisons of Radium Yields

A number of statistical comparisons were made for the percentages of reduction in sample radium content achieved using different types of procedures. These comparisons were made by matching on the relevant variables ("L/S Ratio," "Molarity of Agent," etc.) All of these considerations were conducted at the 5 percent level of significance; the results are summarized below:

Overall, nitric acid gave yields superior to those achieved with hydrochloric acid. This superiority persisted at high molarity but vanished at high L/S ratio.

Nitric acid performed better, in general, than hydrochloric acid/salt mix; this superiority persisted in extractions using high molarities but was not present at high L/S ratio.

Hydrochloric acid was superior to hydrochloric acid/salt mix in an overall sense, but this superiority vanished above 30C.

Funding limitations and the scope of work precluded investigation of a number of other comparisons that were of interest. This investigation was commissioned as a parametric study, and not, for example, as a study of the comparison of any two of these extraction procedures when they are

conducted at near optimum conditions. The reader may find the answer to other questions unanswered elsewhere (e.g., Ref. 7).

Mathematical Modeling of Radium Extraction Processes

The possibility of mathematically optimizing radium extraction procedures was a point of major interest to the Environmental Protection Agency (EPA) in this work. This investigation was undertaken generally following the methods used by Torma (13) in an earlier research endeavor, but more comprehensive computations were made possible in this work through use of the SAS (Release 5) statistical analysis system (14). In reviewing the results discussed below, it must be remembered that only 27 out of the 81 possible extraction configurations possible when varying three variables (L/S, molarity, and temperature) over three levels (Low, Medium, and High) were explored. Further, it was not possible, due to time constraints, to replicate this experiment. Thus, there is substantial opportunity for confounding of effects to appear in the statistics given below. Of course, it would have been better, in principle, to have used a replicated full factorial design for this experiment; but it was not possible to devote this level of effort to this taskthis would have required over 300 extraction experiments. In addition, since there are only 26 degrees of freedom provided by this design, one must regard with caution the results obtained when investigating models having large numbers of predictors. A summary of the modeling effort is given below. In these models the independent variables were L/S Ratio ("LS/R"), Molarity of Agent ("HCl" or "HNO3" or "H+," respectively), and Extr Temp ("TEMP"). The quadratic terms are denoted below in suggestive fashion. Examples: "LS/R_LS/R" (quadratic LS/R term); "LS/R_TEMP" (interaction between LS/R and TEMP), "TEMP_TEMP" (quadratic TEMP term), etc. The dependent variable was the Percent Radium Reduction ("Ra%_Red") achieved. The modeling investigation involved several steps for each procedure considered:

1) First, a comprehensive regression calculation was performedthat is, Ra%_R was modeled as a linear function of all subsets of the nine variables that result from linear and quadratic combinations of liquid/solid ratio, molarity, and temperature. (Note again that only three of these variables are truly independent.)
2) Secondly, the best reasonable model (if any) was identified for use in the optimization calculations: the model offering the best combination of high predictive power and low degrees of freedom (number of independent variables) was chosenthis obviously involved a subjective judgment.
3) Finally, this best model was used to predict the combination of liquid/solid ratio, molarity, and temperature that maximizes yield.

Hydrochloric Acid Model

The data recorded from the hydrochloric acid extractions proved to be exceedingly well-suited to such a quadratic modeling effort. Models with such a high degree of predictive power are very rarely observed in other than very simple physical processes. Even very simple linear models for these data are extremely statistically significant. Even more surprising, however, was the finding that only two of the independent variables, LS/R and TEMP are important in these modelsHCl (when present in the amounts considered in this experiment) is weakly related to the yield achieved.

Best Six-Term Model

Almost 90 percent of the variance of the yield is captured by a full quadratic model based on LS/R and TEMP (again, without the acid molarity variable). An interesting observation is the significant interaction

between LS/R and TEMP that occurs in this regression. Indeed, all of the terms in this model, with the exception of the constant, are significant. This model indicates that large percentage reductions in radium activity (largest t statistic magnitudes) are most strongly associated with the temperature and liquid/solid ratio variables. However, when examining these results, it is important to remember that only 27 data points were available, and, thus, there are only 21 degrees of freedom for errortherefore, the results obtained here must be viewed with caution.

Table IV

Best Seven-Term Model

The model summarized below, which accounts for almost 93% of the variance of dependent variable, is the best of the seven-term regressions. It is included here since it represents the first instance in which acid molarity was a variable in an optimum regression. (There are 20 degrees of freedom for error here.) Note that it is actually the interaction of HCl and TEMP that is significant here. As before, all variables except the constant term are significant.

Table V

Again one finds that large percentage reductions in radium are most strongly associated with the temperature and liquid/solid ratio variables.

Optimization of the Hydrochloric Acid Model

The model chosen for the optimization study was the six-term model above: Eq. 1

The usual equations for the stationary point of this surface are easily solved; the solution is LS/R = 7, TEMP = 89.6. It is important to note again, however, that a number of the variables in this model are correlated and, additionally, that there were no data points actually taken near this stationary point: thus this determination is expected to be of only nominal accuracy. It is the judgment of the authors that the appropriate conclusion from this optimization exercise is that high levels of LS/R and TEMP are optimum.

Hydrochloric Acid/Sodium Chloride Model

The remarkable success of the modeling effort for the hydrochloric acid data was repeated and even slightly exceeded by the hydrochloric acid/sodium chloride data. The results from the two modeling efforts were similar almost in every way. For example, one again finds that only LS/R and TEMP are important in models of these dataH+ was weakly related to the yield achieved. The results of two of the models considered are summarized below:

Best Six-Term Model

Almost 96 percent of the variance of the yield achieved is captured by a full quadratic model based on LS/R and TEMP (again, without the hydrogen ion molarity variable). Note that all of the terms in this modelincluding the constantare significant, and, in fact, to approximately the same degree.

Table VI

Best Seven-Term Model

The model summarized below is the best of the seven-term regressions, but it offers only trivial improvement in predictive power over the previous one. As in the case of the hydrochloric acid data, this regression is included here since it represents the first instance in which ion molarity was a variable in an optimum regression. On the other hand, one

finds that molarity is the only variable in this model that is not significant. The same cautions given above are applicable here.

Table VII

Optimization of the Hydrochloric Acid/Sodium Chloride Model

The six-term model above was optimized. Specifically, this model is:

Eq. 2

The stationary point of this surface is LS/R = 6.9, TEMP = 86.5. The similarity of these values to those obtained for the hydrochloric acid extractions is remarkable. (Of course, the same cautions discussed there apply here too.)

Nitric Acid Model

In contrast to the results above, the statistics for the nitric acid data are rather typical of regressions for physical problems: None of the models considered captures as much of the yield variance as did the previous models.

Best Six-Term Model

About 78 percent of the variance of the yield is accounted for by the variables of this model: TEMP, LS/R, HNO3, and the two interaction terms indicated. In contrast to the previous models, molarity (HNO3) is significant; the constant term is not.

Table VIII

Thorium-230 Extractions

Due to the lack of individual sample thorium-230 measurements, statistical analysis of the thorium data was not possible. However, note that when extraction methods that are effective for the removal of thorium-230 (nitric acid extraction or hydrochloric acid with or without sodium chloride followed by sodium hexametaphosphate) are used, the final thorium-230 specific activity is almost always less than that of radium-226 (7). Furthermore, the general effect of varying extraction conditions for thorium-230 yields closely parallels the patterns earlier observed for radium-226. Therefore, it is possible that thorium-230 extraction may be optimized by approximately the same extraction conditions that optimize radium- 226 extraction.

CONCLUSIONS

Chemical extraction studies of soil samples demonstrated that dilute nitric acid solutions are effective in removing over 90 percent of the radium and thorium activity. Dilute hydrochloric acid solutions removes over 90 percent of the radium but less thorium. However, a sequential extraction procedure, employing hydrochloric acid followed by hexametaphosphate solutions, is equally effective in removing both radium and thorium. Hydrochloric acid can be replaced by a hydrochloric acid/sodium chloride solution, reducing the hydrochloric acid requirements for the process.

A full quadratic model of the bench-scale extraction studies data revealed several important interactions between temperature, concentration of extractant, and liquid-to-solid ratio. For hydrochloric acid and hydrochloric acid/sodium chloride extractions, the model suggests a high degree of prediction that is based on the liquid/solid ratio and the temperature of the reaction. Within the range of concentrations selected for this study, the concentration term is much less important. In contrast, the model for nitric acid extraction includes a significant contribution by the acid concentration, in addition to the liquid/solid ratio and temperature terms.

Physical separation of the whole-soil samples by particle sizing isolated a significant quantity of a soil fraction larger then 50-mesh size (0.30-mm) (+50) with radium-226 and thorium-230 concentrations less than 15 pCi/g. This fraction was extracted under optimum conditions to assess chemical treatment as a remediation technology to reduce the concentration of the contaminants to less than 5 pCi/g above background. Chemical extraction of the +50 soil fraction containing less than 15 pCi/g radium-226 and thorium-230 using the sequential extraction method described above reduced the concentrations to approximately 5 pCi/g.

A combination of physical separation of whole soil followed by chemical extraction of a select particle-size fraction might offer a viable method for site remediation of radionuclide-contaminated sites by volume reduction. As an example, more than one-third of the soil from the Superfund Site described in this paper would be separated by simple particle-size separation, producing a fraction that would contain approximately 15 pCi/g radium-226 (3). Subsequent chemical extraction of this fraction, as described herein, would reduced the concentration to about 5 pCi/g.

Particle-size separation of soil samples from other sites yield soil fractions containing similar concentrations of radium-226 and representing from 50 to 65 percent of the total soil (5). Recovered fractions representing greater percentages of total soil material potentially make them more attractive to further contaminant reduction by chemical extraction processes. Depending on the clean-up criteria, alternate fractions with intermediate concentrations of contamination might also be candidates for extraction, if their contaminant concentration could be reduced to meet the criteria. The fraction might, therefore, be recovered and added to other fractions separated by the physical processes alone. For example, a contaminated soil might render up to 50 percent of its volume as a "clean" product by particle-size separation. A fraction with somewhat higher contaminant concentration but representing another 25 percent of the soil could be remediated by extraction producing a total volume reduction of approximately 75 percent. A physical-separation processes complemented by chemical extraction of select size fractions might result in a remediation technology that would offer, therefore, the optimum approach to volume reduction.

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CHARACTERIZATION OF DU CONTAMINATION AT THE TRANSONIC RANGE AT ABERDEEN PROVING GROUND

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ABSTRACT

A characterization study of radioactive contamination, hazardous waste and unexploded ordnance (UXO) at the Depleted Uranium (DU) testing area of the Transonic Range at the Aberdeen Proving Ground (APG) was conducted in the fall 1995 through the winter of 1996. DU penetrators were tested at a site located at the Transonic Range from 1973 through 1979. Although outdoor testing of DU penetrators ceased in 1979 at the 12 acre site, the need to release the site for nonradioactive use has become the goal of

the U.S. Army Research Laboratory (ARL), and the APG Health Physics Office.

APG is a Test and Evaluation Command installation within the United States Army Material Command. The mission of the installation is to develop and test military materials and to train officers and enlisted personnel in the use and maintenance of munitions. The installation was established in 1917 as two separate military reservations, APG and Edgewood Arsenal. Both were consolidated into APG in 1971. APG is located in northeast Maryland, has approximately 73,000 acres of land and water and has approximately 55 tenant organizations, many of whom are engaged in munitions development and testing.

This paper discusses development of the characterization strategy, the health and safety plan, a quality assurance project plan and the work plan. Also presented are the results of the interviews with personnel responsible for the outdoor DU testing activities, test supervisors and test technicians. Results of the surface survey for UXO and preliminary radiological data are presented.

INTRODUCTION

The purpose of the characterization was to establish the radiological status of the Depleted Uranium Study Area (DUSA) in accordance with licensee requirements and the recommendations of the Nuclear Regulatory Commission (NRC) since DU testing at DUSA ceased in 1979. The need to perform a radiological characterization is found in NRC's Branch Technical Position (BTP) "When to Remediate Inadvertent Contamination of the Terrestrial Environment." (1). According to the BTP, licensee sites which are no longer used to conduct licensed activities should be remediated to unrestricted use levels within two years of last licensed activity to preclude migration of the radioactivity. A determination was made in 1995 to cease licensed activities at DUSA.

The DUSA is located at a site within the boundaries of the Transonic Range which is in the Aberdeen Area of Aberdeen Proving Ground (APG). APG lies along the western shore of the Chesapeake Bay in Harford and Baltimore Counties, Maryland, approximately 15 miles northeast of Baltimore (Fig. 1). APG covers a total of 72,516 acres (land and water) and consists of two distinct areas: the northern portion of APG, referred to as the Aberdeen Area (AA); and the southern portion of APG, referred to as the Edgewood Area (EA) (2). The AA became a formal military post, designated as APG, in 1917. The EA (formerly Edgewood Arsenal) was appropriated by Presidential Proclamation in 1918.

Fig. 1

DUSA is located on the southern portion of the Transonic Range and was known as "Lower X-ray" operated by the U.S. Army Research Laboratory (ARL). The DUSA was used for hard-impact testing of depleted uranium (DU) penetrators. DU penetrators were gun launched against armor targets to test their effectiveness. DU is depleted of the natural percentage of U-235, and contains U-238, U-235 and U-234 all of which are radioactive. The DU penetrator testing is known to have contaminated target material and the surrounding structures and soil. Since 1979 contaminated materials and soils have been removed from the area, however a complete characterization and remediation for free release has not been performed.

METHODS

The initial step was to review the history of DUSA prior to establishing the scope of the characterization in order to determine the level of effort and resources needed to devote to the project. Subsequent to the

historical survey a work plan, health and safety plan and quality assurance project plan were developed.

Historical data was largely obtained from interviews with personnel formerly involved in testing at DUSA as little documentation about the testing program exists. Interviewees included a health physicist that was assigned to the site during the later years, two health physicists that have conducted environmental monitoring at the site, the principal test supervisor, two test foremen, and two test technicians. Data was also obtained from environmental monitoring reports.

RESULTS OF THE HISTORICAL RESEARCH

The DUSA is located on the southeast end of the Transonic Range and was used for DU testing from 1973 through 1979. Testing consisted of gun launching DU penetrators from two locations (next to Target Storage) at targets mounted adjacent to the X-ray units to the south (see Fig. 2). Stripper /deflector plates located in between the launch or shooting location were designed to strip or deflect the sabot away from the penetrator while in flight to the target. Penetrators were either completely stopped in the target or penetrated (either partially or completely) the target and impacted into a backstop located a short distance behind the targets. As a result of the testing, most of the DU melted into the target and backstops, however some DU fell onto the soils around the targets or was scattered into the surrounding area. Test technicians wore protective clothing and dosimetry while in the test area and monitored activities for radiation. Except for shot target and other designated contaminated materials, most radioactive wastes were handled through established radwaste disposal channels. Shot target and other designated materials were stored onsite in a wooded area located to the east of impact area. The shot target and other designated materials have been remediated, recycled and or disposed in an approved disposal site.

Fig. 2

After testing ceased the ARL health physicist conducted an initial site cleanup to reduce the radioactive waste inventory and to allow other non-radiological testing to be conducted at the site. This effort included removing shot target and other materials from the storage area and surveying and removing contaminated soil, sand or other materials. The health physicist estimated that approximately 12 acres should be considered suspect areas (see Fig. 3). Figure 4 shows the suspect area and the site divided into 200 by 200 foot grid.

Fig. 3

Fig. 4

ARL conducted soil sampling from 1973 to 1978, and again in 1991 as part of their environmental monitoring program. Analytical results for soil samples collected from several areas around the radioactive waste storage area and behind the backstops showed levels in excess of the release criteria for unrestricted use of 35 pCi/g (3). Most of the soil contamination was found in storage area which has been posted as a Radioactive Materials Area. Of the 165 soil samples collected from this area, 37 were greater than the 35 pCi/g limit. Concentrations ranged from background to 520 pCi/g.

The areas behind the backstops showed soil concentrations below the limit for free release with few exceptions. Soil samples collected in and around remaining site structures were below the release limit.

Further information was obtained from site visits.

Site Description

Figure 3 shows site details collected from Geographical Information System (GIS), interviews with range personnel and site visits. The 12 acre site contains the items in Table I.

Table I

Each of the structures known as X-ray 1 and 2 contain X-ray equipment mounted in U.S. Army equipment trailers. The trailers are surrounded by plate steel, ranging from two to four inches thick. The floors are gravel.

Test program material was stored in a Security Container. The original security container was moved to a location west of the Transonic Range Building (740 C) and replaced with a newer container. The replacement Security Container has been classed as an unaffected structure.

The Backstops, Stripper Plates and Target Supports are not thought to be contaminated; however, site work will involve scoping surveys to characterize them. The Catch basin for the Transonic Range will be reviewed and likely classed as an unaffected structure. Interviews with range personnel indicate that DU penetrators were not directed at this backstop.

The east side of the site was stabilized over seven years ago when a layer of landscaping plastic was laid along the east road of the site from 740 A7 to the tree line to the east, to the Bomb Safe to the north, and to the path for X-ray 1 to the west. A 4 to 8 inch layer of gravel was laid over the plastic.

Also about 7 years ago, the swampy area to the north of the firing positions was covered with 2 to 4 feet of soil to allow guns to be moved further away from the targets.

PROJECT WORK PLAN

As a result of the interviews and the site visits, the work required for the characterization was broken into six tasks. The tasks were as follows:

- Conduct surface sweeps for UXO using visual techniques and magnetometry
- Conduct radiation surveys using walkover techniques of the 12-acre site
- Conduct a geophysical survey for buried ferrous and non ferrous objects using EM 61
- Conduct radiation surveys of affected structures
- Conduct targeted soil sampling to compare with the survey results
- Develop recommendations for site cleanup if necessary

These six tasks not only formed the basis of the workplan, they were also used to scope the hazards analysis in the site health and safety plan. Individual procedures were developed for each of the tasks.

QUALITY ASSURANCE PROJECT PLAN

A quality assurance project plan (QAPP) was developed to ensure that the data from the surveys was adequate to meet the goals of the data objectives. The plan was developed according to the guidance from EPA Region III and QAMS 005 (4). Specific objectives are to be met by data collected during the characterization are summarized below:

- Determine where radiological contamination of soils and structures exists at DUSA above NRC release criteria.
- Acquire data to be used by the APG Health Physics Office to determine where remediation is required.
- Integrate data into a comprehensive GIS database which may be used to facilitate the feasibility study and license termination.
- Determine the volume of soil requiring remediation.

Determine the extent of structure contamination and, if contaminated, determine possible remediation strategies.

The plan cross referenced to the task procedures.

SITE HAZARD IDENTIFICATION AND ASSESSMENT

The Health and Safety Plan (HASP) was developed to address each type of hazard anticipated for each task of the project. Some of the hazards are seasonal related. Examples of seasonal hazards include heat stress, insect bites and stings, and animal encounters during the late spring, summer and fall and cold stress during the winter. Of particular concern was the potential for deer tick bites and possible development of Lyme disease. The other hazards include exposure to ionizing radiation, detonation of UXO, straying into an impact area during testing and tripping, cuts, and falling hazards.

The principal hazards at the DUSA site during the winter of 1996 included accidental detonation of UXO, cold stress, exposure to radioactive material, and physical hazards associated with field activities (such as tripping, falling, and cuts). The major radiation contaminants known to be on site include uranium isotopes and their progeny. The radionuclides of concern include U-234, U-235 and U-238.

Radioactive materials represent an external and an internal exposure potential while conducting site work. Inhalation, ingestion, and dermal contact hazards may exist. Due to past activities, the DUSA has the potential of having radiological contamination present. The HASP recommended general radiation surveys of the site during initial sweeps for UXO and monitoring personnel for contamination whenever personnel left the site or handled objects located within DUSA. Equipment that may have been used at other radiological sites was given a baseline frisk to compare with subsequent frisk results in order to determine whether contamination of the pieces has occurred.

A variety of ordnance types were expected in the DUSA. The DUSA was a known impact area for DU and kinetic energy penetrators as well as projectiles containing high explosives. In addition, DUSA is bounded by active ranges on three sides.

The exposure hazard associated with UXO items is related to unexpected disturbance causing leakage of contents or detonation. Hazards include possible explosion and leakage of explosive chemicals near the point of UXO discovery.

Hazards from UXO were minimized by practicing avoidance of uncleared areas until the completion of magnetometry surveys, and then conducting a magnetometry survey wherever intrusive field work was to be conducted (i.e., placing grid markers for survey reference, and soil sampling). Other UXO and hazards of straying into impact areas was minimized by establishing the site grid, clearing the site of UXO and cautioning workers to stay within the DUSA and to maintain radio contact with the U.S. Army Test Center's Range Control.

DUSA Physical Hazards

Physical hazards associated with the DUSA characterization activities included light equipment operation (e.g. generators, etc.), traversing undeveloped areas, adverse weather conditions, possible wildlife encounters and water hazards in marshes and creeks. Mitigation measures included safe field practices; safe lifting techniques; and adhering to approved plans, SOPs, and manufacturer's specifications for equipment.

RESULTS and DISCUSSION

Surface Survey for UXO and Site Preparation

The 12-acre site and adjoining ranges have been used to test projectiles containing DU, high explosives and other material therefore, the potential for UXO encounters existed. Prior to working the site, a UXO survey for surface ordnance was conducted using hand held magnetometry instruments. The U.S. Army Technical Escort detachment was contacted to render UXO items safe. Because the radiological history of the site was uncertain, a health physicist accompanied the UXO team while the UXO survey was conducted to ensure that workers did not encounter high radiation or contamination levels.

The duration required to conduct the survey for surface UXO doubled after finding numerous UXO in the first two days. The UXO survey proceeded as follows: a geophysical survey of the site was conducted to establish a 200 by 200 foot grid of the 12 acre site, once the grid was completed, one UXO expert marked lanes and directed the team that walked parallel lanes a distance equal to the swing of the magnetometer. Contacts were flagged and later investigated for ordnance.

The geophysical survey established the boundary for the surface UXO survey, radiation survey and the electromagnetic survey. A total of 24 grid points were placed at 200-foot by 200-foot intervals throughout the 12-acre site. Originally, the reference grid was to be established after the site was cleared of UXO, however because of the number of UXO discovered by the second day of site work, it was necessary to establish the grid earlier in order to conduct the surface sweep in an efficient manner. For the survey, stakes were driven at each node after the UXO contractor verified that the stake would not impact underlying UXO. Physical stake locations were needed as reference points during the radiation and electromagnetic surveys and to accurately determine the extent of remediation.

During the surface survey for UXO, the team determined if the UXO contained explosive material and whether the item was fused and armed. Scrap and nonhazardous UXO were removed to a central location for official verification by the government. Hazardous UXO were cataloged and recorded on a site map. UXO that were too hazardous to move were designated for field disposal. Those UXO requiring field disposal were surveyed for radiation as well as the general area around the item to ensure that detonation of of the UXO would not disperse DU into the surrounding area. A total of 121 UXO items were found during the surface survey. Table II has a listing of the UXO items by type. As the listing shows, approximately two thirds of the items were 105 and 155 mm projectiles and almost half of the items contained high explosives.

Table II

Table III lists the UXO items found by survey grid. A total of 33 UXO was found behind the Transonic Range Catch basin (grid ref. A2), the next highest was 13 UXO found in grid A3.

Table III

Once the survey determined that surface UXO was no longer a concern, site clearance began. Clearing consisted of moving non essential surface objects, scrap metal, materials and equipment, wood articles and debris so that the site could be effectively characterized. Surface objects pose a hindrance to the radiation survey and interfere with the EM-61 survey. This required moving fallen logs, partially buried debris and other material out of the survey area. It was also necessary to cut foliage as close as possible to the ground via brush hog in order to minimize the radiation detector soil distance.

Radiation and Electromagnetic Surveys

At the time of writing, the radiological characterization of buildings, soils and soil samples for DU contamination had not begun. Preliminary surveys of the site showed no removable contamination above background. However, contamination was found in several areas of the site including the asphalt pad leading to the radioactive materials storage area, as well as several pieces of debris discovered during site clearing. Radiation surveys showed levels between 0.15 to 0.20 Sv/h (15 to 20 mR/h) south of the backstops to 0.15 Sv/h (15 mR/h) from the backstops to the X-ray buildings and 0.05 to 0.1 Sv/h (5 to 10 mR/h) from the X-ray buildings north.

The characterization surveys of the site are to be conducted at approximate intervals of 18 inches apart with a field tested gas proportional counter. Data regarding count rate and location on site will be communicated to an on-site computer to plot count rate from the detector in the x and y direction. Once this survey is complete, biased soil sampling will be used to verify isoconcentrations from the site survey.

An electromagnetic radiation survey for submerged items and other materials that had been inadvertently left on site below the surface will be conducted after the radiation survey. The method of choice to detect submerged ferrous and non ferrous objects is EM-61. Originally the EM-61 survey was to be conducted in a manner similar to the radiation survey, that is the transmitter and receiver mounted on a cart that is pushed/pulled along survey lanes. However, there were risks identified during the surface UXO sweep from a device that transmits an electromagnetic signal in an area where proximity fused munitions may be present. The concern for proximity fused munitions is that the electromagnetic device may induce an electrical current sufficient to detonate the munition. Two proximity fused UXO were found at DUSA and for this reason the EM-61 survey will be conducted only in areas where radioactive contamination will be confirmed during radiation surveys and will be conducted using robotic tracked vehicle. The EM-61 survey will be conducted remotely at a distance of 345 meters to protect the operator from the effects of inadvertently detonating a subsurface UXO that contained a proximity fuse.

CONCLUSIONS

Although the full extent of contamination and the number of subsurface items is unknown at this time, there are several conclusions that can be made.

Based on general radiation levels, the areas designated as A2, B1, B2, C1 and C2 may require remediation. Levels varied from 0.15 to 0.20 Sv/h (15 to 20 mR/h) with peak values of 0.5 Sv/h (50 mR/h). Also based on general radiation levels the existing structures do not appear to be contaminated. The levels of DU contamination are less than those requiring personnel monitoring, but appear to be greater than the release criteria in Ref. 3 for some areas and therefore will require remediation. At this time, the most significant hazard is encountering UXO. The density may be 10 UXO per acre.

The need to perform a systematic, well-defined radiological characterization of the DUSA is found in NRC's BTP (1). According to the BTP, licensee sites which are no longer used to conduct licensed activities should be remediated to unrestricted use levels within two years of last licensed activity to preclude migration of the

radioactivity. Thus, the site should be characterized, remediated and released for unrestricted use as within the time frame negotiated with the NRC.

The guidelines in Ref. 5 should be followed. The site history proved an invaluable in particular when developing the data quality objectives and the health and safety plans. The risk from proximity fused munitions was unknown at the beginning of the project but persistence and conservative posture regarding this risk reduced the likelihood that individuals would be harmed.

Weather conditions and seasonal variations should be considered in planning for radiological characterizations of wooded/swampy areas. Timing enabled characterization during the winter months which enabled visualization of site hazards (i.e., UXO) much easier as well as site clearing. The winter characterization allowed the team to work unhindered by dear ticks, insects and snakes. It also allowed work to progress with minimal impact to the environment.

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44-8

AIR CLASSIFIER TECHNOLOGY (ACT) DEMONSTRATION AT A DEPLETED URANIUM SAND BUTT

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ABSTRACT

The Department of Energy recently completed a "heavy metals in soils" treatability project which resulted in the evaluation of several different physical separation technologies. One of these was the air classification technology developed by industry and refined by the U. S. Naval Academy. The purpose of this paper is to share the results of a field demonstration using this new remediation technology for mixed waste products (i.e., soil containing both radioactive and RCRA-regulated materials) at a Department of Defense facility.

BACKGROUND

Naval Surface Warfare Center, Dahlgren Division (NSWC-DD), is located east of Dahlgren, Virginia, on the western shore of the Potomac River. Since 1918, NSWC's primary mission has been Naval ordnance testing and

development, which continues today. Currently, NSWC-DD is preparing to dispose of mixed waste products from two sites at its facility. One site is an outdoor sand butt which, as described by Halliburton (1), consists of approximately 13,000 cubic feet of sand enclosed in a large rectangular steel box with a single vertical open face (Fig. 1.) The dimensions of this enclosure are 24' x 50' x 15' (w x l x h). The back half of the enclosure is completely filled with sand which slopes forward from mid-length towards the open-face entrance. Prior to July 1991, the butt was used for both depleted uranium (DU) and conventional munitions testing. It is no longer in use because of concern over the level of contamination in the fill.

The sand butt was designed so that the fired projectiles would expend their energy in the sand. In addition to DU projectiles, the sand fill contains projectile fragments and metal dust resulting from abrasion of the projectiles on impact. The specific activity of the sand fill is 2.9 nanocuries/gram (estimated) with a radiation dose rate of (approximately) 2.5 mrem/hr measured at one foot above the sand's surface. In addition, the fill is known to contain at least eight RCRA-regulated metals including lead, mercury, and silver. The concentration of lead exceeds the "Maximum Concentration of Contaminants for Toxicity Characteristic" as per 40CFR261.24. As a result, the material in the butt is classified as a mixed waste.

Fig. 1

PROJECT SCOPE

The Department of Energy (DOE) recently completed a "heavy metals in soils" treatability project which resulted in the evaluation of several different physical separation technologies (2). One of these was the air classification technology developed by industry and refined by the U. S. Naval Academy (USNA). Subsequently, a field demonstration was arranged to assess the usefulness of this innovative technology for remediating mixed waste products (i.e., soil containing both radioactive and RCRA-regulated materials). Specifically, a limited quantity of contaminated soil from the outdoor DU sand butt at NSWC-DD was treated. EPA guidance for conducting treatability studies was followed (3).

The principle objective of this field demonstration was to evaluate the application of air classifier technology (ACT) in the recovery of the depleted uranium (DU) fragments. Also to be evaluated was the effectiveness of this technology to concurrently remove certain RCRA-regulated heavy metals; in this case, to reduce the high concentration of lead in the sand fill. As a secondary objective, an assessment of variations in certain system control parameters was undertaken to improve the overall effectiveness of the technology for future mixed-waste remediation efforts. Health, safety and industrial hygiene issues during equipment set-up, operation and disassembly were also assessed to enhance future operations.

AIR CLASSIFIER TECHNOLOGY DESCRIPTION

Air classifier technology is a physical separation technique that uses a centrifugal force effect to achieve separation of particulate matter by size and/or density. The particulate matter is fed into a separation chamber, wherein particles are imparted with a radial velocity through the action of a centrifugal fan. Ideally, the larger and/or denser particles move more rapidly to the separation chamber wall where they fall under gravity and are collected as coarse particle discharge. Smaller, lighter particles are carried upward by a circulating column of

air and collected in an outer chamber as fine particle discharge. Overall success of this demonstration was to be measured by the fraction of contaminants that could be concentrated in either of the two discharge streams.

The test equipment used in the demonstration was a Gayco-Reliance air separator system. The system includes a feed unit and the air classifier unit as shown in Fig. 2. The automatic feed unit is an Acrison model 105-DD volumetric dry materials feeder. It features a unique dissimilar-speed double concentric auger mechanism for accurate metering performance. The unit has a hopper with a capacity of 0.5 ft3 and its feed rate is variable up to 2.4 ft3/hr.

Fig. 2

The classifier unit is mounted alongside and linked to the feed unit by a feed tube. The 18"-diameter Gayco-Reliance centrifugal air classifier includes both a main fan and a centrifugal fan mounted on a common shaft (Fig. 3). The main fan develops a circulating air flow between the separation and outer chambers, while the centrifugal fan imparts radial velocity to particles of matter as they enter the separation chamber. Approval was obtained from its manufacturer, the Universal Road Machinery Company, to replace its constant speed motor (1750 RPM) with a variable speed motor which can operate at speeds as high as 2500 RPM. Motor RPM and, thus, fan speed can be controlled by adjusting a speed rheostat on the motor control box. Air circulation within the separation chamber can be controlled by adjusting the openings of the shutter assembly.

Fig. 3

FIELD PROCEDURES

The project was accomplished through the combined efforts of USNA, NSWC-DD, and Naval Facilities Command, Chesapeake Division (CHESNAVFAC) personnel. USNA personnel had the responsibility for preparing the necessary field documents, procuring field equipment and supplies, performing process operations (i.e., sieving, drying, air classifying, etc.), equipment decontamination, and final report write-up. NSWC-DD personnel performed initial site set-up activities, soil excavation and UXO monitoring, radioactivity airborne monitoring, sample measurements, final disposal of all waste generated by the process, in addition to supporting all field activities. The CHESNAVFAC personnel provided assistance in preparation of the field documents and regulatory guidance. Initially, 18 ft3 (approximately 2500 lbs) was excavated from the face of the sand fill and stored in plastic-lined containers for later processing. Three (approximate) 8-kg samples and three (approximate) 0.2-kg samples were extracted from this initial site soil for field and laboratory analysis of radioactivity and lead (Pb) concentration levels. These measurements would serve as baseline data for assessing air classifier effectiveness.

Following site preparation and equipment setup, it was necessary to remove soil particles (and portions of projectiles) greater than 2.00 mm to avoid jamming of the feeder. Initially, very large projectile fragments (i.e. several inches long) were separated from the excavated site soil by hand. Using a standard geotechnical sieve stack and shaker with a #10-size mesh screen, the initial site soil was then segregated into a coarse fraction (> 2.00 mm) and a fine fraction (2.00 mm). As before, three 8-kg samples and three 0.2-kg samples were collected from each fraction for subsequent radioactivity and lead-level analysis.

A sufficient quantity of the fine-fraction material was then processed in the air classifier system to obtain necessary coarse- and fine-discharge effluent samples. Prior demonstrations of the air classifier indicated that system performance may vary with the settings of certain system parameters -- most notably, feed rate and fan speed. Therefore, in this demonstration, feed rate and/or fan speed were varied to obtain different set points. Once again, three 8-kg samples and three 0.2-kg samples were collected from both discharge streams of each set for subsequent radioactivity and lead-level measurements.

The lab results of different fan/feed rate combinations were compared to determine their net effect on system performance and to identify control values for optimal air classifier effectiveness.

Following collection, each 8-kg sample was weighed and counted on-site for gamma activity using a sodium iodide (NaI) system to selectively measure DU gammas. These samples were then sent off-site and counted for DU gammas in a Marinelli beaker with NIST traceability. The on-site counting gave a relative measure of each sample's activity which was used to adjust parameters during testing. The off-site counting provided an absolute measurement of sample activity to compare the effectiveness of the ACT with possible regulatory standards. A mass balance was also performed on both the feed and effluent streams from the ACT. Finally, the 0.2-kg samples were used to assess lead and other heavy metal concentrations.

Following completion of air classifier operations, all equipment was decontaminated as it was disassembled. All waste products were collected in drums for subsequent disposal and the site was returned, as best possible, to pre-demonstration conditions.

Safety Precautions

Numerous safety precautions were taken including training in air classifier operations and hazardous waste handling and disposal procedures. A site-specific Standard Operating Procedure (SOP) and Site Health and Safety Plan (SHASP) were initially developed to preclude the possibility of harm to personnel involved in the field demonstration and/or in areas adjacent to the site, and to prevent any further contamination of the site and its surroundings while the demonstration was ongoing. The SHASP was forwarded for comment to the USEPA and the State of Virginia Department of Waste Management. All personnel involved in the field demonstration were required to satisfy the training requirements specified in the SHASP, including completion of an appropriate HAZWOPER safety training course and an on-site Hazard Control Briefing. In addition, all field workers underwent a uranium bioassay as well as a physical for respirator fitness.

As shown in Fig. 4, a three-zone approach was used to control migration of contamination and to minimize the potential for personnel exposure. All sieving and process operations were performed in a containment tent, the interior of which was the Exclusion Zone where the potential airborne DU hazard was greatest. The containment tent had dimensions of 142" x 98" x 96" (w x l x h). It was fabricated from PVC with the top half clear for personnel viewing of process operations. A Reduced Contamination Zone (RCZ) encompassed a protective shelter, an adjacent container/sample storage area, the equipment and personnel decontamination areas, the DU sand butt, and all immediately surrounding areas. These areas were known or anticipated to have limited DU contamination. The outermost Support Zone provided an additional buffer from the surroundings.

Fig. 4

Radiation levels within the containment tent and the adjacent areas of the RCZ were continually monitored for excessive levels of airborne radioactive material and dust. Airborne radioactive material levels were monitored by collecting particulate samples on filter paper using a Staplex high-volume air sampler located inside the tent and three HD-732 low-volume air samplers located outside the tent in the RCZ. The filters were later analyzed for radioactivity using a Tennelec alpha-beta counting system, i.e., a gas flow proportional counter. Airborne particulate monitoring was also accomplished by using a MIE model RAM-1 real-time aerosol monitor, located inside the containment tent.

RESULTS

In the field tests, the air classifier was operated at three different set points with dried soil from the fine fraction of the sieving process. These three points are denoted as follows: (1) LH - for low feed rate (15% of maximum) and high fan-motor speed (2250 RPM); (2) LM - for low feed rate and medium fan-motor speed (1750 RPM); and, (3) VLH - for very low feed rate (10% of maximum) and high fan-motor speed (2250 RPM). Several other set points were planned but not taken, because the field data indicated that these points would not improve the classifier's performance. The results obtained from these tests have been divided into the following four categories: mass and volume balance, field radiation analysis, absolute DU activity analysis, and the lead analysis.

Mass and Volume Balance

The sieving process resulted in approximately 75% (by volume) of the gun butt soil being collected as fine fraction and 25% as coarse fraction. No exact mass balance was made at this point of the study. It was later estimated that approximately 40% of the mass was collected as coarse fraction since the density of the coarse fraction was found to be greater than that of the fine fraction.

Table I shows the percent effluent collected in each of the discharge streams for each of the three set points. The mass balance is expressed as percent of the total fine fraction processed (given in parentheses) by the ACT. As can be seen, reducing the feed rate significantly increased the percentage of mass collected in the fine effluent. Also, because the coarse effluent contained more DU than the fine effluent, the LH and LM set points are considered to be impractical operational set points because of the small percentage of clean, fine effluent they produce.

Table I

Field Radiation Analysis

The average normalized radiation levels observed for the samples obtained by sieving are shown in Table II. The activity of the gun butt soil has been arbitrarily assigned a value of unity. As can be derived from this table, the coarse fraction has an average radioactivity level approximately six times that of the fine fraction.

Table II

The average normalized activity levels for samples collected from the ACT are shown in Table III for the VLH set point. For these results, the activity of the fine effluent has been arbitrarily assigned a value of unity. As shown in this table, the coarse effluent has an activity approximately 39% greater than that of the fine effluent. The fractional standard deviation presented is based on a statistical analysis of all collected samples at this set point.

Table III

Absolute DU Activity Analysis

The absolute DU activity levels of the collected samples, expressed in units of pCi/kg, are shown in Table IV. Also shown are the ratios of each sample's absolute DU activity to the gun butt soil's absolute DU activity. The results represent the average of all collected samples within a group.

Table IV

This laboratory data shows the same qualitative results as obtained by the field radiation analysis; namely, that: 1) the fine sieved fraction is lower in activity than the gun butt soil and significantly lower in activity than the sieved coarse fraction; and 2) the fine ACT effluent is lower in activity than the coarse ACT effluent. However, the activity levels of all samples were well above the 30-50 pCi/kg limit suggested by the USEPA for federal DU sites (4). Finally, it should be noted that Table IV data show the fine and coarse ACT effluents with a slightly higher activity than that of their feed (i.e., the fine sieved fraction). The authors believe that this result is due to natural variations in activity of the feed. The samples of fine sieve fraction collected for analysis differed from the material actually processed to yield the ACT effluents.

Lead Analysis

Results from the lead analysis are shown in Table V and represent the average of all samples collected within a group. The data expresses the lead concentration in units of mg/ l as well as a normalized value based on the level in the gun butt soil. As indicated, sieving the soil resulted in a significantly lower lead concentration in the fine sieved fraction compared to the site soil. Additionally, the ACT process resulted in an even further reduction in lead concentration (in the fine effluent). The fine effluent samples had a lead concentration of 1.1 mg/l or approximately 3% of that found in the collected site soil samples. The remediation improvement in the lead concentrations, as compared to the DU, is believed to be due to either of the following reasons: 1) there was a more favorable size distribution of lead and soil particles than DU and soil particles; and/or, 2) there was less chemical bonding between lead and soil particles than between DU and soil particles.

Table V

CONCLUSIONS

A number of conclusions were drawn from these tests and have been divided into the following two categories: technology evaluation and health and safety assessments.

Technology Evaluation

1) Sieving the gun butt soil prior to use of the ACT appears to produce a significant concentration of DU activity (i.e. 75% of the activity into 25% of the volume) into the coarse fraction and produces a substantially cleaner fine fraction compared to the unprocessed soil (i.e., by a factor of 3). Thus, sieving alone could potentially result in a substantial volume reduction of the soil that would need to be remediated.

2) The ACT produces a coarse fraction with greater activity than the fine fraction and thus the ACT can be used to further reduce the activity of a portion the soil. However, the ACT technology did not reduce DU activity to a level which would meet currently proposed remediation standards for federal agencies.

3) The mass split in ACT between the coarse and fine effluents is strongly affected by the soil feed rate and fan speed of the air

classifier. Slowing the feed rate and increasing the fan speed causes more soil to move into the fine effluent stream and appears to improve the separation effect of the ACT.
4) The ACT is easy to operate and maintain for reliable use in field conditions. Once the soil is dried, sieved, and loaded into a feeder, little further effort is needed to process the soil. The technology requires a level surface and a standard 110V power source.
5) The ACT appears to provide a much greater separation effect for lead than DU. Perhaps even more significant, sieving was found to reduce the lead concentration in the fine fraction by better than 6:1 compared to the unprocessed site soil.

Health and Safety Assessments

1) There was little airborne dust generated by the air classifier operation. The most dominant source of airborne dust was observed during the soil transfer operations between the containers, the sieves, the oven, and the feeder.
2) The containment tent was very effective in preventing dust from leaving the exclusion zone. No significant readings were observed in any of the three air samplers employed in the RCZ (i.e., outside the containment tent).
3) No chemical reagents were required at any time in the process and water was needed only during the decontamination and clean-up operations. No equipment or surfaces were found to be contaminated with high levels of radioactivity following the completion of the demonstration.
4) Wetting the soil prior to its excavation from the gun butt reduced the dust generated during that operation, but this necessitated drying the soil prior to sieving.

RECOMMENDATIONS

1) Since sieving alone appears to have strong potential for volume reduction, the effect of different sieve cut sizes should be investigated to determine if larger concentrations of activity and, possibly, improved volume reduction can be achieved.
2) The air classifier should be equipped with a larger fan motor in order to operate the classifier at higher fan RPMs than possible with the current USNA system. The test data indicates that higher fan RPMs may further improve the ACT's performance over what was obtained during these tests.
3) Data should be collected at lower feed rates and higher fan speed than achieved in this study. These adjustments should increase the percentage of soil in the presumably cleaner fine effluent and, thus, increase the amount of soil cleansed by this process.
4) ACT effectiveness of separating uranium from soil may be improved with triboelectrification (5). The soil would first be triboelectrified which would preferentially impart an electrical charge to the contaminate. This soil would then be fed into the ACT where separation would be enhanced under the influence of an applied electric field.

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ACKNOWLEDGMENTS
The authors are deeply indebted to the assistance they received from NSWC personnel. Specifically, the authors would like to acknowledge the contribution made by Ann Swope, Billy Weedon, Jim Stafford, Dr. Gordon Riel, Richard Spurlock, Sheran Phin, and Glenn Monteith. In addition, USNA personnel who participated in the project included Mike Gibbons, ENC Gene Markerink, and BNC Gene Washington. Finally, Ryan Mayer from CHESNAVFAC provided invaluable assistance in preparation of the field test documents.

Session 45 -- MICROBIAL ASPECTS OF NUCLEAR WASTE DISPOSAL
Co-chairs: Marsha I. Sheppard, AECL;
Francois Caron, AECL
45-1
QUANTITATIVE ASSESSMENT OF IN SITU MICROBIAL COMMUNITIES AFFECTING NUCLEAR WASTE DISPOSAL
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ABSTRACT
Microbes in the environments surrounding nuclear waste depositories pose several questions regarding the protection of the surrounding communities. Microbes can facilitate microbially influenced corrosion (MIC), mobilize and facilitate the transport of nuclides as well as produce gaseous emissions which can compromise containment. We have developed an analysis of the extant microbiota that is independent of quantitative recovery and subsequent growth, based on signature biomarkers analysis (SBA). Polar lipids exist in all organisms that have intact cell membranes and intact cell membranes are a requirement for life. Extraction and measurement of polar lipids indicates the biomass of

the microbes that are viable but may or may not be culturable. Phospholipid ester-linked fatty acids (PLFA) give a measure of the microbial community containing intact membranes. Phospholipids are often transformed into diglycerides by endogenous phospholipases in injured cells which retain for a time the characteristic signature profiles of the fatty acids and thus provide a measure of the viable cells with intact membranes and recently non-viable (lysed) cells. The lipid patterns of PLFA and other lipid classes also reflect exposures of the cells to nutritional imbalances, toxicity, and various stresses thereby providing phenotypic insight into the condition of the community. The lipid extraction procedure also allows recovery of the cellular DNA for probing with or without enzymatic amplification that adds extraordinary specificity for analysis for specific organisms, groups of organisms or potential enzyme activities. Specific signature lipid patterns of PLFA, steroids, respiratory quinones and lipopolysaccharide hydroxy fatty acids of the lipid A of gram negative bacteria allow detection of many groups of microbes with a quantitative definition of the community composition. Utilizing the SBA it has been possible to show that there are viable microbial communities in the host rock of potential subsurface waste deposit sites like Yucca Mountain. Research has shown that the viable biomass, community composition and nutritional status of the extant microbial community shifts with contamination, pollution, and disturbance. With SBA it has been possible to document that MIC is related in time and space to the corrosion process and to define which microbial communities are most likely to facilitate localized corrosion. Specific organism involved in uranium reduction can be identified by SBA. With the insight gained with SBA, predictions of potential effects microbial communities may have on the containment of nuclear wastes can be made.

INTRODUCTION

Microbes tend to be ignored because they are difficult to study. The classical methods of isolation and culture of microbes that are taught in most microbiology courses have been enormously successful in clinical medicine where isolation of specific pathogens establishes the diagnosis of disease and the in vitro sensitivities to antimicrobials can often predict the success of treatments. The obvious thing to do was to apply the same methods to repository system to detect the presence of microbes. Unfortunately often less than 1% of the microbes that can be detected in stained microscopic preparations can be cultured. Staining microbes in environmental samples like soils can be difficult as many are attached to soil granules and may be hidden. Agents that release attached microbes are often selective and do not release them quantitatively. The morphology of the microbes does not often reflect the function or activity so very little insight into the community structure or nutritional status is possible. Measurements of metabolic processes are complicated by the facts that most microbes in the soil are inactive, but poised for activity when nutrients appear. Adding labeled substrates to determine rates of metabolic activity induces major disturbance artifacts giving much higher rates than actually exist in the environment. This is possibly best exemplified in studies of the deep subsurface microbiota where oxygen and inorganic carbon are found in groundwater with a ground water age of greater than 1.1 x 105 years. Measurements of metabolic activity based on isotope incorporation experiments by the microbes in subsurface sediments were 103 to 106 times greater than the geochemical

evidence would predict. The metabolic activities by the subsurface microbiota indicate growth rates in centuries (27).

METHOD

A solution to the quantitative detection of microbes in the environment of nuclear storage is in signature biomarker analysis (SBA). We have concentrated on the analysis of lipids (30). Every living cell is surrounded by a lipid membrane. These lipids are quantitatively extracted from the microbiota in situ and analyzed using gas chromatography/mass spectrometry (GC/MS). Several unique classes of lipids, including steroids, diglycerides, triglycerides, respiratory quinones, -hydroxyalkanoate (PHA), phospholipid lipid fatty acids (PLFA), lipo-amino acids, plasmalogens, acyl ethers, sphingolipids, and lipopolysaccharide hydroxy fatty acids can be used as signature lipid biomarkers to characterize microorganisms or communities of microorganisms. Recently the lipid extraction has been shown to yield DNA suitable for gene probing and enzymatic amplification (12).

Phospholipids are one of the most important SBA classes, and are essential membrane components of living cells. Unlike most other biomarkers, phospholipids are typically degraded within hours following cell death. This rapid degradation of the phospholipids establishes the PLFA as ideal biomarkers for viable cells, thus, the quantification of total PLFA is an accurate measurement of living biomass (2). Because different groups of microorganisms synthesize a variety of PLFA through various biochemical pathways, the PLFA are effective taxonomic markers and can be utilized to provide insight into the community composition. PLFA analysis can provide insight into the phylogenetic relationships between organisms similar to phylogenetic analysis based on the sequence homology of 16S ribosomal RNA (8,15). Knowledge of specific lipid biosynthetic pathways can provide insight into the nutritional status of the microbial community as certain fatty acids, such as trans and cyclopropyl fatty acids, provide an indications of environmental stress. Other components indicate unbalanced growth where carbon sources and terminal electron acceptors abound but a critical nutrient prevents cell division but not growth or bioavaiable phosphate is insufficient (31). The redox level of the microbiota can be determined in situ by shifts in the composition of lipids in specific indicator microbes. The signature lipid biomarker techniques have been successfully applied to subsurface materials (33).

Fig. 1

RECOVERY OF MICROBES FROM SUBSURFACE

The detection of microbes from deep subsurface sediments that are possible repositories for nuclear materials requires convincing evidence that the samples recovered from drilling operations were not contaminated by the make-up water, drilling muds, or in handling. Signature lipid biomarker analysis has shown that the viable biomass, community composition, and nutritional/physiological status of the microbial communities recovered with strict sampling guidelines (3,26) were distinctive enough to assure that microbes in the pared cores were from the extant microbiota (17).

DETECTION OF IN SITU MICROBIAL COMMUNITY ACTIVITY

We have ample evidence that the subsurface microbial community responds rapidly to changing conditions and is thus metabolically active. Pollution readily induces shifts in the viable biomass, community composition, and nutritional status of subsurface microbial community

with an increase in viable biomass and increased proportions of PLFA characteristic of gram-negative heterotrophs (29). Increases in the type II methane-oxidizing bacteria were detected in soil columns gassed with methane and air (23). Addition of different fatty acid substrates to anaerobic sediment cores induced marked and expected changes in the bacterial community structure (25). Subsurface sediments perfused with methane, propane, air show shifts in community structure that correlate with trichloroethylene (TCE) biodegradation (4,28). Active biodegradation of petroleum hydrocarbons in subsurface sediments results in increases in viable biomass, shifts to aerobic heterotrophic bacterial PLFA, decrease in biomarkers indicative of stationary phase growth, decrease in PHA/PLFA ratio, and increases in the proportion of benzoquinone respiratory quinones indicative of aerobic electron transport activity (28). The nutritional status of microbial consortia actively degrading petroleum differs markedly from the organisms fortuitously degrading TCE in that effective TCE biodegradation is correlated with a build-up of reducing power indicated by a high PHA/PLFA ratio (23). Clearly the subsurface microbiota responds to shifts in the environment. Unpublished experiments from the Savannah River in situ TCE biodegradation demonstration showed that the changes detected in the recovered sediments were reflected in the ground water microbes collected as membrane filter retentates. The recovery of specific gene probes for methane monooxygenase and the signature PLFA of the methane-oxidizing bacteria correlated well.
Bacterial PHA and microeucaryotic triglyceride (6) are endogenous storage lipids. The relative amounts of these compounds compared to the PLFA, provides a measure of the nutritional status of specific components of the microbial community. Many bacteria form PHA under conditions of unbalanced growth such as when a carbon source and terminal electron acceptor(s) are present but cell division is limited by the lack of some essential nutrient (5,24). The determination of the ratio of PHA/PLFA has proved useful in monitoring the effectiveness of bioremediation in the subsurface--effective biodegradation of petroleum hydrocarbons correlates with a low ratio of PHA/PLFA (28) whereas the fortuitous metabolism of trichloroethylene correlates with a high ratio of PHA/PLFA (23,4).
Specific ratios of PLFA acids have been shown to correlate with physiological stress (7). Exposure to toxic environments can lead to minicell formation and a relative increase in PLFA specific to the exposures. For example, increased conversion from cis to trans PLFA occurs in Pseudomonas species with exposure to higher concentrations of phenol in the absence of bacterial growth (11). Prolonged exposure to conditions inducing stationary growth phase induce the formation of cyclopropane PLFA (7). The respiratory quinones, the detection of plasmalogen lipids and other biomarkers can be utilized to indicate the degree of microbial aerobic activity (10,31). Environments with high potential terminal electron acceptors (oxygen, nitrate) induce the formation of benzoquinones in bacteria in contrast to microbes respiring on organic substrates which form naphthoquinones. There are other lipid biomarkers such as lipid amino acids liberated after hydrolysis of the lipids can yield further insights into the conditions of the subsurface microbial microniches (31).

POTENTIAL FOR MICROBIAL INFLUENCED CORROSION

The application of SBA lipid technology to subsurface sediments recovered with the quality assurance that the communities sampled represent the extant microbiota clearly show the presence of a diverse and viable

microbiota that responds to geochemical and hydrologic gradients. The metabolic activities of bacteria found in the subsurface can produce significant risks to the containment of nuclear waste over extended periods of time. Microbes are associated with pitting corrosion (14). The ready destruction of concrete sewers by Thiobacilli (13) or buried gas transmission pipelines by acetogens (27) attest to the destructive power microbes. Microbially influenced corrosion (MIC) poses potentially serious problems in the maintenance of containment barriers (18). Microbes have been shown to particularly attack weldments (20) and are quite capable of localized MIC on "noble" substrata such as stainless steel (1). Because of these microbial potentials, it becomes necessary that a thorough understanding of the extant subsurface microbiota be known prior to the deposition of nuclear wastes.

TRANSPORT

Microbes have considerably higher hydraulic transmissivity than conservative tracers like Bromine (16). Microbes readily bind nuclides (21) and can thus transport nuclides very effectively. SBA of lipids has shown that subsurface microbial communities can respond to differences in the hydraulic conditions. The viable microbial biomass, based on total extractable PLFA, decreases with depth in sediments from the arid northwest. This viable biomass decrease with depth is slightly less pronounced in areas where there is a high recharge rate. An analysis of microbial community composition throughout the tested depth interval shows that PLFA indicative of Actinomycetes, in particular tuberculostearic acid (10me18:0), continue to represent a constant or increasing percentage of the total in the area of high recharge which was in sharp contrast to the decrease in percentage of these PLFA in the area of low recharge (33).

POTENTIAL EFFECTS

Microbial contamination and metabolic activity in the subsurface if controlled and monitored not be a major threat to the containment of buried nuclear waste. Microbes have a remarkable capacity to immobilize nuclides through the formation of insoluble compounds (19). Bacterial metabolic activities may have been responsible for the original concentration of uranium nuclides in sedimentary deposits and for their containment over the past billions of years at sites like Cigar Lake, Saskatchewan or in fission products like those at Oklo, South Africa.

ORGANISMS AT PROPOSED SITES (YUCCA)

Application of SBA lipids to potential sites has shown that microbes are present in the uncontaminated volcanic tuft of the Yucca Mountain, NV site. Concentrations recovered from the aseptic (in so far as possible) carefully prepared sites show bacteria in concentrations equivalent to 102-104/gm. Some are cuturable and show diversity and heterogeneity in their distribution (9). There are large amounts of as yet uncharacterized glycolipids and terminally branched saturated PLFA characteristic of gram-positive bacteria. No samples have been assessed after the tunnel preparation which would be expected to increase the diversity, activity and biomass of the microbes. Simply aseptically grinding the tuft in the absence of any added nutrients or water significantly stimulates microbial growth and activity (P. Amy and D. Haldeman personal communication).

ACKNOWLEDGMENT

This work has been supported by grant DE-FG05-90ER60988 from the Subsurface Science Program, administered by F. J. Wobber, from the U. S. Department of Energy.

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45-2

MICROBIAL GAS GENERATION FROM CANDIDATE BACKFILL FOR CANADA'S NUCLEAR FUEL WASTE DISPOSAL VAULT

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ABSTRACT
Atomic Energy of Canada Limited is developing a concept for the disposal of nuclear fuel waste in an engineered vault 500 to 1000 m deep in plutonic rock of the Canadian Shield. The concept is based on a multi-barrier system. The fuel waste, isolated in corrosion-resistant containers, would be placed in disposal rooms and the containers would be surrounded by compacted buffer material (50 wt.% sodium bentonite and 50 wt.% silica sand). The rooms and connecting tunnels would be backfilled with a mixture of 75 wt.% crushed and graded host rock and 25 wt.% glacial lake clay. The final barrier would be the host plutonic rock surrounding the vault.
The backfill clay would be mined locally. The reference clay, that exhibits the appropriate sealing properties, is Lake Agassiz clay, deposited ~8,000 years ago. This clay contains 1 to 2% organic matter. The question this work addresses is "Is it possible that microbes, naturally present in the clay and groundwater and introduced during construction of a vault, could use this organic material and generate gas that could affect the integrity of the backfill barriers?". This paper describes how microbial gas production from natural organic matter present in Lake Agassiz clay is being investigated. The gases most likely produced are CH4, H2S and H2 because of the low reduction-oxidation potential expected with time in the backfill environment. A methane production rate of 48 mg CH4/kg clay.d over 193 days was observed in a Lake Agassiz clay/groundwater mixture. However, reducing conditions were achieved using an addition of thioglycolic acid, an O2-scavenger and an additional C source that may be easily-utilized by microbes. Thus, this

methane production rate may not be applicable to a nuclear vault system, and is not even appropriate as an upper bound.

INTRODUCTION

Atomic Energy of Canada Limited is developing a concept for the disposal of nuclear fuel waste in an engineered vault 500 to 1000 m deep in plutonic rock of the Canadian Shield (1). The concept is based on a multi-barrier system. The fuel waste, isolated in corrosion-resistant titanium or copper containers, would be emplaced in disposal rooms or in boreholes drilled in the floor of disposal rooms (2, 3). The containers would be surrounded by compacted buffer material (50 wt.% sodium bentonite and 50 wt.% silica sand) that would swell when saturated with groundwater (4) and ensure that transport of any contaminants was controlled by diffusion. After waste emplacement, the rooms and connecting tunnels would be backfilled with a mixture of 75 wt.% crushed and graded host rock and 25 wt.% glacial lake clay. The final barrier would be the host plutonic rock surrounding the vault (5).

The backfill clay would be mined locally, and the backfill would be prepared on site in sufficient quantities to fill the shaft and adjoining rooms. Since such an engineered facility would be large, some 9.6 million m3 of this material would be required (6). The reference backfill clay, that exhibits the appropriate sealing properties, is Lake Agassiz clay, deposited some 8,000 years ago (7). This clay contains 1 to 2% organic matter (8). It is possible that microbes, naturally present in the clay and groundwater and introduced during construction of a vault, could use this organic material and generate gas that could affect the integrity of the backfill barrier.

The objective of the work presented here is to demonstrate and test the techniques that would be used to measure microbial gas production from natural organic matter present in Lake Agassiz clay. The gases most likely produced are CH_4, H_2S and H_2 because of the low reduction-oxidation potential expected with time in the backfill environment (3). Experimental results are presented and discussed for experimental systems containing deep crystalline rock groundwater, groundwater and clay, and groundwater and backfill.

METHODS AND MATERIALS

Experimental Vessels

A 2-L fermentation vessel capable of holding a large flooded backfill clay sample (0.6 kg dry wt.), including a headspace for gas sampling, was used (Fig. 1). The vessel was fitted with probes through a septum in one of the headplate ports to continuously monitor the pH and Eh in the clay. Water samples could also be removed through this septum. Most gas samples were taken under vacuum through a Nupro valve attached to a second port. A pressure gauge was attached to measure headspace pressure. A strip-paper anaerobic indicator was positioned in the headspace of the vessel to allow a visual check for anaerobic conditions. The other ports were sealed to maintain an air-tight system.

Fig. 1

Materials

Groundwater from AECL's Underground Research Laboratory (URL) northeast of Lac du Bonnet, Manitoba, Canada, was collected from a borehole (HG4-4), intersecting a rock fracture zone identified as Fracture Zone 2 (FZ2), at a depth of 130 m (9). The properties of the Lake Agassiz clay are reported elsewhere (10, 4, 8). Radio-carbon dating, performed during this study, showed the age of the organic matter in the clay to be 23,950

270 years, containing about 5% 0.2 of modern 14C. Total organic carbon of a clay sample, analyzed using the Walkley-Black method was 7.50 mg/g or 0.75% organic carbon. The total inorganic carbon analyzed by combustion in a LECO furnace was 9.30 mg/g. The clay was air-dried, ground and sieved through a 850-m screen. A 2:1 slurry of groundwater (1230 mL) to clay (615 g) was poured into the fermentation vessel leaving a headspace of about 310 cm3. Before adding the groundwater/clay slurry, the fermentation vessel and headplate with attached probes were washed with alcohol. The slurry was left to settle, forming a layer of "surface" water on top of the clay. The fermentation vessel was wrapped with aluminum foil to prevent light infiltration and kept at room temperature (221C) for 212 d. Thioglycolic acid (C2H3O2SNa, FW 114.1, 1.125 g), added to the vessel as an O2 scavenger, forced the system to go anoxic after 32 d.

Sampling

Gas samples, withdrawn through the septum using a 10 mL Gastight syringe under vacuum through a Nupro valve were analyzed with a VGMM8-80 Mass Spectrometer for CH4, O2, CO2, SO2 and H2S. Biological Activity Reaction Tests (BARTs) for iron-related bacteria (IRB-BART), sulphate-reducing bacteria (SRB-BARTTM) and slime-forming bacteria (SLYM-BARTTM) were scored, as recommended, on the groundwater before use, and on the surface water and clay pore water at the end of selected experiments.

Microbe counts, using a modified epifluorescence microscopy counting procedure (9) were carried out in duplicate on samples of the groundwater prior to mixing with the clay. A 5-mL sample of surface water was withdrawn from the vessel during the course of the experiment for microbe counts. Final microbe counts were performed on the surface water and pore water extracted from the clay.

Total water chemistry (anion/cation by Varian SpectrAA-400, DIONEX QIC Ion Chromatograph and ICPS, ARL-3560) and total inorganic (TIC) and organic carbon (TOC) analyses (persulphate method, Astro 2001-MB) were performed on the groundwater prior to use.

Dismantling of the Vessel

At the conclusion of the experiment, the surface water from the vessel was siphoned off and the clay removed at depths of 0-4.5, 4.5-9.0 and 9.0-13.5 cm. The clay was placed into 50 mL vials and centrifuged at 1000 rpm (Relative Centrifugal Force=251 m/s2) for 20 minutes, to yield the clay pore water. The clay adhering to the headplate probes (which comprised a central plug) was scraped off and centrifuged separately. Microbe counts were done on a 10-mL sample of the surface water and on the pore waters with depth. Analyses for S (by titration of SO2 released during combustion in a LECO induction furnace), P (colorimetrically after H2SO4/HF/HNO3/HCl digestion), and N (total Kjeldahl nitrogen) (11) were performed on the clay from each of the sampled depths and adhering to the headplate probes. The remaining pore water was combined to provide sufficient volumes for BARTs, total inorganic and organic carbon, and total water chemistry analyses.

RESULTS AND DISCUSSION

Gas Analysis

The results of the gas analyses from the first experiment are shown in Table I.

Table I

Concentrations of H2 sampled from the headspace were low throughout the sampling period. Detectable amounts of CH4 were observed after only 7

days of closure of the fermentation chamber, although the oxygen levels were still high (Fig. 2) and the anaerobic strip paper indicated the system still contained O2. The system went anoxic after 32 d, when analysis showed an order of magnitude drop in the O2 concentrations (Table I, Fig. 2). The pH (~7.8) of the clay slurry remained constant throughout the sampling period. A substantial increase in the amount of CH4 produced was observed after 50 d and peak CH4 production rates of about 48 g CH4/kg clay.d were reached between 50 and 81 d, whereafter it remained constant for another 100 d (Fig. 2). This CH4 production was offset, as expected, by a drop in the O2 levels in the system (Fig. 2). A slight drop in CH4 production was observed after 119 d when a pore-water sample was removed for microbial counts which introduced air into the system (Table I). After 151 days the CH4 production dropped to 28 g/kg clay.d (Fig. 2), possibly indicating the development of less favorable conditions for methanogens. The gas composition throughout the experiment indicated that CH4 production was accompanied by CO2 production, with a small drop in N2 (Table I). Results from further experiments in which no thioglycolic acid was added, are in progress.

Fig. 2

Groundwater and Clay Analyses

The clay of this experiment contained 7.5 mg/g organic carbon (i.e.,13.3 mg/g organic matter) and 9.3 mg/g inorganic carbon. The total inorganic carbon (TIC) in the surface water of the fermentation vessel after 164 days increased from 305 mg/L to 901 mg/L (the error bars are standard deviations based on three replicate samples). The total organic carbon (TOC) increased from 0.40.02 mg/L to 45513 mg/L. The TIC increase is probably a result of the slow dissolution of the clay carbonate minerals in the groundwater. The TOC increase is partially due to the thioglycolic acid (187 mg C/L) and to the dissolution of organics from the clay. The TOC of the clay accounted for 4600 mg C. Although microbe counts increased with time (13), microbial growth did not account for the increase in TOC (13). There was no consistent variation in the pore-water microbe counts with depth in the clay sediment and significantly more microbes were associated with the pore water and clay than with the surface water.

Calculations show that microbial cells (105-107 cells/mL), assuming one cell contains between 10-13 to 10-15 g C, account for a minimum of 0.1 g C/L and a maximum of 1 mg C/L. Calculations of limiting nutrients in the system and populations of microbes that can be supported by these nutrients were carried out and are discussed elsewhere (12, 13).

Biological Activity Reaction Tests (BARTs) and Major Ion Analysis

The BART tube tests on the initial groundwater indicated a significant Fe-related bacteria (IRB) population with a very aggressive slime-forming bacteria (SLYM) population and a small sulphate-reducing bacteria (SRB) population (13). After 212 days, the SRB population increased to significant levels in the surface water while the IRB population, associated with the pore water, was highly aggressive. The initial population supported by the IRB tubes was significant despite the concentration of Fe in the initial groundwater being very low at 0.043 mg Fe/L. However, the Fe and other dissolved ion concentrations in both the pore water and surface water increased substantially during the experiment, suggesting some dissolution of the clay. After 212 days, the concentration of Fe and other dissolved ions in the surface apparently allowed the bacteria to become highly aggressive (13).

Concentrations of total S increased dramatically from 13.7 mg S/L to 897 mg S/L (surface water) and 1280 mg S/L (pore water), presumably due to dissolution of gypsum in the clay, allowing the sulphate-reducing bacteria to increase. The concentration of P in the groundwater was low enough to be considered a limiting nutrient to the microorganisms (13). However, there was probably sufficient P in the clay for microbial growth. Nitrogen concentrations also increased slightly in the pore water, however, N may still be a limiting factor for microbe growth in this experimental system.

METHANE PRODUCTION

Methane production rates of 48 g CH4/kg clay.d over 193 days of gas sampling gave a total CH4 production of 35 mg of CH4. Using the ratio of C to CH4, shows that 20 mg of C were converted to CH4. The original groundwater contained 1.0 x 105 microbes/mL. Initially there were 1.2 x 108 microbes in the system in the 1230 mL of groundwater alone. Surface and pore water after some 200 days contained 6 to 12 x 106 microbes/mL, for a total of at least 7 x 109 microbes. Assuming the population increased at least two orders of magnitude, and assuming 1013 to 1015 microbes contain 1 g C, an insignificant amount of C (1 x 10-11 g) has been converted to microbial material.

The addition of the thioglycolic acid may have provided the microbes with an additional source of C. The 1.12 g of thioglycolic acid provided 0.27 g of C, enough easily-utilized C to produce this amount of CH4. Since there was enough thioglycolic acid alone to keep this system running another two years, without having to use the perhaps more recalcitrant C of the groundwater or the clay, this system was stopped. Other experiments show methane production in groundwater without the use of thioglycolic acid.

Most likely there will be limiting nutrients (N and P) for the CH4-producing microbial population. Nitrogen has been found to be the most limiting nutrient in flooded soils (14). Oxidation of CH4 as it enters the biosphere (15) will also consume CH4 arising from depths. There are a number of additional factors that must be considered such as the effect of pressure, temperature and radiation on microbial populations within a nuclear fuel waste vault, the moisture content of the buffer material (16, 9), the concentration of salt in the groundwater, as well as the degree of fracturing in the bedrock and distance to the surface.

CONCLUSIONS

We have discussed several techniques that were used (and improved) throughout the course of a backfill gas production experiment. Methane was produced in a laboratory vessel containing Lake Agassiz clay and granitic fracture zone water. A methane production rate of 48 mg CH4/kg clay.d over 193 days of gas sampling was reached. Reducing conditions, however, were achieved using thioglycolic acid, an O2-scavenger and an additional C source that may be easily-utilized by microbes. Thus, this methane production rate may not be applicable to a nuclear vault system, and is not even appropriate as an upper bound. Calculations show N, P and perhaps even C in the groundwater/backfill may limit microbial growth in a nuclear fuel waste vault.

ACKNOWLEDGEMENTS

We thank Connie Hamon for microbe counting and BART tube analysis instruction, Grant Delaney and Merle Brown for gas analyses, Jay Hawton for groundwater collection, Dave Dixon and Dave Woodcock for providing Lake Agassiz clay and information on its source location and Lisa Lucht

for the loan of the fermentation vessel. The Canadian Nuclear Fuel Waste Management Program which has supported this work, is jointly funded by AECL and Ontario Hydro under the auspices of the CANDU Owners Group.

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45-3
MICROBIOLOGICALLY INFLUENCED CORROSION OF DISPOSAL CONTAINERS FOR RADIOACTIVE WASTES

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ABSTRACT

Microbiologically influenced corrosion has been documented for all metals and alloys currently in use or proposed for packaging nuclear wastes, with the possible exception of titanium. Selection of a material that will provide a reliable physical barrier to prevent leakage of radionuclides will depend on a thorough understanding of mechanisms for microbiologically influenced corrosion, the geological/microbiological/chemical environments of storage sites and their evolution over time.

INTRODUCTION

Microbial biofilms develop on all surfaces in contact with aqueous environments. Biofilms composed of immobilized cells embedded in an organic polymer matrix are absorptive and porous, and contain solutes, heavy metals, and inorganic particles, in addition to cellular constituents (1). Biofilm/surface interfacial chemistry can be radically different from that of the bulk medium in terms of pH, dissolved oxygen, organic and inorganic species. Microbiologically influenced corrosion (MIC) is usually used to designate corrosion due to the presence and activities of microorganisms within biofilms at metal surfaces. It is important to extend the definition to include corrosive microbial metabolites that may be produced in one location and diffuse to a corrosion site. MIC is localized corrosion that results in pitting, selective leaching, crevice corrosion, under-deposit corrosion and enhanced erosion corrosion. Microorganisms can accelerate rates of partial reactions in corrosion processes and shift corrosion mechanisms. MIC of storage containers is a potential problem for safe disposal of nuclear wastes. Primary materials being considered for containment of high level nuclear waste in mined geological repositories are alloys of iron, nickel, copper and titanium. Iron and nickel are significant components of alloys being proposed for containment of low- and intermediate-level nuclear waste in either shallow land burial or deep disposal. Aluminum and stainless steel alloys are currently used in the construction of pools and racks for storage of spent nuclear fuels. Carbon steels have been used to contain nuclear wastes in marine sediments (2) and are being considered as suitable materials for the outer shell of a double shell storage container for subsurface burial. With the exception of titanium, MIC has been documented for all metals and alloys under consideration. However, because of a lack of information about subsurface/oligotrophic microbiology and the belief that g-radiation from fuel within containers would effectively sterilize storage environments, recognition of potential problems has been neglected. In

this paper mechanisms and case histories for MIC will be reviewed with particular attention to metals and alloys currently in use or proposed for nuclear waste storage.

Biofilm Formation

Presence of water, availability of nutrients/energy sources and tolerance of microorganisms to high temperature, ionizing radiation and concentrated solutes will influence development of a substantial population of microorganisms on or around nuclear waste storage containers. There is increasing evidence that microorganisms exist and flourish in unlikely environments, including radioactive and nutrient-deficient waters and deep subsurface environments. Algae, fungi, yeasts and bacteria were found in water covering the damaged reactor core at Three Mile Island (TMI) nuclear power plant (3). Bacteria survived on carbon-rich hydraulic fluid leaked from tools during defueling after exposure to g-radiation doses of 10 Gy h-1 (4). Dose rate around a nuclear waste storage container decreases with time as waste decays and decreases with increasing distance because of attenuation in buffer material.

Groundwaters in geological deposits are often saline and typically contain 101 to 105 colony-forming units ml-1 (5). Three significant biofilms containing bacteria have been identified at the Underground Research Laboratory (URL) in the Archean Lac du Bonnet Batholith in southeastern Manitoba: one at the drill hole draining through an underground structure (420 m depth), an extensive biofilm covering walls kept wet by groundwater flowing from fractures and exposed to gaseous carbon and nitrogen-containing explosives (240 m depth), and one found on the side of a large borehole (130 m depth) irrigated by water from drilling operations. The indigenous microbial population in the Canadian design disposal vault was relatively small (103 to 105 microbes cm-3) compared with surface waters (6). However, Stroes-Gascoyne (7) concluded that construction activity will introduce both microorganisms and nutrients. White et al. (8) found biochemical evidence that bacterial populations equivalent to 107 bacteria gm-1 sediment existed at a depth of 410 m in the Bacatunn clay formation near Pensacola, FL. Viable microorganisms were recovered in Nevada from ashfall tuff in a mined tunnel 350-450 m below the surface and hundreds of meters above the regional water table. Microorganisms adhere to rock surfaces and form complex, sessile biofilm populations that intervene between rock and groundwater (9). In soils and on rock surfaces, microorganisms are generally not uniformly distributed but accumulate in discrete microcolonies attached to particle surfaces to form a system composed of more or less discontinuous colonies, each in its own microhabitat (10). In natural oligotrophic environments the formation of biofilms is a survival strategy for bacterial communities. Little et al. (11) reviewed factors influencing adhesion of microorganisms to substrata. Roughness and surface chemistry of the substratum play a major role in biofilm processes during early stages of biofilm accumulation and may influence rate of cell accumulation and distribution in aquatic environments. Electrolyte concentration, pH and inorganic ions influence settlement. Hydrated oxide and hydroxide passivating films on metal surfaces provide bacteria with sites for firm attachment. Similarly, spalling or sloughing of corrosion products forces detachment of biofilms associated with corrosion products. Carbon is not always the growth limiting nutrient for microorganisms. Phosphorus and nitrogen may be limiting in some systems.

Wilkinson (12) found that, when the concentration of nitrogen in a growth medium was gradually lowered until it became limiting with C/N ratios in the range 7-10, cells growing in such a medium tended to reproduce slowly and produced copious amounts of extracellular polymer. Hydrodynamic shear stress, related to flow, influences transport, transfer and reaction rates within biofilms, as well as detachment. Temperature influences the rate of most chemical and biochemical reactions as well as transport processes within the biofilm.

In several proposed storage concepts, used nuclear fuel would be encapsulated in a metal container and buried in an underground disposal site located 500 to 1000 m subsurface. Containers would be surrounded by a compacted mixture of silica sand and sodium bentonite, known as buffer material. Safety is based on a series of barriers designed to prevent or delay toxic radionuclide species from coming into contact with man. The only absolute barrier in the system is the container. All other barriers are porous. Sodium bentonite is a cation-exchanging smectite clay mineral that swells upon contact with water. Mass transport through the buffer is expected to occur only by diffusion. Remaining void spaces will be air-filled, and entrapped oxygen will be the major source of oxidants. Depending upon specific disposal design, containers will be embedded in buffer material in either excavated rooms or boreholes filled with clay, crushed rock, and/or concrete. The next barrier in the system is 500 to 1000 m of granitic rock,clay, salt, limestone, iron ore, gypsum, bitumen, ash or soil.

Storage environments in geological repositories and their microflora will change with time. Initially, conditions will be hot and oxidizing as radioactive decay of fuel produces heat and g-radiation. Initial storage environments will be radioactive, hot, dry and nutrient deficient. Combined effects of elevated temperature, g-radiation and desiccation of the buffer material will severely limit the extent of microbial activity close to the container and might result in an abiotic zone 40 cm into the compacted buffer material (13). As fuel decays and trapped oxygen is consumed, conditions are expected to become cool and anoxic. As the container surface temperature decreases, the buffer material will become fully saturated with ground water. Length of the resaturation period could be several to several thousand years. Microbial transport and growth during resaturation will be limited because of spatial restrictions within the buffer material (mean pore size 0.1 to 0.5 m), absence of large amounts of unbound water and availability of nutrients.

MECHANISMS FOR MICROBIOLOGICALLY INFLUENCED CORROSION

Concentration Cells

The physical presence of microbial cell surfaces, in addition to their metabolic activities, modifies electrochemical processes. Adsorbed cells grow, reproduce and form colonies that are physical anomalies, resulting in local anodes and cathodes and formation of differential aeration cells. Under aerobic conditions, areas under respiring colonies become anodic and surrounding areas become cathodic. The anodic reaction will be metal dissolution and the cathodic reaction will be oxygen reduction. Mature biofilms prevent diffusion of oxygen to cathodic sites and diffusion of aggressive anions, such as chloride, to anodic sites. Outward diffusion of metabolites and corrosion products is also impeded. If areas within the biofilm become anaerobic, i.e., if the aerobic respiration rate within the biofilm is greater than the oxygen diffusion

rate, cathodic reactions can include reduction of water molecules, hydrogen ions, hydrogen sulfides or hydrated metal ions.

Reactions Within Biofilms

It is traditional to discuss microorganisms within biofilms as aerobic or anaerobic and as individual species. However, microorganisms form synergistic communities that conduct combined processes that individual species cannot. Cell death or lysis within a well-developed biofilm does not necessarily mean a cessation of influence on electrochemical processes (14). Volatile corrosive gases such as hydrogen, hydrogen sulfide, carbon dioxide or ammonia produced by microorganisms might permeate porous backfill material and contact container surfaces.

Volcanic ash, granite, clay buffers, soil and seawater contain sulfate ions and sulfate-reducing bacteria (SRB) are common in those environments. Seawater contains 1 to 10 viable SRB ml-1 irrespective of depth or sample site. Marine bottom muds contain 102 to 105 viable cells gm-1 (15). SRB were found in approximately 25% URL ground water samples (16). SRB were isolated from British bentonite, suggesting that there were strains that could survive in the clay environment (17). In anaerobic, sulfate-containing media, SRB use sulfate ions as terminal electron acceptors, producing hydrogen sulfide that reacts with metals to form sulfide corrosion products. In addition, SRB can stimulate the cathodic reaction by removal of hydrogen.

When conditions around storage containers become wet, cool, and anoxic microbial activity is possible in the immediate vicinity of storage containers. It has not been established that individual cells will be able to traverse buffer materials to reach container surfaces and form a biofilm. If the abiotic zone is repopulated, SRB are expected to dominate the microflora and SRB-influenced corrosion might be possible under the biofilm. However, the most aggressive corrosion of metals and alloys in the presence of SRB is observed when there are successive, alternating aeration-deaeration shifts. It is extremely unlikely that such shifts could occur under any disposal scenario. If repopulation of the abiotic zone cannot occur, SRB activity will be restricted to peripheral areas and impact on the container will result from diffusion of reduced sulphur species. King and Stroes-Gascoyne (18) demonstrated that inorganic sulfide will diffuse through a clay layer. It has not been demonstrated that sulfides produced by SRB will behave similarly. Most heterotrophic bacteria secrete organic acids during fermentation of organic substrates. The kinds and amounts of acids produced depend on the type of microorganisms and available substrate molecules. Organic acids may force a shift in the tendency for corrosion to occur. The impact of acidic metabolites is intensified when they are trapped at the biofilm/metal interface. Acetic acid from Clostridium aceticum and sulfuric acid produced by sulfur-oxidizing bacteria (SOB), such as Thiobacillus thioxidans, are obvious contributors to corrosion. Organic acids of the Krebs cycle can promote electrochemical oxidation of a variety of metals by removing or preventing the formation of an oxide film. Acid-producing fungal strains isolated from soil and autotrophic SOB degraded cement used as a coating material for low- and intermediate-level waste (19). Many organisms produce ammonia from the metabolism of amino acids or the reduction of nitrite or nitrate. In solution, ammonia forms NH_4^+ that can react with metal surfaces (20).

Metal-depositing organisms create environments that are conducive to corrosion. Some microorganisms catalyze the oxidation of metals, others

accumulate abiotically-oxidized metal precipitates, and still others derive energy by oxidizing metals (21). Iron-oxidizing genera that are usually cited as causing MIC are Gallionella, Sphaerotilus, Crenothrix and Leptothrix. These organisms oxidize ferrous ions to ferric ions or manganous ions to manganic ions to obtain energy. Dense deposits of cells and metal ions create differential aeration cells that effectively exclude oxygen from the area immediately under the deposit. Under-deposit corrosion initiates a series of events that are individually and collectively extremely corrosive. In an oxygenated environment, the area immediately under the deposit becomes deprived of oxygen. That area becomes a relatively small anode compared to the large surrounding oxygenated cathode. Cathodic reduction of oxygen may result in an increase in pH of the solution in the vicinity of the metal. The metal will form metal cations at anodic sites. If the metal hydroxide is the thermodynamically stable phase in solution, metal ions will be hydrolyzed by water with formation of H+ ions. If cathodic and anodic sites are separated from one another, the pH at the anode will decrease. The pH at the anode depends on specific hydrolysis reactions. The largest pH decreases are observed for alloys containing Cr+3 and Mo+3. For this reason, under-deposit attack due to metal-depositing bacteria is particularly aggressive on stainless steels. In addition, Cl- ions from the electrolyte will migrate to the anode to neutralize any buildup of charge, forming heavy metal chlorides that are extremely corrosive. Under these circumstances, pitting involves the conventional features of differential aeration, a large cathodic to anodic surface area ratio and development of acidity and metallic chlorides.

Walch and Mitchell (22) proposed the following roles for microorganisms in hydrogen embrittlement of metals: 1) production of molecular hydrogen during fermentation, which may be dissociated into atomic hydrogen and absorbed into metals, 2) production of hydrogen ions via organic or mineral acids which may be reduced to form hydrogen atoms at cathodic sites, 3) production of hydrogen sulfide which stimulates absorption of atomic hydrogen into metals by preventing its recombination into hydrogen molecules, and 4) destabilization of metal oxide films.

Microorganisms produce polymers and form gel matrices of bacterial exopolymers central to the structural integrity of microbial films. Polymeric gels can immobilize water, trap metal species/corrosion products and decrease diffusion. In general, exopolymers are acidic and contain functional groups that concentrate metal ions to form metal concentration cells. Geesey et al. (23) developed conceptual models for the acceleration of copper corrosion as a result of Cu+2 bound within exopolymers.

CASE HISTORIES

Ferrous Alloys

Eidsa and Risberg (24) and Sanders and Hamilton (25) analyzed microbial corrosion of carbon steels in North Sea oil exploration. SRB, SOB, hydrocarbon-oxidizing bacteria, iron-oxidizing bacteria, slime-forming bacteria and fungi were identified as causing corrosion. The following forms of SRB-mediated corrosion were defined: pitting caused by SRB growing in the biofilm on metal surfaces, sulfide-induced stress corrosion cracking (SCC), hydrogen-induced cracking, and blistering caused by hydrogen permeation in high dissolved sulfide conditions.

Marsh et al. (2) assessed the feasibility of disposing of heat-generating nuclear waste in deep ocean sediments in carbon steel containers that

would not be breached for 1000 years. They concluded the following: 1) the main threat to long term integrity of containers was the external environment, 2) carbon steel would not be vulnerable to localized corrosion under disposal conditions, and 3) any corrosion threat could be accommodated by making container walls sufficiently thick. The authors did not consider MIC and the specific presence of SRB in marine sediments in their predictions. Carbon steel has also been considered as a container for geological disposal (26). These authors considered three types of corrosion: corrosion by oxygen, corrosion by reduction of water, and MIC. During 1000 years service life the contribution of SRB to localized corrosion was estimated to be 1.8 mm. Overall wall thickness required to accommodate the three types of corrosion was estimated at 300 mm.

Corrosion resistance of stainless steels is due to the formation of a thin passive film. Passivity can break down under the following, usually localized, environments:

1. Dilute and concentrated HCl, HBr and HF and salts that hydrolyze these acids,
2. Oxidizing chlorides such as $FeCl_3$, $CuCl_2$ or NaOCl,
3. Seawater, except for brief exposures or when cathodically protected,
4. Some organic acids including oxalic, lactic and formic acids.

Microorganisms can produce the organic acids listed above. In addition, they can set up conditions for the formation of HCl and heavy metal chlorides. A persistent problem with the use of stainless steels is their susceptibility to crevice corrosion. Kobrin (27) identified iron- and manganese-oxidizing bacteria with localized corrosion of 304 and 304L stainless steels almost exclusively at weld seams. Pope (28) and Soracco et al. (29) described several cases of MIC in fossil fuel and nuclear power generating plants. Similarly, Puckorious (30) described the sudden failure of 304 stainless steel condenser tubes at an electric utility. There are no documented corrosion failures that can be attributed to MIC for the super stainless steels containing 6% molybdenum or more.

Sridhar and Cragnolino (31) evaluated localized corrosion of 316L stainless steel for possible storage of high-level nuclear waste materials in geological deposits without any specific reference to MIC. However, there are numerous reports of pitting and crevice corrosion due to MIC for this alloy exposed to sea, lake and service waters (11). Localized corrosion may be an important failure mechanism due to the propensity for under-deposit corrosion. Miller et al. (32) exposed 304L stainless steel samples in both vertical and horizontal planes to an inoculum of microorganisms from the TMI reaction vessel. Welded and non-welded vertical specimens showed no damage for twelve months. After twelve months, however, samples placed in a horizontal plane showed signs of corrosion at the weld.

Copper Alloys

Copper alloys are vulnerable to the following types of MIC: differential aeration cells, selective leaching, under-deposit corrosion and cathodic depolarization. Pope (28) proposed that the following microbial products accelerate localized attack: CO_2, H_2S, NH_3, organic and inorganic acids; metabolites that act as depolarizers; and sulfur compounds such as mercaptans, sulfides and disulfides. Pure copper and high-copper alloys are susceptible to SCC in ammonia-containing environments and in solutions containing nitrite or cupric acetate. Such environments might

form within biofilms. Geesey et al. (23) demonstrated that extracellular polymers produced by bacteria play a role in corrosion of copper.
The impact of sulfides on corrosion of copper alloys has received considerable attention. In soils, sulfides increased the corrosion current of brass by a factor of 40 (33). Alloy 90/10 Cu/Ni suffered accelerated corrosion attack in seawater containing 0.01 ppm sulfide after 1-day exposure (34). In the presence of waterborne sulfides, copper alloys form a porous layer of cuprous sulfide. Copper ions migrate through the layer, react with more sulfide, and produce a thick, black scale. Even if such a sulfide film were technically passivating, the film's mechanical stability is so poor that sulfide films are useless for corrosion protection. In the presence of turbulence, the loosely adherent sulfide film is removed, exposing a fresh copper surface to react with sulfide ions. For these reasons turbulence-induced corrosion and sulfide attack of copper alloys cannot easily be decoupled. In the presence of oxygen, the possible corrosion reactions in a copper sulfide system are extremely complex because of the large number of stable copper sulfides, their differing electrical conductivities, and catalytic effects. Transformations between sulfides, or conversions of sulfides to oxides, result in changes in volume that weaken the attachment scale and oxide subscale, leading to spalling (35). Bared areas repassivate, forming cuprous oxide. Neither circumstance is relevant for copper nuclear waste containers. Once anoxic conditions have been established within the disposal vault, they should remain anoxic indefinitely and there should be no mechanical disruption of intact sulfide layers should they form.
Copper canisters with 100 mm thick walls are being considered for final disposal of waste material from nuclear reactors by Swedish companies building nuclear power installations. Canisters would be placed in igneous rock 500 m below ground level. Copper plates that have been buried in three separate locations for more than 50 years as earth electrodes for lightning conductors were studied to determine the geochemical/microbiological properties of the ambient soil. Microbial activity was very low in comparison to that found in marine and brackish water sediments. Predominance of sulfides in the corrosion products did not correlate with redox potentials measured at the time of collection. Similarly, sulfide in corrosion products did not correlate with sulfate concentrations in soils. The pitting factor for two of the samples was five while the third sample was not pitted. Pitting factor is the ratio between maximum pit depth of a certain area and the mean depth of general corrosion in the same area (36).

Nickel Alloys

Commercially pure nickel and its alloys are susceptible to acid production at metal/biofilm interfaces and dealloying due to reactions with sulfide. Passivity of nickel is due to the formation of an oxide or hydrated oxide film several nanometers thick. Formation of protective films on nickel is aided by the presence of iron, aluminum and silicon. Passive films similar in structure to that observed on pure nickel are formed on Ni/Cu alloys having more than 30% nickel. Alloys containing less than this amount of nickel behave like copper. In high velocity seawater, nickel alloys are superior to predominantly copper alloys because the protective surface film remains intact under highly turbulent and erosive conditions.
Nickel alloys are susceptible to pitting and crevice corrosion attack under stagnant conditions. Monel has a marked tendency for the initiation

of pitting in chloride-containing environments where the passive film can be disturbed. Under stagnant conditions chlorides penetrate the passive film at weak points and cause pitting attack. Sulfides can cause either a modification of the oxide layer as described for copper or breakdown of the oxide film of nickel alloys. Brennenstuhl et al. (37) indicated that pitting of 3 nickel alloys containing 21-27% chromium exposed to untreated freshwater from Lake Ontario involved biofilm formation.

Aluminum and Aluminum Alloys

The corrosion resistance of aluminum and its alloys is due to an aluminum oxide passive film that is 20 to 100 thick for air-formed films. Anodizing produces thicker insulating films and better corrosion resistance. The susceptibility of aluminum and its alloys to localized corrosion makes it particularly vulnerable to MIC. After 12 months both Al 1100 (commercially pure aluminum) and Boral (a composite with a layer of boron carbide dispersed in aluminum and sandwiched between two sheets of Al 1100) showed pitting with accumulation of corrosion products in and over pits after exposure to organisms isolated from the TMI reactor vessel (32). Al 1100 also showed signs of crevice corrosion due to MIC. Kalaiyappan et al. (38) demonstrated MIC of Al 1100 and Al 6061 after a few weeks exposure to deionized water in a storage basin.

Titanium and Titanium Alloys

There are no documented case histories of MIC of titanium and its alloys. Schutz (39) reviewed titanium's resistance to MIC by discussing mechanisms for MIC and titanium's corrosion behavior. He concluded that at temperatures below 100C titanium is not vulnerable to SOB, SRB, acid-producing bacteria, differential aeration cells, chloride concentration cells or hydrogen embrittlement. Formation of occluded regions of low pH underneath a biofilm could conceivably initiate crevice corrosion in titanium alloys (18).

CONCLUSIONS

Materials being considered for containment of nuclear waste in mined geological repositories are alloys of iron, nickel, copper and titanium. With the exception of titanium, MIC has been documented for all metals and alloys under consideration. The likelihood that MIC of nuclear waste storage materials will occur is directly related to the geological/microbiological/chemical environment of the storage site and vulnerability of the container material. Combined effects of elevated temperature, radiation and desiccation of buffer materials will severely limit the extent of microbial activity close to the container and might result in an abiotic zone into the compacted buffer material. If the abiotic zone is repopulated, SRB are expected to dominate the microflora and SRB-influenced corrosion might be possible for copper and nickel alloys and carbon steels.

ACKNOWLEDGMENTS

This work was supported by the Office of Naval Research, Program Element 0601153N, the Defense Research Sciences Program, NRL Contribution Number NRL/PP/7333--95-0089.

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45-4

DECOMPOSITION EXPERIMENT OF LLW ON A LARGE-SCALE - PREPLANNING

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ABSTRACT

Low- and medium-level radioactive wastes generated at TVO I and II nuclear power units are disposed of in silo-like vaults of the VLJ Repository, constructed in the bedrock at the Olkiluoto site. With the present knowledge it cannot be excluded that after sealing of the repository, microbial degradation of the organic wastes and corrosion of the steel packages by carbon dioxide would lead to comparatively rapid gas generation and pressurizing of the silo. To acquire a realistic forecast for the post-closure scenario of the silo of low-and medium-level wastes, a pilot-scale experiment on microbial disintegration is being planned. The outlines of the pilot-scale experiment are presented here.

The experiment will take place in the Research Tunnel in the VLJ Repository. The volume-scale planned is 1:1000 (referring to a completely filled repository) requiring 16 drums filled with compacted waste. The waste to be used in the experiment will be sampled from the waste stream of the power plant. The experimental set-up will be installed in a large hole (diameter 1.5 m, depth 7.5 m) bored in the floor of the Research Tunnel. The entire experimental installation in the borehole will be placed inside a tightly sealed stainless steel tank. The environmental conditions during the experiment will be chosen to be as realistic as possible also including concrete as a supporting structure inside the tank for the 16 drums.

The rate of gas generation and the composition of the gas will be monitored. Also chemical composition of the water inside the tank as well as the disintegration products, both soluble and particulates, will be followed. The changes in the redox-conditions are particularly important after closing the pilot test tank. The experiment is designed to be run without the need of constant on site follow-up by personnel. However, manual sampling is needed for water chemistry follow-up, as well as for some additional specially prepared specimens containing waste materials which are planned to be withdrawn from the installation regularly.

INTRODUCTION

Teollisuuden Voima Oy (TVO) operates two boiling water reactor units at Olkiluoto, on the South-Western coast of Finland. The operation of TVO I and II power units produces annually approximately 100 m3 of low-level maintenance waste.

The waste consists of more or less dry materials accumulating mainly during the maintenance of the power plant. The majority of waste is packed into 200 l drums before it is disposed of into the VLJ Repository excavated in the crystalline bedrock near the TVO I/II nuclear power units. The repository has been in operation since 1992.

The VLJ Repository consists of two separate silo-like disposal vaults, one for the bituminized ion-exchange resin waste drums (capacity 17360 drums), and the other for the dry, low-active waste drums (capacity 24800 drums). The silos are situated at the depth from 60 m to 100 m in water saturated crystalline bedrock.

Before the transportation to the repository, the waste drums are loaded into concrete boxes, usually 16 drums per box, and then stacked layer by layer in the silos. An empty box weights 4000 kilograms. No backfill is placed around the drums or boxes. Figure 1 shows the silo concept.

Fig. 1

One tunnel in the VLJ Repository, the Research Tunnel, has been reserved for research purposes. One task in the Research Tunnel program has been the testing of the full face boring method. In the test three full-scale (diameter 1.5 and depth 7.5 m) deposition holes, suitable for spent fuel canisters, were bored down from the tunnel floor. One of the holes is available for studying biodegradation of the maintenance waste.

If biodegradation and corrosion proceed rapidly, the formed gases can fill an area in the crane hall, above the LLW silo. According to a simple gas dissipation calculation the gas generation rate of 6.000 m3/a (NTP) can lead to a gas volume of 500 m3 (at 6 bar) in the ceiling area of the crane hall. A corresponding volume of contaminated water would have to be displaced into the bedrock. This would be the most severe effect of the gas generation.

THE MAINTENANCE WASTE

Miscellaneous waste is generated mainly during the maintenance periods of the power plant. The wastes, consisting of cleaning towels, protecting sheets, thermal insulation material etc., are collected in transparent polyethylene bags. A sorting campaign on dry, compressible waste (1.6 tons) was performed 1995 giving distribution of waste as given in Table I. The average estimated biodegradability was 28 %. The average individual weight of a bag (after repackaging) was 8.7 kg.

Table I

In the compacting station a ram using a pressure of 12 tons compacts the waste bags into 200 l drums. Around 15 bags are pressed into one drum, which then contains 100 kg of waste on the average.

CONDITIONS AND DEGRADATION PROCESSES IN THE REPOSITORY

The void volume in the repository is planned to be filled with water. Because the seepage flow rate of the groundwater is very low, the filling is planned to be performed by using local brackish sea water or fresh surface water from a local pond. Contact with concrete will raise the pH of the water up to around 12.5. Oxygen will be consumed in aerobic microbial degradation and in corrosion of steel. The flow rate of the groundwater is very low. Resulting in an estimate it will take several thousands of years to change the whole water volume inside the silos once. Thus the groundwater flow is awaited to have only a negligible effect during the most active phase of the microbial degradation.

The buffering capacity of concrete is known to be high, the amount of free calcium hydroxide in the hydrated concrete being in the order of 100 kg/m3.

The concentrations of sulfate, chloride and magnesium are distinctly higher in sea water than in surface water or groundwater and could have some influence on the corrosion processes in the considered system.

MICROBIAL DEGRADATION

The main parameters affecting microbial activity are the availability of a carbon source and a suitable energy source. In environments like a repository, the most common electron acceptors for respiration are the sulfate ion and carbon dioxide. The former is used by sulfate reducing bacteria and the latter by methanogenic organisms. Some oxidized metals, like Fe^{3+}, Mn^{4+} and U^{6+} have also been suggested to act as electron acceptors in anaerobic conditions (1).

In the VLJ Repository, the site-specific factors affecting micro biological activity are.

1) anaerobic conditions
2) pH value around 12 in the concrete water
3) temperature around 8C
4) 6 bar pressure
5) darkness.

The microbe populations surviving in the repository environment are of two different origins (2,3):

1) autochthonous populations that exist naturally in groundwater and on rock surfaces in the deep subsurface surroundings. These microbes are able to survive in the above mentioned conditions except the high pH. However, studies on natural alkaline groundwater have shown that alkalotolerant and alkalophilic microbes were able to grow at pH values up to 11.1. On the other hand, the most alkalophilic organisms were aerobic (4).
2) allochtonous microbe populations that are introduced into the repository first during the construction of the facility and later within the waste drums and with water, as well as with other backfilling materials from the surface. These populations will partly be destroyed in the extreme conditions of the repository, but a part of them can be expected to adapt to the new environment.

The potential activities of microbes in a repository can be listed as follows (5):

1) gas generation
2) corrosion
3) redox reactions
4) formation of complexing agents
5) radionuclide migration.

The possibilities to estimate different activities are mainly based on indirect studies, e.g. on measurements of parameters connected with microbial metabolism. Estimations of microbial amounts can be done by direct calculation of cells trapped on microscope slides. This gives a rough estimate of the cell concentrations, because an independent environment is formed in each drum and the variation between drums can be considerable. ATP measurements could be one possibility to estimate the viability of the microbes.

Important parameters in assessing microbiological activity are the amount of dissolved carbon dioxide, amount of dissolved methane, pH, redox potential, conductivity and some dissolved ions, like sulfide. In the beginning of the experiment, estimation of biological oxygen consumption and determination of total bacterial counts might give valuable information of the aerobic-anaerobic transition phase.

ESTIMATE OF THE RATE OF GAS GENERATION

The potential of gas generation in the silo is based on the amount of corrodiable steel and carbon bound to biodegradable compounds. The role of the ionizing radiation in the gas generation will be negligible. The planned pilot experiment is designed to decrease uncertainties in the conservatively estimated rate of gas generation.

The estimated rate of disintegration for the safety assessment was here based on measurements from municipal dumping grounds, where a variety of different waste types is present. Anaerobic degradation takes optimally place at 35C and at pH values from 6.8 to 7.2. Deviation from the optimal conditions result in decreasing of the maximal potential gas amount and the rate of gas generation (6). The time dependency of the gas generation can be described by the maximal potential gas amount and the rate-constant of gas generation:.

Eq. 1

where

G = the amount of gas generated,

Ge = the gas generation potential per weight unit,

a = disintegration coefficient (1/a)

and

t = time (a).

In the buried layers of municipal dumping grounds a measured value for the disintegration coefficient is 0.07 /a at 10C. This rate of disintegration leads in ten years to a situation, where half of the degradable material is disintegrated. At dumping ground conditions the gas generation potential at 10C is Ge= 160 m3/ton. Half of the gas is methane, half is carbon dioxide. In the case of low-level waste disposal one waste drum has been estimated to contain 47 kg of disintegrable waste, which has a potential of generating 7.5 m3 of gas. Without any new evidence it is not justified to use much lower rates of disintegration than the above mentioned. In the estimations a value of 0.046 was used for coefficient a.

CORROSION OF THE DRUM

The stability of iron-water -system depends on the redox potential and pH conditions (7). Metallic iron is always thermodynamically unstable. The corrosion reactions producing hydrogen are

Eq. 2

in the pH range -1 - 9 and.

Eq. 3

in the pH range 9 - 13.6.

It has been estimated that microbially assisted corrosion based on carbon dioxide could take place (8, 9). In the corrosion reaction carbon dioxide is consumed at the same rate as it is produced. $FeCO_3$ is a stable end product.

Eq. 4

The above mentioned reaction is assumed to be valid, when there is ample supply of carbon dioxide. However, the possibility of the reaction to take place in the alkaline environment dominated by concrete water has not yet been evaluated.

It is assumed that the package drum will corrode according to Eq. (3). In addition, the inner surface is subjected to microbially induced corrosion; carbon dioxide generated will be bound to iron carbonate according to equation (4). The latter corrosion process is thus dependent from the microbial process through Eq. (1). The 200 l drum is made of 16

kg steel plate, thickness 1 mm and a total surface area of about 4.2 m2. The amount of steel in one drum gives 290 moles of iron yielding in corrosion 390 moles of hydrogen, which is 8.7 m3 NTP. The estimated corrosion rates are between 0.1 mm/a and 10 mm/a (10). Thus the total corrosion process would take from 100 to 10,000 years. The rate of anaerobic corrosion is assumed to be 0.1 mm per year.

Figure 2 presents the estimated gas generation as a function of time. The calculated results are presented for the amount of 16 waste drums each containing 47 kg of biodegradable material. According to this estimate

The biodegradation is complete in hundred years,

The corrosion process takes longer than biodegradation in the selected calculation example.

When a closer look is taken at the first ten years, the above conservatively estimated reaction rate leads to a gas generation rate of 15 l/day. The share of corrosion is 0.23 l/day. In this estimate it is assumed that the steel surface of the drums is attacked uniformly by corrosion. The surface is, however, protected by painting, which may give considerable protection for the drums during the initial period of the experiment.

Fig. 2

PREPLAN OF THE PILOT SCALE EXPERIMENT

The full utilization of one of the large bored holes in the Research Tunnel makes it possible to study the degradation process in the scale of one disposal unit: the concrete box, which contains 16 drums of waste. The following dimensions are derived from the initial conditions:.

The inner diameter of the protective vessel can be up to 1480 mm, four drums fit in each layer.

The height of the vessel can be 5 - 6 m (volume 9 - 10 m3) allowing four layers of drums vertically.

The experiment will consist of 16 drums, each having a volume of 200 l, weighing 16 kg (steel), and containing 100 kg of compacted waste. The drums will be perforated with small holes to allow water intrusion as well as escape of gases.

Concrete blocks are suggested for the inner structure of the vessel to support the stacked waste drums. The equilibration of concrete with water will create alkaline water chemistry. Approximately 2,000 kg of concrete structures are needed to construct the supporting system for the drums, the amount can be increased to 4,000 kg (corresponding to the whole amount of concrete in one disposal box).

Figure 3 shows the vertical cross section of the experimental set-up.

Fig. 3

It is planned that the pilot scale experiment will be started by filling the vessel with water. After that the conditions are let to develop naturally.

EXPERIMENTAL PARAMETERS TO BE MEASURED

The gaseous reaction products from the microbial degradation and corrosion are the most important parameters to be measured. This involves analyzing of dissolved gases in the water phase and released gases. If the rate of gas generation is large, the gases will be conducted out of the experimental system through a gas meter; but if only small amounts of gases are generated, they are collected, sampled and released in known quantities.

It is suggested that dissolved oxygen and hydrogen are measured directly from the water, the separating gas is sampled and its components analyzed regularly: carbon dioxide, methane, hydrogen, oxygen.

The properties of the water inside the vessel will develop in the course of time. Some parameters can be measured on-line and other parameters must be measured from water samples. It is possible to install several sampling lines to the system for sampling water inside each individual drum. It must be emphasized, that collecting and handling (analyzing) of the water samples should be performed in an atmosphere (e.g. atmospheric box) comparable to that in the studied system. The alkaline water will be very sensitive to CO2 and thus any CO2 contamination should be avoided. Follow-up of water chemistry during the experiment will give additional information about the processes involved in the control of pH and redox. Particulates and colloidal entities can be studied from obtained water samples, giving insight to radionuclide migration/retardation phenomena. Particulates can be collected on filters and the colloidal fraction can be characterized from a concentrated sample. Changes in TOC/DOC will give information on the development of organics in the water phase.

Sampling of the actual waste materials gives direct information of the degree of disintegration of the various waste components and also enables sampling of the microbes attached on the surfaces. For this purpose it is suggested that specially prepared samples representing the different waste materials and packaging materials are assembled inside the waste drums. These samples can then be withdrawn from the system at suitable intervals and studied.

Because there are no full-time workers in the Research Tunnel for taking samples, it is necessary to measure some indicative parameters on-line. It would be practical to have the sensors located in an instrument cabin nearby outside the experimental hole. Water from the experiment will be circulated constantly between the vessel and the sensors.

The prerequisite to an on-line measurement is that the sensors can operate and keep their calibration at least for the time between the sampling and inspection visits. The suggested parameters to be measured on-line are: pH, Eh and conductivity. Also dissolved gases O2 and H2 can be measured reliably by specific electrodes.

All measured signals are connected to a data acquisition system for registration and control. The measured data is saved in data files for later presentation and evaluation. The data acquisition software is capable of showing the measured data graphically on the screen in real time or to draw history graphs of quantities measured earlier. The software has a graphical user interface and it is possible to show the state of the test process continuously in a graphical process diagram. The system can be operated with a remote computer using a standard phone line.

THE PRELIMINARY TIME TABLE

After filling and closure of the experimental set-up, various environmental parameters will change simulating the closure and sealing of the real repository. After filling with water the conditions will develop from neutral to alkaline and from aerobic to anaerobic. The experiment will reach an advanced stage, when the microbial population has adapted to the conditions and the painting of the drums is not preventing the corrosion of steel. The transition phase is important as such; the growth of microbes may be so slow that it leads to an impracticable long follow-up time.

The experimental program can be divided roughly to three phases: Preparatory phase including preplanning and preliminary experiments in the laboratory, set-up phase including detailed planning and construction and research phase including start and follow-up of the experiment, Table II.

The preparatory phase will consists of background studies for the pilot plant experiment. Laboratory experiments and modeling lead to a more detailed forecast of the rate of the microbial disintegration and corrosion reactions in the repository conditions.

The set-up phase will consist of a detailed design, manufacturing and installation of the experimental equipment. The detailed characterization of waste to be brought into the experiment is included here.

The research phase will begin with filling of the system with water. The follow-up will last for several years.

Table II

ACKNOWLEDGEMENT

This work belongs to the nuclear waste research program of Teollisuuden Voima Oy and it will be a part of the European Commission Nuclear Fission Safety Program (1994 - 1998).

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OVERVIEW AND DYNAMICS OF ORGANIC CARBON IN A LOW-LEVEL RADIOACTIVE WASTE DEGRADATION EXPERIMENT*

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ABSTRACT

Low-Level Radioactive Wastes (LLRW) contain substantial quantities of organic materials that will eventually decompose by microbial action after disposal, producing gases and dissolved organic compounds. An experiment was undertaken to study microbial degradation processes with LLRW, with three major purposes: i) to characterize the bulk properties of the Dissolved Organic Matter (DOM) produced by decomposition; ii) to determine its influence on the aqueous speciation of selected radionuclides, and iii) to determine the main thermodynamic parameters that can define the near-field chemical environment.

The bulk properties of the leachates and the DOM generated in the experiment were similar to those of landfill leachates. The DOM content was dominated by volatile fatty acids (VFA), likely produced by the microbial degradation of cellulose and other structural polysaccharides. The DOM has a limited complexing capacity for Cd and Co, but the elevated levels of DOM in the leachates may significantly affect Co speciation. Chromatographic separations indicated that at least 20 discrete organic compounds were present, and the total organic carbon was primarily hydrophilic (70-95%). This technique also showed that Co can form a mixture of labile and stable complexes with compounds contained in the DOM.

Calculations based on a mass balance approach suggested that microbially-mediated organic decomposition will persist for periods of decades to centuries, and this process will likely dominate the redox potential in the near-field. This, in some cases at least, simplifies the specification of near-field chemical conditions.

INTRODUCTION

Low-level radioactive wastes contain a large fraction of organic material (office wastes, paper, clothing, etc.), upon which microbial decomposition will produce a mixture of organic compounds. In addition to this, man-made complexing agents and natural organics may also contribute to radionuclide subsurface mobility (1-3).

Upon degradation, the easily available substrates (such as polysaccharides) will be utilized first, producing low molecular weight organics such as alcohols and fatty acids. Other compounds coming from the partial degradation of the source material will also be produced (e.g., gluconic acid, cellobiose, etc.). This Dissolved Organic Matter (DOM) itself is subject to further degradation to CO_2, CH_4, humification, and formation of colloidal material. The resulting DOM in the leachates will contain a mixture of various degradation compounds of different metal-binding affinities, and different degradation rates. The remaining carbon source material will eventually be more recalcitrant to decomposition, and this process can be very long (years to centuries), especially if the environment is reducing. All of these processes can affect the subsurface mobility of radionuclides by complexation and they can exert a control over the redox potential for several years, hence buffering against redox changes in the near-field of a waste disposal site.

An overview of a low-level waste degradation experiment is presented in this work. The aim is to present two major aspects that microbial degradation has on degrading wastes: (i) the production of dissolved organic compounds, which can affect speciation of nuclides and possibly their subsurface mobility; and (ii) the control that microbial degradation could have on the redox potential in the near-field, with an estimate of how long this control could last. This information is applicable to near-surface waste disposal sites, landfills, etc., and an example of application is presented for a generic near-surface repository.

EXPERIMENTAL

Description of the Experiment

Low-level wastes were compacted in bales measuring approximately 0.6 0.6 1.2 m (see (4) for a detailed description). Eight bales were placed into separate airtight carbon steel boxes, and then connected to an overflow reservoir for leachate collection in a closed-loop recirculation system (Fig. 1). The bales were monitored for gas generation for the first year after they were prepared, without addition of water ("as-received", (5)). De-ionized water was then added to each box and recirculated daily over each bale to promote degradation. Four of these bales were under "wet" mode by letting excess water seep out by gravity into the overflow container, and the other four were "flooded" by raising the outlet tube to keep the water level to about half of the height of the bale. This simulates bale decomposition under wet unsaturated conditions in a soil, and wet flooded conditions below or at the water table.

Fig. 1

Sampling and Analysis

Aqueous samples were analyzed for major cations (Na^+, Mg^{2+}, Ca^{2+}, K^+), selected metals (Fe, Zn, Mn, Cu, Al), major anions (Cl^-, SO_4^{2-}, NO_3^-, I^-), total organic carbon (TOC) and volatile fatty acid (VFA). The DOM of selected samples were also analyzed by high performance liquid chromatography (HPLC; see (6)) and for complexing capacity using an ion exchange technique (IET; see (7)). Gaseous samples were also taken to determine the major gases in the box atmosphere using gas chromatography (GC). The TOC content was analyzed using a Dohrman DC-80 Carbon analyzer, and the VFA by GC (8). The total amount of exchangeable protons was done by titration with a Tanager 8901 autotitrator.

The samples for cation and metal analysis were acidified immediately upon collection. For the anions, a 15-mL untreated leachate aliquot was passed through a bed of cation exchange resin to remove cations that could precipitate. The samples used for titrations, HPLC, IET were mixed with a cation exchange resin added in batch mode (in Na form), to remove excess metals that could precipitate. Equilibration was done for one week, after which the solution was decanted and the samples were allowed to stabilize further under room conditions in the dark without resin for 2-3 weeks (IET), 12-16 months (HPLC) and 18 months (titration).

RESULTS AND DISCUSSION

Dissolved Organic Matter Characteristics

The TOC concentrations in the bale leachates ranged from ~170 to ~7000 mg C/L (Table I). The TOC levels in the four "wet" bales (1,4-6) were generally higher than in the "flooded" bales (7-10). The amounts of TOC did not vary systematically with time for the "wet" bales, however they generally decreased for the "flooded" bales. These amounts are within the range for landfills or commercial nuclear Low-Level sites (9-11).

Table I

The VFA made up most of the TOC (Fig. 2; also see (12)). The relative VFA distribution remained nearly constant for the ~500 d period with the "wet" bales, which is evidence that the microbial degradation process controls the relative distribution of the VFA, hence likely the decomposition mechanism of the source material.

Fig. 2

Fermentation is taking place in the bale boxes, where bacteria and fungi use the cellulose matrix as a carbon source to produce alcohols, VFA and CO_2 + CH_4 gases. Both aerobic and anaerobic processes may take place simultaneously, however anaerobic processes may dominate because of oxygen depletion. Anaerobic microbial degradation may take place in several but distinct steps (13): i) fermentative bacteria living under anaerobic or facultative conditions, producing acetic acid and other VFA; ii) acetogenesis bacteria become involved in a bioenergetic symbiosis with methane-producing bacteria or other hydrogen assimilators; and iii) methanogenic bacteria predominate, producing CO_2 and CH_4 in nearly equal amounts. Rees, (1980) (11) has shown that the gas production rate was correlated with the moisture content of the wastes in landfills, and this has also been observed in this experiment. Methane and CO_2 production dominated in bales 7-10, and VFA production was dominant in the other bales, with acetic acid being the most abundant VFA species. Work on the "wet" bales (1, 4-6) is emphasized in the remainder of this work because this situation is more applicable to a near-surface waste disposal site located above the water table.

Knowing that most of the TOC is made up of VFA, the chemical composition and behavior of the balance of the TOC are unknown. Microtitration of the leachates gave curves closely related to the behavior of the VFA [unpublished], with other material of intermediate acidity in the pKa range of ~8-9 (pKa = - log Ka, where Ka is the acid dissociation constant), which is characteristic of phenolic groups. The charge density of the TOC, defined as the total amount of exchangeable protons per unit of carbon (mEq/mg C), was also typical of the values of ~15 mEq/mg C found in landfill TOC (Table II; (14)).

Table II

HPLC separation of the leachates indicated that the DOM could be separated into two general regions: hydrophobic and hydrophilic (Fig. 3a; (6, 15)). In our previous study, most of the TOC (~70-93%) was in the hydrophilic region. Several sharp peaks were found in each one of these two regions, indicative perhaps of single compounds or a small amount of compounds of similar properties. The hydrophilic peaks, eluting early (Fig. 3a), could be further separated into ~10 discrete peaks using ion pair HPLC (Fig. 3b).

Fig. 3

The three techniques and analyses used (VFA analysis, titration, HPLC) indicated that the DOM from this waste degradation experiment has properties similar to that in municipal landfill leachates. Volatile fatty acids were the most abundant class of compounds identified, making up most of the TOC in leachates in the "wet" bales. The "flooded" bales were in the methanogenesis mode, and its VFA content was smaller and distinct from the "wet" bales.

Complexation with Radionuclides

Two approaches have been used to investigate the complexation of organic material with radionuclides: the first one used the overall complexing

capacity of the leachates with the radionuclides 60Co and 109Cd, using an ion exchange technique (IET) (7). The second approach was to equilibrate the DOM with 60Co radiolabel, followed by HPLC separation coupled with a fraction collector to identify more specifically the types of compounds that are responsible for the complexation (6). The IET showed that ~4000 mg/L of TOC could affect the speciation and complex approximately 50 mg/L of Co. Complexation with Cd was much smaller, 0.54 mg/L. The HPLC technique indicated that a mixture of labile and non-labile complexes could form. The non-labile complexes were mostly associated with hydrophobic compounds, and they made up ~5% of the total radiolabel spiked in solution. Mixtures of labile and non-labile complexes of cobalt were also observed in separate experiments (1,2,17).

The impact of DOM complexation on the speciation of Cd and especially Co is somewhat uncertain. The IET is quantitative, yet non-specific, and it suggests that a substantial amount of Co could be complexed with leachate DOM. The HPLC approach is non-quantitative, but it suggests that stable complexes could be formed with hydrophobic DOM compounds.

Near-field Chemical Environment

The second aspect presented in this work is the influence and rates of organic carbon degradation on the redox potential. Carbon degradation is a dynamic system that can be related with a series of chemical redox equations. For instance, using a mass balance approach for Bale 6 between day 32 and day 391, the major changes in redox-sensitive components can be related to a series of half-cells and combined into full chemical reactions. These four reactions balance the assimilation/production of CO2, Fe(II), acetic acid (taken for VFA), N2, and CH4 in bale 6:

Eq. 1

Eq. 2

Eq. 3

Eq. 4

These reactions do not necessarily describe processes, but rather they explain the thermodynamic end result, which is path-independent. In bale box 6, a total of 0.44 mole of dissolved iron, Fe(II), was present in the leachates at day #32, compared with 2.98 moles at day #391, for a difference of 2.54 mole. This means that 0.64 reaction (1) has produced this amount of Fe(II) in this bale during this period (0.64 4 = 2.54). All four reactions above were balanced iteratively with a spreadsheet, using the initial and final amounts found in this bale. Similarly, the total contribution of reactions (2), (3) and (4) were 0.69, 3.32 and 1.68, respectively. The Gibbs free energy of each reaction (DGr) is related to the redox potential using a modified Nernst equation [18] with the measured quantities of oxidized and reduced species in the bale:

Eq. 5

The redox potential in Bale box 6 was calculated to be ~ -2.9 to -4.7 (EH ~ -0.17 to -0.28 V) at pH 5.74. It is recognized that this approach necessitates the assumption of equilibrium, which is not achieved in these boxes, as the system is dynamic, and there are undoubtedly inhomogeneities within this bale. Nevertheless, this approach provides an independent estimate of the redox potential, devoid of measurement artifacts present if an electrode is used. The mass balance approach also suggests that reactions (2), (3) and (4), involving the electron transfer from carbon atoms, account for ~94% of all electrons exchanged, whereas ~6% comes from Fe to produce Fe(II) (steel corrosion). Microbial processes, therefore, likely control the organic carbon degradation rates

and speciation, which in turn would control the redox potential inside these boxes.

Estimates of Carbon Degradation Rates

Organic carbon decomposition generally follows a first-order decay curve (19). In short-term experiments, the most easily degradable form of carbon likely controls the gas generation and the decomposition rates of the bulk of the organic carbon present. However, in long-term experiments or predictions, the model can be more complex to reflect the different types of organic carbon in soils, landfills, etc. In the case of a two-compartment model, the relationship is:

Eq. 6

The two compartments (#1 and 2) indicate organic carbon of different biodegradability. The term C0,(1) and the decay constants are usually difficult to evaluate in waste material due to the unknown composition of the contents and specific decay rates, even in these bales.

Alternatively, approximately 70% (by weight) of the wastes contains organic material (paper, plastic, textiles, etc.), and the balance is not degradable (ceramics, metals, etc.). From this 70%, approximately two thirds is moderately slowly biodegradable (MSB), and the other third is slowly biodegradable (SB) to recalcitrant material (RM). Using these approximations, and also assuming that only the fastest degrading material (i.e., MSB) is degraded over the 1-year period of the degradation experiment, equation (6) simplifies to a single decay term (the right-hand part is equal to zero). The total content of MSB carbon in bales 1, 4, 5 and 6 would be ~300 kg, of which our mass balance has indicated that a total of ~123 moles (~1.5 kg) have degraded to CO2, CH4 and DOM between day #32 and 391 for the four bales. This corresponds to a degradation half-life of 55.9 a (or k(1) = 1.24 10-2 a-1). This value is very close to the half-life of 49.5 a obtained for the "physically stabilized organic carbon" in the model of Jenkinson and Rayner (19). Under these conditions, the bale box chemical environment is likely to be dominated by the organic carbon degradation rate at least for a few decades.

Example Application to a Generic Near-Surface Repository

In this generic example (Fig. 4), a near-surface vault is located in sandy material, above the water table. The vault features a concrete roof with a service life of 500 a, designed to isolate the wastes from infiltrating water. Most of the wastes will consist of compacted bales, and our calculations assume that two thirds (65%) of the organic C in the vault is MSB, 30% is SB and 5% is RM. It is also assumed that the RM does not significantly degrade over the 10000 a period of the calculation, and the SB has a half-life of 2000 a (compared to 1980 a for recalcitrant carbon in Jenkinson and Rayner (19)).

Fig. 4

The degradation rate obtained for MSB in the bales has to be corrected for the expected dry vault environment under an intact roof. In the absence of a specific rate indicator, the gas generation rates in the "wet" bales, ranging from 0.1 to 1.3 L kg-1a-1, has been normalized to the reference gas generation rate of 0.021 L kg-1a-1, obtained previously for the dry bales (5). This is believed to represent degradation in a dry vault. This correction corresponds to a half-life of 1400 a for the MSB material, hence an initial rate constant to be used in Eq. (6). At the onset of roof failure (assumed at 500 a), water will infiltrate and accumulate in the moist bales, which is equivalent to the "wet" bales

described above, and the much faster 55.9 a half-life applies thereafter. In Fig. 4, Eq. (6) was applied in a two-step approach to describe the behavior of organic carbon before and after roof failure, using the rate constant of MSB material under dry conditions for the first 500 a, and MSB material under wet conditions after 500 a. A more simple one-step approach, neglecting the effect of roof failure, was fitted visually for comparison, and gives an overall half-life of 600 a for all the organic carbon present in the vault. This example of calculation suggests that most of the organic material will be degraded after 2000-3000 a.

In order to estimate an overall redox potential in the vault and its evolution with time, one would have to balance the reducing capacity of the material in the vault with the sources of oxidizing materials. The degradation of organic material, given above, and metal corrosion would provide the reducing capacity. Metal corrosion is likely to follow a constant rate with time, often expressed as a corrosion rate (in mm/a or equivalent) in models (20). This rate depends upon the surface available for corrosion, which is often difficult to obtain. Both these rates have to be compared to the rates of oxygen ingress in infiltrating water (after roof failure) and other mechanisms in order to estimate the overall redox potential in the vault, hence the near-field chemical environment. This type of calculation needs to be addressed separately, as, for example, low humidity in the near-field also features a high gas-filled porosity and low degradation rates, which would favor oxygen ingress; a wet environment favors higher degradation and higher oxygen consumption rates than for a dry environment, plus a smaller gas-filled porosity available for oxygen migration. In addition, after the MSB carbon is degraded, the organic carbon degradation rates would gradually decrease, potentially allowing more oxygen ingress. Hence the variations of redox potential in the near-field are expected to vary over several centuries.

CONCLUSION

A low-level waste degradation experiment was done at CRL to investigate the degradation processes. The leachate DOM was dominated by VFA and this process is likely controlled by the microbial flora present in the wastes. This resulting DOM forms labile and non-labile complexes with Co, potentially affecting its subsurface mobility. A mass balance suggested that microbial degradation is holding a reducing environment in these boxes, and this could last at least for a few decades. Some aspects of this experiment are applicable to wastes degrading within a dry vault, however, there are still uncertainties as to how long or if a reducing environment could last in the near-field.

ACKNOWLEDGEMENTS

The author wishes to thank J. Torok, who has started this waste degradation experiment, and turned over the project to me. Also, G. Manni, M.K. Haas, S. Elchuk, Z.H. Walker, Chalk River Laboratories, P. Vilks of Whiteshell Laboratories, Pinawa, and C. Fortin, INRS-Eau, Ste-Foy, Qubec, have been outstanding collaborators. E.L. Cooper, Chalk River, has also provided timely help with discussions. S. Stroes-Gascoyne, Whiteshell, and K. King, Chalk River, have helped reviewing this document.

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45-6

PRELIMINARY SCREENING RESULTS FOR IN SITU BIOREMEDIATION AT AN UMTRA SITE

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ABSTRACT

In 1978, Congress passed the Uranium Mill Tailings Radiation Control Act (UMTRCA). The U.S. Department of Energy (DOE) is developing a program to meet the U.S. Environmental Protection Agency (EPA) ground water standards (60FR2854) at 24 uranium mill tailings sites nationwide. The inactive uranium processing site at Tuba City, Arizona, has been designated as one of the 24 sites.

As part of the U.S.-DOE's Uranium Mill Tailings Remedial Action Project (UMTRA), the DOE is currently studying alternative ground water remediation technologies to determine the most appropriate and cost effective technology. At the DOE/UMTRA Tuba City site, the University of New Mexico's (UNM) Center for Radioactive Waste Management (CeRaM) is developing an in situ bioremediation process as one possible alternative methodology. Bioremediation uses bacteria to clean the ground water of the contaminates. These contaminates include uranium, selenium, strontium, molybdenum, cadmium, nitrate, and sulfate. Through bacteria catalyzed reactions the contaminates are either converted to non toxic substances, such as nitrogen gas in the case of nitrates, or in the case of the heavy metals, they are immobilized as insoluble precipitates (ores). The paper describes the development of the in situ bioremediation process for the Tuba City site. This includes site characterization for indigenous bacteria, laboratory batch and column studies that simulate the field process, and our plans for a field observational test. The current laboratory and field site characterization tests have shown that the necessary bacteria are present and that they can catalyze the remediation reactions under field conditions.

INTRODUCTION

As part of the U.S.-DOE's Uranium Mill Tailings Remedial Action Project (UMTRA), DOE is developing a program to meet the U.S. Environmental Protection Agency (EPA) ground water standards at 24 uranium mill tailings sites nationwide. This effort is a direct result of the 1978 Uranium Mill Tailings Radiation Control Act (UMTRCA) as amended. For each UMTRA site, the DOE is currently studying alternative ground water remediation technologies to determine the most appropriate and cost effective technology (1,2). The inactive uranium processing site at Tuba City, Arizona, is one of the 24 UMTRA sites. The surface contamination was remediated in a disposal cell that was completed in 1990 and now the ground water remediation is being investigated. The University of New Mexico's (UNM) Center for Radioactive Waste Management (CeRaM) is

currently developing an innovative in situ anaerobic bioremediation process for cleaning the ground water at the UMTRA Tuba City site (3,4). Research and technology development work are conducted at the University of New Mexico in collaboration with these institutions: Montana State University, Bozeman, MT; Navajo Community College, Shiprock, NM.

REMEDIATION CONCEPT

Description of Technique

The ground water remediation technique involves in situ anaerobic biological reduction of the contaminates uranium, selenium, strontium, molybdenum, cadmium, nitrate, and sulfate. We plan to selectively grow bacteria that are currently present in very low concentration in the ground water on site. In turn these anaerobic bacteria will catalyze a series of chemical reactions transforming the water soluble contaminates into either a gas in the case of nitrates (5-9) or insoluble precipitates in the case of the heavy metals (10-19).

The in situ biological treatment should be implemented using conventional drilling and well water technology. The advantages of in situ bioremediation are that the contaminate plume is remediated and the aquifer is restored in a natural process with minimum environmental impact.

Site remediation requires that the upper most water bearing formation be restored to meet regulatory ground water requirements. This restoration process should consider cost, environmental impact and the total remediation time period. At the Tuba City, Arizona, site the plume of contaminated ground water is approximately 1500 ft down-gradient of the site and is moving on average 30 ft/year (18) towards the Moenkopi Wash. The ground water is located at a minimum depth of 60 ft below the ground surface (18) and the background ground water contains calcium, sodium carbonate and has a slightly basic pH, an oxidizing Eh and TDS 450mg/L. The plume waters contain the various chemicals (e.g., sulfates, nitrates, etc.) used in the uranium processing operations and have lower pH (6 pH 7) and lower Eh and 750LTDS7200mg/L.

Because uranium and other heavy metals appear to be retarded in their transport, as illustrated in Fig. 1, we do not believe that conventional ground water treatment methods, such as pump and treat, are likely to succeed. Also, the water table is too deep and the plume is moving too slowly to apply the permeable barrier technology. Well #906 is in the most contaminated region of the plume located at the South edge of the tailings pile, while well #908 is about 50 meters down gradient in the plume and is the second most contaminated well. Note, that well #906 has a uranium concentration 40 times the EPA MCL, while in well #908 the uranium concentration is only slightly above the EPA MCL. However, in both wells the nitrate and sulfate concentrations remain about the same, i.e., 30 and 15 times the EPA MCL standard. This decrease in uranium and molybdenum concentrations indicates strong retardation. If this initial trend persists,after more sampling wells are completed and tested, then the remediation of the heavy metals may be limited to a much smaller area, as compared to the nitrate and sulfate regions. This suggests the possible application of different in situ technologies over the various regions of the plume.

Fig. 1

Therefore, we are proposing for the Tuba City site an innovative in situ biological remediation approach. The strategy is for bacteria to chemically convert the mobile toxic species into either a gas or into

immobile precipitates. The ground water at the Tuba City, Arizona, site is contaminated with uranium, selenium, strontium, molybdenum, cadmium, nitrate and sulfate. Anaerobic bacteria from ground water samples at the UMTRA site have been shown to transform these chemical species into passive forms. Various reduction reactions are catalyzed by the bacteria as illustrated below. Sulfide is produced from sulfate reduction and then reacts to form insoluble precipitates with various heavy metals. The proposed biochemical remediation reactions are shown schematically in Fig. 2. The most contaminated region of the plume has a very low dissolved oxygen concentration of about 0.2 ppm and we anticipate, with the addition of appropriate amounts of organic nutrients, that the anaerobic bacteria will increase in number and promote the reactions illustrated in Fig. 2.

Fig. 2

As shown, these reactions convert the toxic species into materials that render the ground water harmless either through precipitation (formation of insoluble solids) or denitrification (formation of nitrogen gas). In this concept the amount of present nutrients is the limiting and controlling factor of reaction progress. As an example, only a portion of the sulfate will be reduced as needed to precipitate the heavy metals. By limiting nutrients and the amount of sulfate reduction we do not anticipate methylation. Excessive reduction of sulfate may lead to odor of the water and reactions of HS- with components of the host rocks that have not yet been evaluated. On the other hand present sulfate concentrations are too high (about 3-4 g/L) to be reduced to acceptable levels (about 250 mg/L) by bacteria without producing excess HS-. Yet this biotechnology is the least harsh methodology at hand as it enhances a natural process and accelerates the restoration of the ground water. After remediation of the ground water, bacteria are no longer needed. Solids such as UO2, CdS, etc. are stable under the given environmental conditions (Eh, pH, temperature and water composition) at a depth of >60 ft. in the sandstone formation. Though the solid compounds of the hazardous elements are not absolutely insoluble, their redissolution is limited by their extremely low solubilities, if not by slow dissolution kinetics. Solution concentrations are expected to be much lower than the regulatory limits established by 60FR2854 that parallel the drinking water standards.

The procedure is to inject organic nutrients, such as acetate, into the contaminated plume in order to support the growth of the indigenous bacteria, which in turn will perform the necessary anaerobic remediation reactions. The field implementation strategy is described in the next section. Studies to date have shown that the appropriate bacteria species are present, however, in very low concentrations because of limited organic nutrients in the water. The sandstone's permeability is high enough for in situ biotreatment, i.e., greater than 10-4 cm/sec and the dissolved oxygen concentration in the plume is low, 0.2 ppm.

Laboratory Studies

With and without the presence of sandstone core material, several batch experiments were conducted to assess the impact of the sandstone on the activity of bacteria. The number of viable bacteria in the experimental bottles was determined by standard dilution procedures with growth on Plate Count Agar. We found that the core matrix has no apparent deleterious effect on bacterial viability. Experiments will be designed to assess the interactions between sandstone surfaces and bacterial

cells.Laboratory studies using bacteria from the site have shown the ability of these bacteria to rapidly reduce nitrates. Using two isolates of bacteria, strain 906F and 906R, the rate of nitrate utilization was examined with three different carbon sources. An inorganic medium containing 200 ppm nitrate was employed with 40 mM sodium acetate, sodium lactate or sodium pyruvate added to support growth. The anaerobic cultures were examined over a period of 15 days with incubation at 16C, the ground water temperature in the plume. Nitrate concentration was determined using ion chromatography (Dionex Corporation, Sunnyvale, CA). Fig. 3 shows that nitrate reaction rates depend on the type of nutrient and that significant amounts of nitrate are reduced in a matter of days.

Fig. 3

We have performed similar experiments showing the equally rapid reduction and precipitation of uranium at temperature of 16 C using the Well #906 strain of bacteria. The transformation of the uranyl ion to uraninite (UO_2) has been documented in the literature to proceed by bacteria of special physiological types: Fe(III)-reducing, spore-forming Clostridium sp., and sulfate-reducing bacteria. Our interest in bacterial transformation of uranium has focused on the sulfate-reducing bacteria because these bacteria are widely dispersed in the ground water. Published laboratory studies indicate that high levels of U(VI) are reduced in a short time frame of several minutes to hours by enzymatic processes associated with cells of sulfate reducers. However, the potential reduction of U(VI) to U(IV) by sulfide has not been fully explored. Since hydrogen sulfide would diffuse widely in the aquifer system, it would be important to determine the effect of this reducing agent on U(VI) in the ground water. Our experiments involved the use of simulated ground water with the composition listed in Table I to match the chemical characteristics of water from well 906 at Tuba City. Various additions were made to anaerobic 150 mL bottles containing 100 mL of simulated plume water. Four different concentrations of sulfide and two concentrations of uranium were studied while incubation was conducted at two different temperatures under nitrogen atmosphere. The experiment was conducted in duplicate and values presented are averages.

Table I

The results of the study with water containing 1.2 mM uranyl acetate are presented in Figs. 4a through 4e. With no sulfide added to the anaerobic bottle, U(VI) remained in solution at 1.2 mM and no figure is shown to represent this. In the presence of 0.1 mM sodium sulfide, the concentration of U(VI) steadily declined at both temperatures; however, at 36C the transformation of U(VI) to U(IV) was more rapid than at 16C, see Fig. 4a. The results of U(VI) reaction by 0.2 mM sulfide are given in Fig. 4b, while the results with 0.5 mM, 0.7 mM and 1.0 mM sulfide are given in Fig. 4c, Fig. 4d and Fig. 4e, respectively. This experiment shows that uranium can be reduced by the production of sulfide. Hence, in our process the presence of sulfate and SRBs should lead to the precipitation of uranium dioxide.

Fig. 4

CONCLUSIONS

The plume of contaminated ground water at the Tuba City mill tailings site in Arizona contains the necessary indigenous bacteria to develop an in situ bioremediation process. Bioremediation is an attractive and potentially the preferred technique to decontaminate the ground water at Tuba City. Strong retardation, i.e., slow migration of uranium and

molybdenum was detected in the plume relative to sulfate and nitrate, indicating sorption of heavy elements on surfaces of the host rock. If desorption is slow, a pump and treat process may not be efficient, whereas in situ bioremediation takes place in the pores of the host rock (Navajo sandstone) and reaches internal surfaces. Experiments in the laboratory have shown that NO3- and U(VI) are reduced to N2 and insoluble U(IV), respectively, using indigenous bacteria from the site and organic nutrients, e.g., acetate. Sulfate reducers were also detected in the ground water and will be used to generate sulfide. Sulfide was shown to reduce UO22+ within days, supporting the respective bacteria catalyzed reaction. Other heavy metals, e.g., Cd, Mo, will be precipitated and immobilized as sulfides.

ACKNOWLEDGMENT

This work was supported by the U.S. Department of Energy through the New Mexico Waste-management Education and Research Consortium (WERC), Contract No. PR 04-94AL638055.507.

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Session 46 -- STRATEGIC PATH FORWARD FOR USDOE OWNED SPENT NUCLEAR FUEL
Co-chairs: Ed J. Bentz, EJB&A;
G. Frank Cole, USDOE
46-1
OVERVIEW OF THE U.S. DOE-OWNED SPENT NUCLEAR FUEL NATIONAL PROGRAM

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ABSTRACT
The U.S. Department of Energy (DOE) faces a complex challenge in managing the inventory of spent nuclear fuel (SNF) generated by past production and ongoing test reactor and research reactor operations. Because ultimate disposition of DOE-owned SNF will require geologic disposal, and because DOE no longer processes SNF to recover strategic materials, SNF will need to be stored for decades. The Office of Spent Fuel Management in the DOE Office of Environmental Management (EM) is responsible for overcoming that challenge. This paper reports on SNF management program's ongoing efforts to manage the inventory of SNF. The paper discusses the recent accomplishments of the SNF management program, the program's progress on a number of outstanding issues, and plans for the near-term. Additionally, the paper serves as an introduction for the other papers in this session.
The paper shows that the SNF management program has made significant progress in a number of areas affecting the safe interim storage of DOE-owned SNF. Examples include decisions on removal of the deteriorated spent fuel from the Hanford K-Basins and the treatment and/or transfer of SNF from facilities at Savannah River and the Idaho National Engineering Laboratory. However, numerous technical, managerial, and policy issues remain as obstacles on the path forward for DOE-owned SNF. An example is

resolution of questions regarding acceptable forms and methods of storing, packaging, transporting, and disposing of the spent fuel.

THE SPENT NUCLEAR FUEL LEGACY

The U.S. Department of Energy (DOE) faces a complex challenge in managing the inventory of spent nuclear fuel (SNF) generated by past production and ongoing test reactor and research reactor operations. Because ultimate disposition of DOE-owned SNF will require geologic disposal, and because DOE no longer processes SNF to recover strategic materials, SNF will need to be stored for decades. The Office of Spent Fuel Management in the DOE Office of Environmental Management (EM) is responsible for overcoming that challenge. This paper reports on SNF management program's ongoing efforts to manage the inventory of SNF.The decision to phase out processing of DOE-owned SNF has resulted in a growing inventory. As of January 1996, DOE held 2646 metric tons of SNF in its storage facilities. The challenge of DOE SNF management is compounded by the significant differences between commercial SNF and DOE-owned SNF in terms of variety, condition, characterization, enrichment, cladding, and other characteristics. Additionally, certain vulnerabilities are associated with this SNF, including corrosion and structural degradation.

Fig. 1

DOE SNF MANAGEMENT

The general goals and practices of DOE-owned SNF management are to achieve and maintain safe storage of SNF for an interim period to ensure the protection of the environment, workers, and the public. Attaining this requires some treatment and packaging of the SNF to meet both interim storage and permanent disposal criteria.

Key SNF management guidelines are: to regionalize interim storage primarily at three sites (Hanford, Savannah River and Idaho National Engineering Laboratory [INEL]) by fuel type, and to ultimately dispose of SNF in a geologic repository.

The regionalization by fuel type approach will occur as follows:

Hanford production reactor fuel and fuel not requiring treatment will remain at Hanford; sodium-bonded Fast Flux Test Facility fuel will be shipped to INEL for treatment.

Naval fuel will be shipped to INEL for examination and interim storage.

Non-aluminum clad fuels will be consolidated at INEL, excluding Fort St. Vrain fuel, which will remain in Colorado.

Aluminum-clad fuels will be consolidated at the Savannah River Site.

More information on this approach can be found in the paper given in this session by H. Eckert.

DOE SNF MANAGEMENT AT HANFORD

Fig. 2

DOE is undertaking at Hanford a number of significant SNF management activities, and is developing or implementing certain plans for SNF management. Conditions at the K-West and K-East irradiated fuel storage basins present challenges. The K-East Basin contains irradiated N-Reactor SNF stored in open canisters, and some of the fuel has corroded. Further, the K-East Basin contains debris and approximately 50 cubic meters of sludge. Both basins have a history of leaking.

To improve near-term conditions in the K-Basins, DOE has installed seismic barriers in both basins. An Environmental Impact Statement (EIS) for the K-Basins will be completed in March 1996. Plans for Hanford SNF activities are to begin removal of the SNF and sludge from the basins in December 1998 and place it in a Staging and Storage Facility, which is to

be located in the 200 Area and completed by May 1998, at the Hanford Site by December 2000. The K-Basins path forward would complete placing the fuel and sludge in interim dry storage by the year 2006.
More information on SNF management at Hanford can be found in the papers by J. Fulton and E. Sellers.
DOE SNF MANAGEMENT AT SAVANNAH RIVER
Fig. 3
DOE is conducting numerous activities and making plans for SNF management at the Savannah River Site. An important issue at the site is the corroded reactor target slugs at the L-Reactor disassembly basin. Planned SNF activities at Savannah River include the Interim Management of Nuclear Materials (IMNM) EIS decision. An expedited EIS process for the material at greatest risk (F-Canyon Plutonium Solutions) resulted in the issuance of a Record of Decision (ROD) in February 1995. The final IMNM EIS on the remaining material was issued in October 1995. In December 1995, DOE issued an initial ROD covering the majority of the other material, which stated that the Mark 36 fuel targets are to be processed. Subsequent ROD(s) are expected to address chemical processing options from the canyon utilization study and stabilization of the Mark 16 and Mark 22 fuel targets. Additionally, DOE is developing a technical strategy to treat, package, and store the fuel in a manner suitable for placement into long-term storage or a geologic repository. In November 1995, DOE formed a Research Reactor Fuel Task Team to assist in resolving interim storage and disposal issues for the aluminum-clad highly enriched uranium (HEU) SNF at Savannah River. Last, a dry storage facility or other treatment facility is planned to open by 2003.
DOE SNF MANAGEMENT AT INEL
Fig. 4
DOE SNF management activities and plans at INEL include preparation for the dry storage facility for the Three-Mile Island core debris SNF by 1998. DOE is currently making a decision on obtaining NRC licensing of the facility. At the ICPP-666 SNF storage facility, plans are to complete reracking of the ICPP-666 basin for additional storage after the court injunction is lifted. DOE also is planning construction of a dry modular storage facility by 2003.
A related issue involves SNF at Fort St. Vrain, Colorado. DOE is negotiating with the Public Service Company of Colorado to purchase and take over the facility, which will include taking over the license of the facility. Plans are to keep the SNF at Fort St. Vrain rather than ship the fuel to INEL.
PROGRAM MANAGEMENT TOOLS
A number of documents guide SNF program management activities. The following table gives information on the documents, date of issuance, and significance to program management:
Table I
The paper by J. Baker provides information on SNF Program technology needs and capabilities, and the paper by C. Head discusses FRR SNF.
SNF PATH FORWARD
As the above indicates, DOE has projected a number of important activities for the achievement of the path forward for the ultimate disposition of DOE-owned SNF. These include:
  Develop strategy for Storage/Disposal of Al-Clad HEU at Savannah River (6/96)
  TMI Core Debris Dry Storage at INEL (9/98)

Remove SNF from Hanford K-Basins (12/99)
Dry Storage facilities at INEL (2003)
Dry Storage or treatment facilities at Savannah River Site (2003)
Continuing preparations for disposal of SNF at a geologic repository.

However, various known or unforeseen considerations may affect these plans. One obstacle is determining and agreeing on acceptable forms and methods of storing, packaging, transporting, and disposing of DOE-owned SNF. For example, the DOE Office of Civilian Radioactive Waste Management recently issued interim waste acceptance criteria that impact SNF Program chemical processing activities, among others; the final requirements are due out in March 1996. More information on this topic is to be found in the paper on preparations for geologic disposal by K. Chacey. Another obstacle involves funding uncertainties, which greatly impact SNF Program resources, activities, and schedules.

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IMPLEMENTATION OF DOE-OWNED SPENT FUEL REGIONALIZATION

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ABSTRACT

In June of 1995, the Department of Energy (DOE) issued a Record of Decision (ROD) on Programmatic Spent Fuel Management for the next 40 years. The ROD authorizes the complex-wide regionalization of DOE-owned spent nuclear fuel by fuel type. This decision affects the approximately 2700 metric tons heavy metal (MTHM) of DOE-owned spent fuel currently in inventory, as well as the approximately 100 MTHM which is expected to be added to the inventory through the year 2035. Subsequent to the issuance of the ROD, a Settlement Agreement between the Department of Energy, the Department of the Navy, and the State of Idaho was reached on October 17, 1995. Requirements set forth in the Settlement Agreement will result in modifications to the ROD.

Based on the ROD, as modified by the Settlement Agreement, three DOE sites were selected for regionalized management of the Department's spent fuel inventory. Spent fuel currently in storage at the Hanford Site will remain at Hanford, except sodium-bonded fuel which will be shipped to INEL. The INEL was designated as the storage site for non-aluminum clad spent fuel, including Naval spent fuel. The Settlement Agreement restricts the amount of spent fuel to be shipped to the Idaho National Engineering Laboratory (INEL) over specified time intervals. The Savannah River Site (SRS) was designated as the storage site for aluminum-clad spent fuel. The spent fuel currently in storage at Fort St. Vrain (FSV) in Colorado will remain in dry storage there until a repository or interim storage facility outside of Idaho is developed and is accepting fuel from INEL. Once this requirement is met, FSV fuel may be transferred to INEL for treatment, then returned to Colorado for storage or transferred directly to the repository.

The Department has developed a DOE-owned Spent Nuclear Fuel Interim Storage Plan to integrate national and site-specific plans needed to safely and efficiently manage spent fuel over the next 40 years. The

scope involves the integration of planning and implementation activities associated with the spent fuel, with the facilities utilized for fuel storage, handling, characterization or conditioning, and with the transportation activities required to implement regionalization and transition of DOE-owned spent fuel into improved long-term interim storage.

This paper summarizes the activities and challenges associated with the management of spent fuel within existing facilities, the acquisition and management of new spent fuel facilities, and the transportation of spent fuel both onsite and between sites to implement regionalization by fuel type.

THE RECORD OF DECISION

During the last 40 years, the Department of Energy and its predecessor agencies have generated, transported, received, stored, and processed spent nuclear fuel at various facilities in the Department's nationwide complex. Currently, approximately 2700 metric tons heavy metal (MTHM) of DOE-owned spent nuclear fuel remain that have not been processed. Additionally, DOE-owned spent fuel containing approximately 100 MTHM is expected to be generated in the next 40 years, including possible receipt of Foreign Research Reactor spent fuel, if a decision is made to reinstitute the policy of returning such fuel to the United States.

In order to evaluate the potential impact of actions proposed to safely, efficiently, and responsibly manage existing and projected quantities of DOE-owned spent fuel through the year 2035 (pending ultimate disposition), the Department developed a Programmatic Environmental Impact Statement (PEIS). The PEIS provided four basic approaches to spent fuel management, which were then evaluated through a formal screening process and weighed against established criteria. The environmental impacts of implementing any of the alternatives were evaluated in detail and determined to be small. Two of the four candidates, the "1992/1993 Planning Basis" and "Regionalization by Fuel Type," were selected for more detailed analysis. These two candidate alternatives were then evaluated against a number of technical and nontechnical considerations, including environmental impact perception, indicated stakeholder preferences (drawn from public meetings), implementation factors, regulatory risk, spent fuel processing potential, environmental justice, and fairness. The ROD documented the bases for the Department's decision to implement the regionalization of spent fuel by fuel type (1).

This decision authorizes the regionalization and consolidation of existing and newly generated spent nuclear fuel at three major DOE sites based on the fuel type (pending future decisions on ultimate disposition). Under this decision, the fuel type distribution would be as follows:

- Hanford production reactor fuel will remain at the Hanford Site;
- Aluminum-clad fuel will be consolidated at the Savannah River Site (SRS); and
- Non-aluminum clad fuels (including Naval spent fuel) will be transferred to the Idaho National Engineering Laboratory (INEL).

For planning purposes, it is assumed that DOE-owned spent nuclear fuel that is not otherwise dispositioned will be emplaced in the first geologic repository for spent nuclear fuel and high-level radioactive waste, subject to physical and statutory limits, payment of fees, and compliance with waste acceptance requirements. (Current limitations allow for approximately 2330 MTHM of DOE-owned spent fuel, based on a

repository total of 70000 MTHM.) Except for some special-case commercial fuel, the decision does not apply to management of spent nuclear fuel from commercial nuclear power plants. The ROD does not address ultimate disposition of the Department's spent nuclear fuel.

THE SETTLEMENT AGREEMENT

Subsequent to the issuance of the Record of Decision, a Settlement Agreement between the Department of Energy, the Department of the Navy, and the State of Idaho was reached on October 17, 1995 (2). The Settlement Agreement places strict controls on shipment and storage of spent fuel at the INEL. The following is a brief summary of the requirements/restrictions imposed by the Settlement Agreement:

Allows the shipment of Naval spent fuel through the year 2035 (up to 575 total shipments with a total of 55 MTHM) required to refuel or defuel nuclear powered warships, and to examine the spent fuel created to support continued safe operation of existing warships.

Allows for essential shipments of DOE-owned spent fuel to enable the United States to recover spent fuel from reactors in foreign countries for purposes of nuclear non-proliferation, if such an acceptance policy is adopted. Allows 61 shipments of DOE-owned spent fuel prior to December 31, 2000, and 497 shipments after December 31, 2000, with maximum average annual shipment limits (includes the spent fuel at the West Valley Demonstration Project, possible additional foreign research reactor spent nuclear fuel, U.S. university research reactor spent fuel, and DOE laboratory and other spent fuel that DOE is obligated to accept for storage). Of the total 558 possible shipments, those coming from SRS would be received at INEL only as equivalent shipments of spent fuel leave INEL for storage at SRS.

Allows 244 additional spent fuel shipments from the Fort St. Vrain facility in Colorado if a repository or interim storage facility outside the State of Idaho is open and is accepting spent fuel from INEL. The fuel may then be transferred to INEL for the purpose of treatment (not storage) so that it is acceptable for the repository or interim storage facility. In the interim, this spent fuel will remain in safe NRC-licensed dry storage in Colorado.

Commits DOE to remove spent fuel from water storage pools at INEL by the year 2023, and to place the spent fuel into dry storage which is removed from the Snake River Plain aquifer to the extent technically feasible.

Prohibits any shipment of DOE-owned spent fuel to INEL after April 30, 1999, unless shipments of transuranic waste from INEL to the Waste Isolation Pilot Plant (located in New Mexico) or another facility are proceeding at a specified rate. Requires the removal of all transuranic waste from INEL by the year 2018, and the removal of all spent fuel from INEL by 2035, with enforceable requirements including payments to Idaho if spent fuel is not removed by that date.

DOE-OWNED SPENT NUCLEAR FUEL

The current inventory of DOE-owned spent nuclear fuel is composed of approximately 2700 MTHM, with an additional 100 MTHM projected to be generated over the next 40 years. Table I provides a summary of the locations and quantity (in terms of 3 metrics) of spent fuel currently existing within the DOE complex. Approximately ninety-nine percent (99%) of the existing DOE-owned spent fuel (by MTHM) is stored in facilities at four locations: the Hanford Reservation (81%); the INEL (10%); SRS (8%); and the West Valley Demonstration Project Site (1%). Naval propulsion

spent fuel comprises the majority of the projected receipts (as calculated by MTHM), with spent fuel from foreign and domestic research reactors providing another large category of spent fuel receipts. The projected increase in DOE-owned spent fuel inventories through the year 2035 represents approximately 4% by MTHM, or approximately 70% when using total mass or volume as the metric.

Table I

THE DOE-OWNED SPENT NUCLEAR FUEL INTERIM STORAGE PLAN

Planning for complex-wide management of DOE-owned spent fuel pending ultimate disposition has been impacted by both the ROD and the Settlement Agreement. DOE has developed an Interim Storage Plan to integrate national and site-specific activities proposed to consolidate, regionalize, and manage spent fuel during the interim period (3). Summaries of individual site plans are provided for the Hanford Site, the INEL, Oak Ridge National Laboratory (ORNL), and the Savannah River Site. Discussed are the national and site-specific activities needed to assure safe, interim storage within existing facilities as well as the actions underway to acquire new, primarily dry storage facilities for the longer term prior to transferring the spent fuel to the geologic repository for disposition.

A baseline Master Logic Schedule (MLS) has been developed to assist in the interim management of spent nuclear fuel and integration of complex-wide activities. This tool serves as an integral component of the program implementation strategy by establishing the logic and key milestones for the entire national program and displaying the paths that lead to permanent disposition of DOE-owned spent nuclear fuel. It is structured at several levels, with the highest levels summarizing key activities for the entire program over a 40-year period, and the lower levels providing more detail by site and facility.

Major activities proposed for the interim management of DOE-owned spent fuel are based upon managing spent fuel within existing DOE facilities, acquiring new facilities, and facilitating transportation of spent fuel between facilities. For implementation purposes, these major activities can be categorized into three general time frames or phases. Program and site priorities, budget constraints, resources availability, and scheduling/transportation requirements act to define the time frames for each phase, with some overlap expected.

Phase 1 activities are scheduled to take place approximately through the year 2000. These are the activities that are the most well-defined, and for which budget allocations or projections are already in place. These activities emphasize efforts associated with existing facilities, and in many cases represent actions that must be taken to prepare for major activities proposed in Phase 2. Budget projections and schedules reflect the priorities which have been placed upon resolution of vulnerabilities and Defense Nuclear Facilities Safety Board issues, facility upgrades, and onsite fuel consolidation. Initiation of spent fuel regionalization activities will take place on a small scale during Phase 1. Facility acquisition activities are focused on acquiring the Canister Storage Building at the Hanford Site and Three Mile Island Dry Storage facility at INEL.

Phase 2 is defined as the period beginning in the year 2001 and continuing through approximately 2006. Activities proposed in this phase are less well defined than in Phase 1, and budget projections and schedules are still under development. This phase will focus on the large

scale implementation of regionalization of spent fuel by fuel type, including transfers of fuel between the designated sites. Requested funding levels will reflect storage costs, transportation costs, and costs related to bringing on-line additional new storage facilities (e.g., the Dry Modular Storage Facility at INEL).

Phase 3 is defined as the period beginning approximately in 2007 and continuing through approximately 2035. This phase will include the completion of regionalization activities, but will focus more on the actual interim management of the spent fuel awaiting the availability of the geologic repository. During this phase there will be a measurable decrease in site budget projections, reflecting the transition to surveillance and maintenance activities.

MANAGEMENT OF SPENT FUEL WITHIN EXISTING FACILITIES

Most existing DOE storage facilities either have reached, or are approaching, the end of their original design life. Many of the facilities have vulnerabilities that may require costly upgrades or preclude reasonable life extension efforts, thereby warranting phaseout of the facility. Given sufficient funding, the prudent course of action would be to phase out these facilities at the earliest point in time. However, due to limited funding and a need to improve the near term storage conditions of much of the DOE-owned spent fuel, efforts must be undertaken to continue the use of some of these facilities. This Plan identifies those existing facilities that, while not fully compliant with today's standards, are technically adequate for continued near term use pending the acquisition of new facilities.

Hanford Site

The Hanford Site manages the largest amount of DOE-owned spent nuclear fuel with a current inventory of approximately 2133 MTHM, representing approximately 81% (by weight) of the Department's entire spent fuel inventory, the vast majority of which is production fuel.

At the Hanford Site, the most pressing issue has been the removal of N-Reactor spent production fuel from the aged K-Basins. In October 1995, all spent production fuel remaining in storage at PUREX was transferred to the K-West Basin. This fuel has been consolidated with the remaining K-Basin fuel which will be moved to a staging facility serving as the first phase of the new Canister Storage Building (CSB) scheduled to begin operation in late 1997. The completion of the removal of spent production fuel from the K-Basins is scheduled for December 1999. Once the conditioning system becomes operational at the CSB, the spent production fuel will then be cycled through a conditioning process, repackaged and moved to the main dry storage area located in the CSB, where it will remain until its scheduled transfer to the repository. The transfer of all spent production fuel to the repository is scheduled to occur between 2015 and 2035.

In addition to the spent production fuel currently in storage in the K-Basins, small amounts of non-production fuel, once used for experimental and other purposes, are also stored at other onsite facilities. Currently, this spent fuel is located in the Low-Level Burial Grounds (LLBG), T-Plant, Plutonium Finishing Plant (PFP), Fast Flux Test Facility (FFTF), and several 300-Area Buildings. In the near future, FFTF and the 300 Buildings spent non-production fuel will be temporarily moved to a 400-Area storage pad. When additional space becomes available, the spent fuel will be transferred to a 200-Area storage pad adjacent to the CSB, where all the spent non-production fuel will be consolidated with T-Plant

fuel and temporarily stored prior to being transferred to the repository site via the CSB pathway. PFP and LLBG fuels, once conditioned and repackaged at the Canister Storage Building, will await disposition via the CSB pathway at the PFP facility and 200-Area storage pad, respectively. The transfer of all spent non-production fuel, with the exception of sodium-bonded fuel from the FFTF, is scheduled to occur between 2015 and 2035. The sodium-bonded fuel is scheduled to be transferred to INEL for electrometallurgical treatment at Argonne National Laboratory-West (ANL-W), if this process is found to be feasible and practical. (This topic is addressed in Session 46 -Spent Fuel Conditioning and Processing Technologies: Needs and Capabilities," John F. Baker, William D. Clark, Jr., and James J. Laidler)

Idaho National Engineering Laboratory

The current inventory of spent fuel at INEL is 261 MTHM, representing approximately 10% (by weight) of fuel in the DOE Complex. Five major facility areas exist in and around INEL that store spent fuel: ANL-W, Idaho Chemical Processing Plant (ICPP), Power Burst Facility (PBF), Test Area North (TAN), and Test Reactor Area (TRA). Stored fuels are kept in a variety of both wet and dry configurations. The ROD identified INEL as the recipient of non-aluminum clad spent fuel, including Navy fuel, in the DOE complex (within certain limitations specified in the Settlement Agreement) for interim storage.

The INEL will consolidate spent fuel at the Idaho Chemical Processing Plant. To resolve existing vulnerabilities, fuel is being moved from an older wet storage facility (CPP-603) to a modern wet storage facility (CPP-666) or to dry storage. A new dry storage facility will be built at INEL to store damaged Three Mile Island-2 core debris . An additional dry storage facility will be built to store future receipts of Navy fuel and non-aluminum clad DOE-owned spent fuel and for further consolidation of fuel stored at various INEL facilities.

The Idaho Chemical Processing Plant historically received spent nuclear fuel from many onsite and offsite reactors for processing. However, DOE decided to phase out processing activities in 1992. The new mission of this facility area is to receive and store spent fuel. In implementing this mission, the majority of spent fuel will be consolidated in CPP-603 (Dry) from a number of onsite storage locations for temporary storage. Fuels currently stored at CPP-603 (Wet) will either be moved and placed into wet storage at CPP-666, or canned and placed in storage at CPP-603 (Dry). These activities will be completed by December 2000. As for other activities at ICPP, CPP-666 will serve as the interim storage location for Navy fuel and the near term storage location for Advanced Test Reactor (ATR) fuel. All wet storage operations within CPP-666 will cease by 2023, at which time all stored materials will have been transferred to dry storage either at the Navy's Expended Core Facility (ECF) located at INEL or at the CPP-666 location. All subsequent Navy fuel shipments which are scheduled to continue until 2035 will be received for dry storage at either the ECF or the CPP-666 directly. Furthermore, Peach Bottom fuel currently stored at INEL in CPP-603 (Dry) will be transferred to CPP-749 over a period of 12 months beginning in early 1997 where it will await transfer to the new Dry Modular Storage Facility. Beginning in 2001, receipts from ORNL will be placed in storage at CPP-749 and CPP-603 (Dry). Transfer of fuel to interim storage from CPP-603 (Dry) and CPP-749 to the new Dry Modular Storage Facility is scheduled to begin in 2003 and continue for at least a decade.

The Test Reactor Area and the Power Burst Facility, which currently store both aluminum-clad and non-aluminum clad research reactor fuel, will transfer their non-aluminum clad fuels to CPP-603 (Dry) for short-term interim storage. Aluminum-clad fuels including ATR fuel will be transferred offsite from CPP-666 to SRS, while the remaining non-aluminum clad fuel will follow the same disposition path as other CPP-603 (Dry) fuels. Fuel transfers from TRA and PBF will occur within approximately the next five years.

Test Area North stores special-case commercial fuel (including Three Mile Island core debris) and DOE experimental fuel similar to commercial nuclear fuel. A new facility, Three Mile Island (TMI) Dry Storage, is currently being planned, and will begin receiving TMI fuel currently stored in TAN (Wet) in 1998. The remaining fuels in TAN (Wet) will be transferred at the same time to the current TAN (Dry) facility for temporary storage and eventual transfer to the new Dry Modular Storage Facility. The TAN (Dry) facility will also receive fuel from West Valley for temporary storage beginning after 2000 and it will eventually be transferred to the new Dry Modular Storage Facility, along with the spent fuel currently in cask storage at TAN (Dry).

The remaining onsite fuel is stored at ANL-W, which generated spent fuel as a result of research and development activities related to advanced reactor design. Primarily, ANL-W will be designated the disposition facility/area for sodium-bonded fuel, including fuel from the Hanford Site. Following processing, fuel would no longer be classified as spent fuel and will not be managed by the spent fuel program. The high-level waste component from processing will ultimately be disposed of in the repository. If it is determined to be feasible and practical, processing of fuel at ANL-W will begin in approximately 2001 and continue for approximately eight years. An option to use a modified electrometallurgical treatment technique for the fuel at Oak Ridge's Molten Salt Reactor Experiment is being investigated and a determination as to its use will be made in the next 1-2 years.

The spent fuel currently in dry storage at Fort St. Vrain in Colorado will remain there for the near term. Receipt of Fort St. Vrain fuel will only take place if and when an interim storage facility or permanent repository is operating outside of Idaho and accepting INEL spent fuel. If these conditions are met, INEL will receive FSV fuel for the purpose of treatment only; storage or disposition of FSV fuel will occur elsewhere.

A number of different options are still being considered at INEL for long-term disposition, but the current plan is to consolidate spent fuel for interim storage at three facilities: Dry Modular Storage Facility, TMI Dry Storage, and CPP-666. Under the current plan, fuels stored in TMI Dry and CPP-666 (other than Navy fuel) would be transferred to the Dry Modular Storage Facility for conditioning/packaging before being transferred to the repository, beginning in 2015.

Oak Ridge National Laboratory

The Oak Ridge Reservation stores a relatively small amount (approximately 1 MTHM or 0.3% by weight) of DOE-owned spent fuel in comparison to the other DOE sites. Nevertheless, the large variety of types of spent fuel managed by Oak Ridge necessitates the need for an integrated disposition plan. Oak Ridge aluminum-clad spent fuel will be shipped to SRS for interim storage, and non-aluminum clad fuel will be shipped to INEL. Currently, spent fuel is stored in a variety of onsite storage areas

including retrievable dry storage units (SWSA 5N), as above ground hot cell units (Bldg. 3525 and 7920) and wet storage basins at the High Flux Isotope Reactor (HFIR) and the Bulk Shielding Reactor (BSR). Spent fuel generated at the Tower Shielding Reactor (TSR) is currently stored within the inactive reactor core. Spent fuel from the Molten Salt Reactor Experiment (MSRE) is housed in isolation within critically safe storage tanks.

Oak Ridge will ship the majority of its spent fuel located at the HFIR, BSR, and TSR facilities to the Savannah River Site beginning in 1996. Transfer of TSR and BSR spent fuel is expected to be completed by 1999. Production and transfer of HFIR fuel is scheduled to continue throughout its operating life, currently anticipated through 2015. The small amount of aluminum-clad spent fuel stored in SWSA-5N and Buildings 3525 / 7920 will be repackaged and transferred to the Savannah River Site between 1997 and 1999.

As for the non-aluminum clad fuel, Peach Bottom, Dresden and other miscellaneous fuels are stored at SWSA 5N and Building 7920. These fuels will be repackaged at Building 3525 and either returned to SWSA 5N prior to transfer to INEL, or transferred directly to INEL. Transfer of these fuels to INEL is scheduled to take place over five years, beginning in 2001, in accordance with the Settlement Agreement between DOE, the Navy, and the State of Idaho. Spent fuel from the MSRE may be processed utilizing a modified pyroprocessing technique onsite through 2005, with a determination as to using this process being made in the next 1-2 years. The high-level waste component from processing will be transferred along with other spent fuel to the repository.

Savannah River Site

The Savannah River Site manages 207 MTHM of DOE-owned spent fuel, representing approximately 8% (by weight) of spent fuel in the DOE complex. SRS currently stores the majority of the aluminum-clad spent fuel in the DOE-complex at the Receiving Basin for Offsite Fuels (RBOF), the K-,L-, and P-Reactor Disassembly Basins, and basins in the F- and H-Canyons. The F- and H-Canyons and the L- and P-Reactor basins presently manage production fuel and targets. The K-Reactor basin manages production fuel, while RBOF is currently the only facility to manage offsite and non-aluminum clad spent fuel.

SRS is finalizing a site-specific Environmental Impact Statement (EIS) evaluating options to store or process spent fuel. A combination of options including processing of deteriorated aluminum-clad fuel and dry storage of remaining spent fuel may be the preferred alternative. If the processing option is selected, deteriorated aluminum clad fuel from INEL may also be shipped to SRS for processing.

Aluminum-clad HFIR fuel from ORNL, domestic research reactors (DRRs) and other DOE facilities will be transferred to SRS for storage in RBOF or the K/L-Basins over a ten year period ending in December of 2005. During the first 24 months of these receipts, K/L-Basins will undergo upgrading to ensure the facilities continued integrity. In addition, RBOF or K/L-Basins would receive ORNL transfers of BSR and TSR fuels from late 1996 to the end of 1997, and ORNL 3525 and SWSA 5N spent fuel from 1997 through 1999. Spent fuel received at RBOF and the K/L-Basins will remain in storage until transfer to a new dry storage facility which is proposed to come on-line in about 2003. However, production fuel receipts to K/L-Basins may either be transferred to F- and H-Canyons for processing, at which time the fuel will no longer be classified as spent fuel, or the

production fuel will remain in storage at K/L-Basins until transfer to interim dry storage in 2003. Finally, those transfers from ORNL, DRRs, and other DOE facilities received after December of 2003 may go directly to a dry storage facility to await conditioning and ultimate disposition beginning in 2015.

Pending a Record of Decision on the Foreign Research Reactor (FRR) EIS, offsite receipts of FRR spent fuel to RBOF or the K/L-Basins for storage could start in late 1996. FRR spent fuel received between 1996 and 2000 may be subsequently transferred to F- and H-Canyons for processing, or remain at K/L-Basins until the transfer of all fuel from K/L-Basins to a dry storage facility in 2003. Aluminum-clad and ATR fuel receipts from INEL to RBOF or K/L-Basins will begin in the year 2001. FRR spent fuel and INEL fuel receipts to RBOF or K/L-Basins will cease in 2005; subsequent receipts might then go directly to a new dry storage facility to await conditioning and final disposition beginning in 2015.

Non-aluminum clad fuels currently stored at RBOF will be transferred to INEL. This will occur between 2001 and 2010 in accordance with the Settlement Agreement, requiring an equivalent annual number of spent fuel shipments between SRS and INEL until all aluminum-clad spent fuel presently stored at INEL has been shipped to SRS.

ACQUISITION AND MANAGEMENT OF NEW STORAGE FACILITIES

The safe, effective interim management of DOE-owned spent nuclear fuel will require the construction of several new interim storage facilities, including support facilities needed to characterize, stabilize or condition, and package the spent fuel, most probably under dry conditions. One primary driver that influences decisions as to facility need is safety, which dictates that new facilities are needed to provide improved safe management for many of the DOE-owned spent fuel types. A second driver exists in the need to prepare the spent fuel for final disposition in a geologic repository. To bring on-line these new facilities will require substantial efforts in the areas of contract management and funding, development and implementation of regulatory requirements, technology development, design selection, and National Environmental Policy Act (NEPA) analysis. When possible, new spent fuel storage and conditioning facilities are to use existing commercial technologies to minimize life cycle costs and to take advantage of commercial standards and licensing bases.

One consideration that arises during the acquisition of new facilities is the degree to which commercial industry regulatory requirements are to be adopted. The primary Federal agencies that promulgate requirements that potentially impact the management of DOE-owned spent fuel are the Nuclear Regulatory Commission (NRC) and the Environmental Protection Agency (EPA). Under the Atomic Energy Act (AEA), the NRC has regulatory authority for nuclear safety over commercial nuclear facilities as well as select DOE facilities that are identified by Congressional lawmaking. The EPA is the agency responsible for promulgating regulations implementing a number of environmental laws including, among others, the Clean Air Act, the Clean Water Act, the Resource Conservation and Recovery Act (RCRA), and the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). Of these, the AEA and RCRA provide the majority of requirements specific to spent fuel.

In addition, a Federal Advisory Committee on External Regulation of Department of Energy Nuclear Safety has been established to provide advice, information, and recommendations on whether and how new and

existing DOE nuclear facilities and operations should be externally regulated to ensure safety. The committee's final report recommends external regulation of essentially all aspects of safety at DOE's nuclear facilities and sites. DOE spent fuel storage and treatment facilities would be directly impacted when the recommendation is implemented.

At the site level, a number of regulatory initiatives are underway in conjunction with the acquisition of new facilities. As an example, at the Hanford site the K-Basin Replacement Project has established a Regulatory Requirements Team and an Independent Review Panel to help determine the regulatory requirements needed for design and construction of the new Canister Storage Building. The Regulatory Requirements Team is comprised of DOE-Richland, Westinghouse Hanford Corporation and expert personnel, and is responsible for the identification, evaluation, and recommendation of requirements for use on the project. An Independent Review Panel consists of three nuclear industry experts who review the recommendations provided by the Regulatory Requirements Team.

The acquisition of a new Three Mile Island Dry Storage facility at the INEL also affords an opportunity for the DOE to consider new approaches towards regulatory requirements. The DOE has evaluated proposals for a company to design and construct a dry storage facility based upon proven commercial facilities that have been designed in accordance with NRC requirements. The selection of the approved vendor was completed in late 1995.

SPENT FUEL TRANSPORTATION / MULTI-PURPOSE CANISTERS

Although the main focus of the Interim Storage Plan is to document DOE plans for managing spent fuel over the next 40 years, plans must also account for eventual disposition of the spent fuel. As ultimate disposition for the majority of DOE-owned spent fuel will be in a geologic repository, the national spent fuel program is working with the Office of Civilian Radioactive Waste Management (OCRWM) to develop disposal package designs, waste acceptance requirements, and technical requirements for emplacement of DOE-owned spent fuel within a geologic repository.

One concept being pursued is the use of multi-purpose canisters (MPCs) that provide for spent nuclear fuel interim storage, transportation to the repository, and repository emplacement. The Navy is currently evaluating the use of multi-purpose canisters for interim storage and permanent disposition of naval propulsion spent fuel.

The DOE will monitor progress in this area, and effort will be made to extend MPC designs for use with DOE-owned spent fuel as necessary for preparations for disposal proceed.

COSTS

The costs for implementation of interim storage for Fiscal Years 1996 through 2001 are estimated to be on the order of $1.5 Billion, which includes both the costs for operating existing facilities as well as the cost for the acquisition of new facilities.

CHALLENGES

In order to effectively implement the complex-wide regionalization of DOE-owned spent fuel, a large number of activities must be undertaken concurrently or consecutively, with close coordination among the sites, and with consideration given to long-term interim storage and preparation of the fuel for permanent disposal in a geologic repository. There are many challenges which must be addressed, including but not limited to the following:

a) Availability and suitability of transportation casks - An adequate number of casks must be available within relatively strict time frames in order for transportation to occur as needed. The suitability of existing casks must be evaluated for the myriad of potential spent fuel types to be transported. If existing casks are found to be lacking, either modifications will be required or new casks must be designed and constructed.
b) Availability of funding - All of the proposed plans assume that the necessary funding will be provided at adequate levels and in the time frame needed in order to accomplish all the required tasks. With the current turmoil concerning government funding in general, and Department of Energy activities specifically, obtaining necessary funding will be a challenge.
c) Stakeholder acceptance - Gaining stakeholder acceptance, whether it be members of Congress, State Governors, Tribal leaders, or members of the general public, will be difficult based on the many areas of concern. These areas include economic impact, equity, risk aversion, as well as concerns that potential delays in the Geologic Repository Program may lead to de facto permanent storage in the states in which the spent fuel is being stored.
d) Technology development - Some technology development will be required in order to stabilize the condition of many of DOE's spent fuel types so that they can be safely placed into interim storage.
e) Repository Program - The current funding problems being experienced by the Department's Repository Program could impact the development work in the areas such as Multi-Purpose Canisters and the development of waste acceptance requirements for the disposal of DOE-owned spent fuel in the geologic repository. Delays in either of these two areas could require multiple handling of spent fuel in order to place the spent fuel in a form and in packages that will be compatible with the yet to be developed MPC's and meet currently unspecified waste acceptance requirements.

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PROGRESS ON THE HANFORD K BASINS SPENT NUCLEAR FUEL PROJECT

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ABSTRACT

This paper highlights progress made during the last year toward removing the Department of Energy's (DOE) approximately 2,100 metric tons of metallic spent nuclear fuel from the two outdated K Basins at the Hanford Site and placing it in safe, economical interim dry storage. In the past year, the Spent Nuclear Fuel (SNF) Project has engaged in an evolutionary process involving the customer, regulatory bodies, and the public that has resulted in a quicker, cheaper, and safer strategy for accomplishing that goal. Development and implementation of the Integrated Process Strategy for K Basins Fuel is as much a case study of modern project and business management within the regulatory system as it is a technical achievement.

A year ago, the SNF Project developed the K Basins Path Forward that, beginning in December 1998, would move the spent nuclear fuel currently stored in the K Basins to a new Staging and Storage Facility by December 2000. The second stage of this $960 million two- stage plan would complete the project by conditioning the metallic fuel and placing it in interim dry storage by 2006. In accepting this plan, the DOE established goals that the fuel removal schedule be accelerated by a year, that fuel conditioning be closely coupled with fuel removal, and that the cost be reduced by at least $300 million.

The SNF Project conducted coordinated engineering and technology studies over a three- month period that established the technical framework needed to design and construct facilities, and implement processes compatible with these goals. The result was the Integrated Process Strategy for K Basins Fuel. This strategy accomplishes the goals set forth by the DOE by beginning fuel removal a year earlier in December 1997, completing it by December 1999, beginning conditioning within six months of starting fuel removal, and accomplishes it for $340 million less than the previous Path Forward plan.

DOE approval of the Integrated Process Strategy was received in late July 1995 and its implementation is well underway. Because this is a strategy as compared to a fully engineered solution, technical challenges must be overcome and continually adjusted while accelerated engineering, procurement, construction, and startup activities are proceeding in parallel. A comprehensive technical integration process including the requirements baseline, process flow diagrams, interface control, issues management, technical data management, and configuration control has been implemented to manage and communicate these adjustments throughout the Project. These modern business practices are central to the success of the Integrated Process Strategy for K Basins Fuel.

INTRODUCTION

At last year's WM '95 Conference, the K Basins Path Forward was described. This was a strategy for the early removal of fuel and sludge from the K Basins to quickly mitigate environmental concerns and public and worker health and safety issues. In the year since then, many changes to those plans have occurred through working with the customer, regulatory bodies and the public and incorporating changing and improved requirements. The result of this evolutionary process is the Integrated Process Strategy for K Basins Fuel. This new strategy and how it effectively meets the challenges is the subject of this paper.

Development of the Integrated Process Strategy for K Basins Fuel is as much a case study of project and business management within the

regulatory system as it is a technical achievement. The paradigms of DOE's production past are no longer operative in today's complex political, regulatory, and public involvement environment. Previously, a technical problem would be thoroughly studied and evaluated before any plans were made. A technically elegant solution would be developed without public or regulatory input. A budget and schedule would then be adopted that supported this solution, a Congressional funding appropriation obtained, and the project executed.

The SNF Project at Hanford has shown that modern business practices can be applied within the DOE system, albeit, dramatic shifts away from the old cost-plus processes are required. The business practices being applied by the SNF Project are not new or different from those used in private industry. In today's environment many factors drive the technical solution which must often be adjusted before it is fully implemented as those factors change. The SNF Project has successfully combined the practical "can do, get the job done" attitude of the past with the political, regulatory, and public realities of the present.

SPENT NUCLEAR FUEL PROJECT AND THE K BASINS PATH FORWARD

The inventory of spent nuclear fuel at the Hanford Site covers a wide variety of fuel types (production reactor to space reactor) in many facilities (reactor fuel basins to hot cells) at locations all over the Site. The 2,129 metric tons of spent nuclear fuel at Hanford accounts for over 80% of the total U.S.DOE inventory. Most (98%) of the Hanford spent nuclear fuel is 2,103 metric tons of metallic uranium production reactor fuel currently stored in the K Basins located near the Columbia River at the northern end of the Hanford Site.

These two basins each contain about 3,800 cubic meters (one million gallons) of water and were part of the K East and K West production reactor complex constructed in the 1950s. They were used to cool discharged fuel prior to chemical processing at the PUREX facility. After the K Reactors production mission ended in the 1970s, the basins were used as temporary storage capacity for N Reactor fuel while the PUREX processing facility was being refurbished and restarted. When the defense production mission ended in 1990 and PUREX operations were terminated in December 1992, part of the N Reactor fuel inventory remained in the basins with no means for near-term removal and processing. The K East Basin contains 1,146.2 metric tons of spent fuel and the K West Basin contains 956.8 metric tons of spent fuel. (This includes the 2.9 metric tons of spent fuel recently transferred from PUREX to the K West Basin.) The fuel in the K West Basin has been stored in water filled, lidded canisters containing a corrosion inhibitor and the basin environment is relatively clean and free of corrosion products. The fuel in the K East Basin, however, was stored in open canisters and some of the fuel has corroded extensively. The deteriorating fuel with no viable "path forward," and worker health and safety issues, coupled with aging facilities with seismic vulnerabilities, has been identified by several groups, including regulators, stakeholders, and the DOE, as being one of the most urgent safety and environmental concerns at the Hanford Site.

To meet these challenges, the Spent Nuclear Fuel Project was formed in February 1994. The mission accepted by the SNF Project is to "Provide safe, economic, and environmentally sound management of Hanford spent nuclear fuel in a manner which stages it to final disposition." A year ago, the SNF Project developed the K Basins Path Forward, a two-phase strategy to allow early removal of fuel and sludge from the K Basins

followed by its stabilization and interim storage. In the expedited response phase, fuel and sludge from the K East and K West Basins would be packaged in large multi-canister overpacks (MCO). The fuel would remain in the original canisters while the sludge would be placed in containers resembling the fuel canisters. The canisters would be loaded into the MCOs under water and the MCOs would remain water filled during shipping to temporary storage at a new Staging and Storage Facility. During the second phase, MCOs would be transferred from the Staging and Storage Facility to a newly constructed Fuel Stabilization Facility which would be co-located with the Staging and Storage Facility. Here the fuel and unseparated sludge would be dried and passivated in the MCO to reduce the potential hazards associated with metal fuel storage. Following this stabilization process, the MCOs would be returned to the Staging and Storage Facility for interim dry storage. This strategy would begin removal of the spent fuel and sludge from the basins in December 1998 and place it in a new Staging and Storage Facility located in the 200 Area at the Hanford Site by December 2000. The K Basins Path Forward would complete placing the fuel and sludge in interim dry storage by the year 2006 for an estimated cost of $960 million.

K BASINS PATH FORWARD APPROVAL AND REFINEMENT PROCESS

In February of 1995, the DOE formally approved the K Basins Path Forward and provided the following goals for its refinement:

Accelerate the Path Forward schedules by one year to commence fuel removal by December 1997, and to complete fuel removal in approximately two years, i.e. by December 1999.

Accelerate the schedules for fuel conditioning activities so that fuel stabilization is closely coupled with fuel removal from the K Basins.

Achieve significant cost reductions, with every attempt to complete implementation of the accelerated schedules including close coupled fuel stabilization within the fiscal year (FY) 1995 through FY 1997 budget projections.

Revise the process to avoid reopening the MCOs once fuel is loaded so that the MCOs can be welded at the K Basins.

Utilize the partially constructed Canister Storage Building (CSB) located in the 200 East Area as the Staging and Storage Facility recommended in the Path Forward strategy.

In response to the DOE's goals, the SNF Project conducted coordinated engineering and technology studies over a three-month period to establish the technical framework needed to design and construct facilities, and implement processes compatible with these goals. To achieve a timely and optimum evaluation and solution, Westinghouse Hanford Company, in conjunction with Pacific Northwest National Laboratory (PNNL), DOE staff, and several subcontractors, formed an integrated team to coordinate the studies and arrive at decisions. The overall process and timeline for developing the Integrated Process Strategy was aggressive. Beginning in April 1995, the technical options and criteria were developed, cost, schedule, and safety input provided, and the evaluations were performed. The resulting integrated process strategy was reviewed by an external advisory panel of experts between July 11-14, 1995, and their advice incorporated. The recommended strategy was finalized and provided to the DOE on July 18, 1995, for their consideration.

The decision process built upon the foundation established by the K Basins Path Forward and incorporated the schedule acceleration and

compression, and cost reduction goals. The team identified the following key technical issues which afforded flexibility for improvements:

- Selection of the process for drying and conditioning the fuel.
- Wet (flooded) versus dry shipment of fuel.
- Wet (flooded) versus dry staging of fuel.
- Canister preparation and desludging requirements.
- MCO venting and hydrogen management requirements.
- Definition of an acceptable (defensible) safety basis.

Technical and engineering trade-off studies to resolve these issues were performed and a systems engineering process then resulted in process alternatives as shown in the summary block diagram in Fig. 1. Eight integrated process options were developed from these alternatives for evaluation. These options, the evaluation results, and cost comparisons are summarized in Table I. A comparison with the initial Path Forward of the same project scope is also included. The evaluation of these alternatives included the use of systematic multi-attribute decision analysis techniques to assure a comprehensive, balanced treatment. The evaluations included assessments of technical viability, health, safety and environmental risk, and cost, schedule, and programmatic risk.

Fig. 1

Table I

Cost was a major factor in evaluating process alternatives. The guidelines provided by DOE required that about $300 million be cut from the total $960 million project cost of the Path Forward and that the near-term budget profile be met. Major cost savings are realized by accelerating fuel removal from the basins through an early significant reduction in K Basins operating costs. Options that increase the packing density of fuel and a corresponding reduction in the number of MCOs can also have a significant impact on cost. Simplified fuel conditioning processes using commercially available equipment that treats the fuel within the MCO can also reduce costs. Because cost is not the only evaluation concern, higher cost alternatives were still considered in selection of the preferred alternative.

Another major evaluation criteria focuses on achieving initiation of fuel transfer away from the Columbia River at the earliest possible date, while maintaining worker safety and minimizing life-cycle costs. Those options with high or very high schedule risk are, therefore, eliminated from further consideration. Option 6, which features shipping fuel as-is in water-filled MCOs and wet staging, has the lowest schedule risk. However, this scenario has safety concerns and high costs. Option 1 is the lowest cost, but has a number of technical and safety uncertainties. Option 7, the recommended option, has moderate schedule risk, good technical viability, and acceptable safety/risk features. Option 7 will also produce a cost savings of approximately $340 million over the original Path Forward estimate and is consistent with the DOE budget goals.

INTEGRATED PROCESS STRATEGY

Option 7 was named the Integrated Process Strategy for K Basin Fuel and recommended to the DOE as the best method of achieving the cost and schedule goals. The DOE accepted the recommendation in late July 1995 and the SNF Project began implementing the strategy. The major elements of the strategy and the schedule for their acquisition are shown in Fig. 2. Each of the individual process steps are closely interrelated. For example, the fuel retrieval operation is designed to prepare the fuel for

the drying operations that follow. The features of each process step are as follows:

Fuel Retrieval During fuel retrieval the fuel canisters will be moved to a centralized work location within pool, the fuel removed, separated from the sludge, and loaded into tier baskets. After the tier baskets are filled with fuel, five or six baskets (depending on fuel length) will be loaded into a multi-canister overpack (MCO) and the closure welded in place. The MCO will be placed in a cask, removed from the basin, and transported to the Cold Vacuum Drying Facility.

Cold Vacuum Drying At the Cold Vacuum Drying Facility, the water will be removed from the MCO before shipping it to the Canister Storage Building (CSB) by a combination of draining and cold vacuum drying. Vacuum drying will require only 24 to 48 hours to remove all of the free water. Removal of the free water greatly reduces the hydrogen generation rate and allows the MCO to be shipped to the CSB with the process valves closed, greatly enhancing safety during shipping.

Staging in the CSB Upon arrival at the CSB, the MCOs will be staged in the storage tubes until they are prepared for interim storage with the conditioning process. Venting of hydrogen generated during staging will be through a high efficiency particulate air (HEPA) filter into inerted storage tubes.

Hot Vacuum Conditioning The MCOs will be removed from staging and placed in the Hot Vacuum Conditioning Facility within the CSB where the metallic uranium fuel in the MCOs will be prepared for long-term interim dry storage in the CSB. The hot vacuum conditioning process will be designed to eliminate most of the chemisorbed water and mitigate further fuel corrosion and hydrogen generation. The MCO will be used as the process vessel and operated at approximately 300C. The process, pending validation by the testing program, will consist of heating, vacuum drying, surface oxidation, cooling, and monitoring phases. The total cycle time for hot vacuum conditioning is estimated to require approximately six to eight days. Following conditioning, the MCOs will be sealed and returned to the storage tubes where they will remain until their ultimate disposition is determined.

Fig. 2

TECHNICAL INTEGRATION

Because the Integrated Process Strategy for K Basin Fuel is a strategy as compared to a fully engineered solution, there are many technical issues to be resolved as implementation proceeds. Effective technical integration is crucial to the success of the strategy because many accelerated engineering, procurement, construction, and startup activities are proceeding in parallel. Maintaining a traceable record of how issues are resolved and how key decisions are reached is even more challenging. The SNF Project has developed a technical integration process to effectively communicate the large volume of rapidly changing technical information.

The cornerstone of this technical integration process is the Technical Baseline Description. The Technical Baseline Description is a systems engineering based body of technical information that documents the project-level functions and requirements along with enabling assumptions, issues, trade studies, external interfaces, and products. A key feature of the Technical Baseline Description is the process flow diagrams (PFD). The PFDs have been developed to provide an integrated picture of the overall Integrated Process Strategy. They document and illustrate the

material flows and interfaces among the subprojects and external interfaces. Management of the interfaces identified through the PFDs is performed by an interface control procedure. A centralized database has been developed to aid in the management of the interfaces and document agreements to maintain interface control. A technical issues management system has been implemented within the SNF Project to aid in the tracking, documentation, and timely resolution of issues. A two-tier system is used to maintain a traceable record of how issues are resolved and how key decision are made. The final element of the technical integration process is configuration management. The SNF Project has developed and implemented a configuration control procedure that keeps Project documentation, design requirements, and physical configuration of equipment and facilities consistent with each other.

SUMMARY

The SNF Project at Hanford has employed modern business practices to develop and implement a strategy, consistent with DOE budget and schedule goals, for accelerated removal of the spent nuclear fuel stored in the two K Basins. An Integrated Process Strategy for K Basins Fuel was developed through trade studies and evaluation of alternatives to select a process strategy that minimizes schedule risk and provides opportunities for cost savings. Compared to the previous Path Forward plan, the Integrated Process Strategy will accelerate the schedule for fuel removal by a year, compress the schedule for fuel conditioning and its placement in interim dry storage by more than five years and reduce the life cycle costs by more than $340 million. Implementation of this strategy is underway with engineering, procurement, construction and startup activities occurring in parallel. The evolving technical solutions supporting the strategy are continually being adjusted to accommodate changing requirements as the Integrated Process Strategy is being implemented. A comprehensive technical integration process has been established to manage changes to the technical baseline and communicate them throughout the Project.

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SPENT FUEL TREATMENT TECHNOLOGIES: NEEDS AND CAPABILITIES

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ABSTRACT
The ultimate disposal of fuels in the DOE spent nuclear fuel inventory in a mined geologic repository may require that certain fuel types be treated prior to disposal. This could be mandated by chemical reactivity, instability of the fuel in a repository environment, or by concerns over the potential for criticality events. The Department of Energy has at hand a variety of treatment methods that could be applied to spent nuclear fuels for disposal; some of these methods represent well-established technology (the PUREX aqueous solvent exchange process, for example), some are at advanced stages of development and demonstration (e.g., the electrometallurgical treatment process and the plasma arc melting process), and other methods that have promise (such as the GMODS process, and the chloride volatility process) are at very early stages of development. Institutional issues arise when consideration is given to spent fuel processing, and these issues are clouded by misconceptions about the nature of the treatment processes proposed: in the treatment of DOE spent nuclear fuel for disposal, there is no intention, and in some cases no ability, to recover plutonium or other fissile materials, for military purposes. The fissile elements (unless the uranium is recovered for commercial use) are incorporated in the final waste forms to be disposed in a high-level waste repository. The sole purpose of the process is to place the radioactive material in a form that will be acceptable in the repository.
INTRODUCTION
The U.S. Department of Energy (DOE) presently holds title to a variety of spent nuclear fuel (SNF) types, with the total inventory amounting to approximately 2,700 metric tons (heavy metal content). This SNF is stored at a number of DOE sites around the United States, with the major components of the inventory being located at the Hanford site in the state of Washington, the Idaho National Engineering Laboratory in eastern Idaho, and the Savannah River site in South Carolina. Over 130 different fuel designs are represented in the DOE inventory; these can be grouped into 53 different categories according to common design features such as composition, cladding material and enrichment level. The fuel can be further reduced into six categories according to fuel composition: metal, oxide, graphite, cermet, hydride, and aluminum based. A large percentage of the fuel is chemically reactive and precautions must be taken during preparation of the fuel for interim storage and final disposal. A small fraction of the fuel may have Resource Conservation and Recovery Act (RCRA) features that presently would preclude them from repository disposal. A large number of the fuel types are highly enriched in either 235U or plutonium, making the final disposition of the spent fuel in a geologic repository subject to resolution of potential criticality issues. The process of qualifying such a large number of fuel types for repository disposal, which may be necessary for those fuel types differing substantially from zirconium alloy-clad low-enrichment oxide fuel (i.e., commercial light water reactor spent fuel), could be more expensive than some form of treatment.
The DOE spent fuel is presently in storage in a variety of facilities, many of which do not meet the current standards of safety and

environmental protection required of commercial spent fuel storage facilities. In some of the DOE facilities, the spent fuel has begun to degrade significantly (1) and prompt action is necessary. Current DOE planning would place the spent fuel in interim dry storage at DOE sites until such time (some 20-40 years hence) that means are available for permanent geologic disposal or a national interim storage facility is available. While this interim step would serve to prevent serious environmental consequences that could result from containment failures in existing storage facilities, it cannot be regarded as the final answer to the spent fuel disposal problem.

Disposal of DOE spent nuclear fuel in a geologic repository will be subject to the same demanding requirements on waste performance that will be imposed on commercial spent nuclear fuel as a condition for acceptance for emplacement in the repository. Even though the quantity of commercial spent fuel greatly exceeds the amount of spent fuel in the DOE inventory, the DOE spent fuel represents a set of significantly different disposal problems that may require that the fuel be processed to separate the various constituents of the spent fuel prior to disposal, or treated to place it in a form that is amenable to disposal.

INSTITUTIONAL ISSUES

The position of the U.S. Government, as held by several recent administrations and formalized by Presidential Decision Directive #13 ("PDD-13"), is that the United States will not engage in reprocessing for the purpose of separating plutonium, for military or civil uses, from spent nuclear fuel, nor will the U.S. encourage such practices by other countries (although it will be tolerated by those countries which already have such industrial capabilities or ongoing reprocessing programs). Reprocessing of spent fuel is being carried out on an industrial scale by the United Kingdom and France, and Japan is building an 800 MT/yr plant that will go into operation in the next century. The United States reprocessed irradiated fuel for several decades for the purpose of recovering defense materials, employing the PUREX process for separating plutonium in plants with capacities as high as 3,000 MTHM/yr.

European and Japanese organizations are proceeding with the reprocessing of commercial reactor spent fuels and plan to recycle recovered plutonium as mixed oxide fuel (MOX) for their commercial reactors. No such plans exist in the United States, except as an option for denaturing the stockpile of surplus fissile materials (2).

The American Nuclear Society recently convened a group of distinguished experts in the field for a review of national policy concerning plutonium recycling and future energy supplies. Among their conclusions (3) was a finding that R&D into separation and processing technologies should continue, for possible application in the future. In a somewhat related study, the National Research Council's Committee on Separations Technology and Transmutation Systems also concluded (4) that continued development of separations and transmutation technologies is desirable.

The treatment of DOE spent nuclear fuel, when it involves a separation of certain radionuclides for the purpose of facilitating their disposal, bears some relationship to the reprocessing technologies that are the subject of present controversies. However, there is a fine line between spent fuel treatment as proposed by the Department of Energy for its spent fuel and the more conventional reprocessing methods that have been used in the past for the separation of fissile materials for defense purposes. Both applications are based on fuel dissolution and separation

of fission products, but it must be emphasized that the application for disposal of spent nuclear fuel is not intended for recovery of plutonium for military use. The existing aqueous technology can be modified to limit the decontamination of plutonium, and advanced technologies that are being developed are incapable of producing separated plutonium.

PUREX PROCESSING

The capability still exists for large-scale PUREX processing in the United States, as embodied in the F- and H-Canyon facilities at the Savannah River site. The F-Canyon facility, which began operation in 1954, is a typical PUREX process facility designed primarily for low-enriched uranium metal fuel and targets. The H-Canyon facility, which also began operation in 1954, employs a modified PUREX process to support highly-enriched uranium processing, with additional facilities for production of special isotopes (e.g., 238Pu and 237Np).

In typical F-Canyon operations, aluminum-clad low-enriched or depleted uranium fuel and targets are charged to the dissolver at a rate up to 10 metric tons per day. Sodium hydroxide (NaOH) is added to the dissolver in the presence of sodium nitrate ($NaNO_3$) to dissolve the aluminum cladding. The uranium metal is then dissolved using nitric acid (HNO_3), and the resulting aqueous uranyl nitrate solution is transferred to the first solvent extraction cycle where fission products are removed and uranium is separated from plutonium. The fission product stream is diverted to the high-level waste tanks, the organic stream is recycled, and the plutonium and uranium streams proceed to the second solvent extraction cycle, which further purifies the plutonium and uranium solutions. The plutonium stream is sent to the plutonium finishing facility (F B-Line) for conversion to metal or oxide, and the uranium stream is transferred to the F A-Line facility where it is denitrated and converted to uranium oxide.

The modified PUREX process utilized in H-Canyon is specially designed to support operations with enriched uranium-aluminum alloy fuels. In the head-end portion of the facility, nitric acid catalyzed by mercuric nitrate is used to dissolve the entire uranium-aluminum fuel element. The dissolver solution is clarified and fed to the solvent extraction process. In the first solvent extraction cycle, uranium and neptunium are extracted from the fission products. The fission products and the very small quantity of plutonium from the fuel are diverted to the high-level waste tanks while the uranium and neptunium proceed to the second solvent extraction cycle. The uranyl nitrate hexahydrate solution historically was converted to uranium metal or oxide at the Oak Ridge site, and the neptunium solution is converted to oxide in the Savannah River site H B-Line. Conversion at SRS of the highly enriched UNH to oxide after blending down to low enrichments for potential commercial use is being considered.

Continuing equipment upgrades and process modifications were made over the years in both the F- and H-Canyons. In 1992, both facilities were shut down for resolution of safety concerns, conduct of operations improvements, and general maintenance. When uncertainties in the long-term mission arose, this temporary suspension of operations became an extended standby period, leaving large quantities of highly-enriched uranium and plutonium in solution awaiting conversion to stable forms. The F-Canyon and F B-Line facilities have recently restarted to support the conversion of in-process plutonium and uranium solutions to forms

suitable for long-term storage. A similar operation in H-Canyon is being planned to deal with the in-process solutions stored in that facility. There are nearly 200 metric tons (heavy metal content) of aluminum-based fuel in the DOE spent fuel inventory, including both fuels related to defense materials production and fuels from research and test reactors, both domestic and foreign. If it is determined that these fuels cannot be directly disposed in a geologic repository without prior processing, the Savannah River facilities can easily accommodate their treatment. The cost of process development is nil, and the incremental increase in high-level waste volumes would be relatively minor.

TREATMENT TECHNOLOGIES

The treatment of certain DOE spent fuel types has a sound technical basis. A number of distinct treatment technologies are available, at various stages of development. The PUREX process, has been in practice for decades and is in current use in several countries. Technologies that are still under development are designed to deal with the drawbacks usually associated with the PUREX process, such as proliferation risk, cost, and volume of wastes (both primary and secondary) produced.

Vitrification

Vitrification of DOE SNF is being developed by the Savannah River Site (SRS) to convert SNF into a borosilicate glass waste form containing fissile material. This process is a variation on the PUREX process previously described.

The technique is based upon a dissolution process without fissile material separation followed by a vitrification process. The proposed dissolution process would be similar to the existing dissolution used in the SRS processing facilities to place the SNF into a liquid state. Aluminum SNF could be dissolved in the existing canyon dissolver. Stainless steel and zirconium clad SNF would be dissolved in a electrolytic dissolver. If the existing canyon dissolvers are unavailable for processing the aluminum based fuels, the new electrolytic dissolvers could be used. The solution in the electrolytic dissolver is nitric acid saturated with boric acid. The dissolvers are operated in a batch process. The uranium concentration in the solution is maintained below 10 grams per liter for nuclear safety (criticality control). The mass ratio of cladding to uranium will vary between 3 and 60. In contrast to the historic canyon operations, the resulting solution is then transferred to a holding tank without the fissile material being chemically stripped away.

Chemical and isotopic analysis will be performed on the solution in the holding tank. The solution will be adjusted with the addition of appropriate chemicals (e.g. to a desired solids concentration). The solution is then transferred to a melter feed tank.

The feed solution, glass frit and other glass forming chemicals will be fed to a joule-heated melter. In the melter the materials will be fused together to form a borosilicate glass. The molten glass will be poured out of the melter and into stainless steel canisters. This process is similar to the process used by the Defense Waste Processing Facility (DWPF) currently completing startup at the Savannah River Site. The melter can be sized to process the feed material from at least four electrolytic dissolvers. The resulting waste form is expected to be bounded by current constraints on borosilicate glass as an approved waste form for placement in a geologic repository.

Melt-Dilution/Poison

A melt-dilution process is being developed at SRS to provide a simplified, easily demonstrated technology for the conversion of aluminum based spent nuclear fuels into a waste form suitable for emplacement in a geological repository. The product of the melt reconfiguration treatment is a compacted (volume reduced), metallic waste form that contains uranium, fission products and plutonium in an aluminum matrix, all in a "container" which has corrosion characteristics superior to those used for the LWR waste forms. This technology used a single step, melting process to reduce the volume of the spent fuel, reduce the uranium enrichment to minimize criticality risks and provide an easily characterized metallic waste form which minimizing the creation of secondary wastes. Further, if desired, by using slag-metal technology, certain radio nuclides may be selective leached so as to produce the final metallic waste form containing little volatile fission products species.

Benefits of this process include very low capital investment for the operation; a superior waste form comparable to the LWR waste forms; a flexible process which can be readily adapted to the addition of poisons; adaptable to the addition of glass frit to develop borosilicate glass waste form if necessary; and low potential for airborne contamination and radiation exposure.

The process is based on the simultaneous melting of spent fuel sections in a permanent mold that is contained in a resistance furnace. The furnace is enclosed in a chamber which is connected to an off-gas system. The spent fuel gradually fills the container in the furnace. A lid or cap would be welded to the top of the mold to provide the waste form. Depleted uranium and/or other neutron poisons, if required, could also be incorporated into the melt during melt-dilution operation. Further, glass frit can also be added during the melt-dilution process if a borosilicate glass waste form is desired for a certain variety of spent fuels.

A Laboratory scale melt facility capable of melting sections of Mark 22 aluminum clad spent nuclear fuel elements was developed at SRS. A prototype furnace was also constructed. The melt-dilution process was demonstrated with non-irradiated aluminum clad fuel tubes. Full scale demonstrations, including cap welding and both destructive and non-destructive characterization and evaluation of the simulated waste form will follow the selection of optimum container material(s). Demonstration testing using full length irradiated fuel can be subsequently initiated. Although, the melt-dilution technology is primarily applicable to the aluminum based fuels, utilization of this technology would convert this category of DOE owned, spent nuclear fuels into a repository suitable waste form with minimal impact to future disposition options should other options be required. The melt-dilution process will provide a standard waste form package that retains the uranium in form very suitable for recovery, should future generations desired to utilize the available energy. Additionally, if repository suitability is achieved simply through the introduction of nuclear poisons rather than dilution with depleted uranium, the highly enriched uranium could be recovered through a relatively simple metal smelting operation.

Electrometallurgical Treatment

A standard, cost-effective means for treating the broad variety of DOE spent fuel for ultimate disposal is needed, and the electrometallurgical treatment process being developed by DOE's Argonne National Laboratory appears to meet these requirements. By use of this process, all of the

(stainless steel or zirconium alloy clad) metal and oxide spent fuel stored at the INEL and Hanford sites can be treated by one common method, producing two common high-level waste forms. One waste form is a mineral waste that contains most of the fission products in a ceramic material produced by hot-pressing a composite of borosilicate glass and zeolite. This waste form will be of essentially invariant composition regardless of the type of spent fuel processed. The other waste form is a metal alloy, the composition of which depends largely on the structural materials present in the fuel. For stainless steel- or zirconium alloy-clad fuels, the metal waste will have an iron-zirconium alloy matrix. Minor variation of the process can be used to treat the Molten Salt Reactor Experiment (MSRE) fuel salt at Oak Ridge National Laboratory and the aluminum-based fuels at the INEL and Savannah River sites. The commonality of process equipment and waste forms offers the potential for significant cost savings in the ultimate disposition of these spent fuels.

The key element of the electrometallurgical treatment process is the electrorefining step. This process is the same as the electrorefining process used for many years in the minerals industry: an impure metal serves as the anode in the electrorefining cell, and the pure metal is deposited at the cathode by electro-transport through a suitable electrolyte. In the electrometallurgical treatment process, the electrolyte is a salt compound, lithium chloride and potassium chloride. This electrolyte was selected because the chemical properties of the actinide elements and fission product elements in the chloride system are particularly well-suited to precise control of separation parameters. Because of these properties, it is possible to deposit virtually pure uranium, free of fission products and free of contamination with transuranic elements, at a simple iron cathode suspended in the electrolyte salt. The "impurities" in the spent fuel, the active metal fission products (for example, cesium and strontium) and the transuranic elements, remain in the electrolyte salt. The noble metal fission products (including technetium) remain in the anodic dissolution baskets in the cell, together with the cladding hulls from which the fuel has been dissolved.

The cathode deposits, consisting of pure uranium, are recovered after the desired amount of material has been collected and are then sent to a cathode processor, where they are consolidated by melting; in the process, any volatile materials (such as adhering electrolyte salt) that were included in the deposit are removed by vaporization. The uranium metal ingots resulting from the cathode processing operation are packaged for interim storage until their final disposition is decided.

The combined high-level waste volume produced in electrometallurgical treatment of spent fuel is approximately 1% of the volume of high-level wastes produced in the conventional reprocessing of spent fuel, and about 20% of the volume of the spent fuel when packaged for direct disposal without treatment. The process offers the advantage that it does not produce secondary wastes, and low-level wastes are limited to the pure uranium product and to discarded process equipment components.

The electrometallurgical treatment process has been demonstrated at pilot scale with unirradiated fuels containing representative concentrations of transuranic elements and non-radioactive fission product elements. Extensive testing has been done with waste forms arising from the treatment of these unirradiated fuels. The process has been demonstrated

with unirradiated N-Reactor fuel and with simulated oxide spent fuel. Process equipment has been installed in the Fuel Conditioning Facility at Argonne National Laboratory-West at the Idaho National Engineering Laboratory, and the facility is ready for operation with spent fuels from the EBR-II reactor. Pending approval to proceed with hot operations, the equipment is being operated with depleted uranium to verify operating procedures and aid in operator training. Demonstrations with other irradiated fuel types are planned.

Glass Material Oxidation and Dissolution System (GMODS)

ORNL recently invented the GMODS for the direct conversion of SNF to borosilicate HLW glass. The process is generically applicable to many SNF types and would address the issues of multiple waste forms, SNF instability, and classified SNF configuration related to disposal. GMODS directly a) converts metals, ceramics, and amorphous solids to borosilicate glass, b) oxidizes organics with the residue converting to glass and c) converts chlorides to a clean secondary NaCl stream and glass.

GMODS is designed to produce a HLW form acceptable to the repository within the acceptance criteria as defined by the two repository mainline waste forms: light water reactor (LWR) SNF and HLW borosilicate glass. The process would produce a borosilicate HLW glass log similar in size to those from other DOE vitrification plants. No new waste form must be developed. As with other processes, it would enable the addition of depleted uranium to HEU SNF to resolve repository criticality concerns. This includes the option of processing high-enriched SNF with Hanford N and other low-enriched SNF to adequately isotopically dilute the 235U so the uranium fissile-to-fertile ratio in the glass is similar to LWR SNF and hence acceptable to the repository. The option also exists to process high-burnup SNF with low-burnup SNF to ensure that repository waste package heat load limits are not exceeded. GMODS would not separate any constituents and thus would not contribute to proliferation concerns.

The basic concept of GMODS is to directly add unprocessed SNF to a cold-wall, induction-heated glass melter with molten lead borate glass. Oxide and amorphous components of SNF directly dissolve into the glass. Metal and organic components do not dissolve in normal glasses. The lead oxide in the lead borate glass acts as a sacrificial oxide and oxidizes in situ carbon to carbon dioxide (CO_2) and metals to metal oxides. The CO_2 exits the melter as a gas. The metal oxides (uranium, cladding, etc.) dissolve into the glass. The lead reaction product of these chemical reactions (a) sinks to the bottom of the melter, (b) is removed from the melter, (c) is oxidized back to lead oxide, (d) the lead oxide is fed back to the melter, and (e) the lead oxide oxidizes more metals and organics in the melter.

After dissolution of the SNF, additional glass additives (SiO_2, etc.) are added to improve the glass quality before the HLW molten glass is poured into containers. If desired, the lead oxide can be removed from the final glass solution before solidification. This is done by adding carbon, which reduces the lead oxide to lead metal and produces gaseous CO_2. The lead metal separates from the glass and can be converted to lead oxide to process the next batch of SNF.

When chloride containing materials are fed to the melter, the non-chloride components enter the glass and the chloride reacts with the lead borate glass to produce volatile lead chloride. The lead chloride reacts with sodium hydroxide in the aqueous off-gas scrubber to yield soluble

sodium chloride salt and insoluble lead hydroxide. The lead hydroxide is recycled to the melter where it decomposes to lead oxide and steam. The sodium chloride stream is purified by ion exchange and discharged as a clean chemical waste. This potential capability allows GMODS to treat chloride containing SNF.

Laboratory demonstrations have been made of the direct conversion of stainless steel, Zircaloy-2, aluminum, uranium, cerium, and other metals to glass. Various oxides such as uranium oxide and zirconium oxide have been directly dissolved into the glass. Graphite has been oxidized. Chloride feeds have been treated with the chlorides separated to a secondary sodium chloride stream.

Plasma Arc Process

A plasma arc process for converting SNF (i.e., metal, oxide, carbide, sodium-bearing, aluminum, and zirclad) and the associated wastes (sludge, soil, metal fragments, metal containers, and concrete grits) into vitreous ceramic final waste forms is being developed at Pacific Northwest National Laboratory (PNNL). This is a direct and single-step process applicable to both containerized fuels and other types and forms of wastes. There is minimum pretreatment, sorting, and size-reduction needed for the treatment, resulting in reduced radiation exposure and treatment cost.

This process enables the addition of sludge, soil, metal containers and fragments, and concrete grits to HEU waste to satisfy the criticality concerns. The energy released in oxidizing the uranium metal fuel is used in vitrification. Minimal or no secondary wastes are generated since the particulate material removed in off-gas cleanup is recycled to the furnace for immobilization in the waste form. The fissile elements can be immobilized into a matrix that can be recovered only with difficulty or can also be immobilized into a matrix that is almost impossible for recovery, depending on the formulations of vitreous ceramics. The vitreous ceramics are durable and suitable for permanent disposal as an HLW form, or for interim storage, until a geologic repository becomes available.

Nuclear fuels in canisters or entire fuel elements are loaded directly into the furnace. A second feeder simultaneously add contaminated soils, sludges and concrete grits. The rate of the oxidation of the metal fuel charged is controlled by controlling the rate of oxygen addition, the oxygen lance standoff from the bath, and the plasma torch standoff. The waste form final compositions are controlled by the optimized combination of the nuclear fuels, sludge, soil, concretes, and minimum additives. Vitreous ceramics are formulated in such a way that they contain stable and low-solubility (in water) crystalline phases embedded in a network-former rich glass matrix. These crystals a) incorporate a large amount of metal ions, such as iron oxide, to accommodate the large iron content of the sludge; b) incorporate large contents of uranium, plutonium, and other fission products in the lattice structure; c) utilize little or no network-forming elements, such as silicon and aluminum, so these latter elements are enriched in the residue glassy matrix; and d) tightly bind to the glassy matrix so the physical integrity and mechanical strength of the waste form can be maintained.

The plasma process has been demonstrated by directly oxidizing 55-gallon drums of non-nuclear wastes into durable vitreous ceramics. The vitreous ceramics were produced on a bench scale (6 kg per hour) and pilot-scale (150 kg per hour) with metal contents of over 80 wt% and waste loading up

to 100%. The chemical durabilities of the vitreous ceramics were shown to be more durable than the most durable high-level nuclear waste glasses through extensive sets of dynamic (flow-through), static (PCT), and vapor hydration tests.

This technology, bases on established commercial equipment, was demonstrated for simulated wastes with high metal contents, and can be further developed and implemented to process actual nuclear fuels and wastes within 4 years. The development team for this technology consists of PNL (leader), Westinghouse Hanford Company, Retech, Inc. (A Lockheed Martin Subsidiary), Science Applications International Corporation, MSE, Inc., and the University of New Mexico.

Chloride Volatility Process

A chloride volatility concept is being investigated at the INEL as an advanced process for stabilizing SNF. This new technology would require 15 years to be developed as an implemented plant process. The distant schedule for repository disposal would allow the luxury of pursuing such an advanced technology that may not otherwise be considered. The concept offers the significant advantage of a single, compact process that is applicable to most fuels. It would minimize secondary wastes, segregate major nonradioactive constituents from the HLW for volume reduction, segregate fissile uranium from the HLW to resolve potential criticality concerns, and produce a common waste form glass or glass-ceramic.

The process is based on completely volatilizing the fuel element and separating the gaseous constituents. It consists of reacting the fuel with chlorine gas at high temperature (greater than 1,200oC), which causes all the fuel constituents to form volatile chlorides. The gases are then separated by molten salt scrubbing and fractional condensation. There are four major unit operations: (a) chlorination and volatilization of all the fuel components at 1,200oC, (b) removal of fission product, transuranic, and any nickel and chromium chlorides in a $ZnCl_2$ scrubber at 400oC, (c) three condensers for removing, by fractional condensation at temperatures ranging from 164oC to 2oC, $ZrCl_4$, $FeCl_3$, $AlCl_3$, UCl_6, $SnCl_4$, and I_2 vapors that pass through the scrubber, and (d) regeneration of the transferred spent molten salt by vacuum distillation to recover $ZnCl_2$ and $ZnCl_2$-soluble $ZrCl_4$ for recycle, leaving the fission product-, transuranic-, nickel-, and chromium-chlorides as residue that would be converted to oxides or fluorides for vitrification. Argon carrier gas and unreacted chlorine gas would be recycled, the Cl_2 content adjusted, and the stream split and passed through the unit operations in a continuous closed loop. Periodic shutdown of the coupled unit operations would occur for batch removal of fission product xenon and krypton gases from the carrier gas (such as by cryogenic distillation), batch transfer of the molten salt to the molten salt regenerator, and batch removal of nonradioactive constituents and uranium from the condensers.

The small quantity of fission-product/transuranic-product HLW would be converted into a waste form for repository disposal. The conversion steps to a glass or glass-ceramic form could involve fluorination and then melting with glass frit additives, or conversion to oxides by heating at 1,000oC with boric acid. Solvent extraction processes, such as transuranic extraction (TRUEX), that was developed at Argonne National Laboratory, could be applied to the fission-product and transuranic-product chlorides dissolved in 6 M HCl if their separation were desired or needed prior to immobilization.

In the chlorination step, the rate of reaction is controlled by the feed rate of Cl_2 and temperature is controlled by appropriate blending of Ar gas with Cl_2. An oxygen scavenger, such as carbon monoxide, is added as needed to prevent formation of oxychlorides when oxides are present. A CO_2 absorption bed in the off-gas system collects the carbon dioxide that is formed. Zinc chloride was selected for the scrubber medium because its low melting point and favorable vapor pressure permit its use to scrub the chlorinator off-gas at a low temperature, while being sufficiently volatile at 725oC to allow evaporative separation from the radioactive waste chlorides for subsequent recycle. For fuels with Zircaloy, some $ZrCl_4$ will dissolve in the $ZnCl_2$ until saturation is reached, after which $ZrCl_4$ will pass through. The dissolved $ZrCl_4$ can be recycled with the molten salt without further complicating the flowsheet.

Other Treatment Concepts

A number of other treatment concepts have been proposed and are being evaluated to determine their validity for further conceptual development. These include concepts of mechanical volume compaction, direct disposal in small poisoned canisters, and Can-in-canister disposal. These technologies are being explored for use with the aluminum based fuel at Savannah River, but may be applicable to fuels at other locations in the DOE Complex.

Mechanical volume compaction techniques, such as Chop and Dilute, are relatively simple methods of isotopic dilution of HEU in Al-clad fuel. The primary advantages are criticality control using either depleted uranium or poisons and the possibility of a reduced number of disposal canisters for repository disposal. Chop and Dilute involves chopping the AL-HEU SNF elements into small pieces and mixing them with similar pieces of depleted uranium aluminum alloy. To ensure against criticality in the repository, the resulting enrichment level of the mixture may have to be an equivalent low 235U enrichment. This option has the significant disadvantage that it requires a larger number of waste canisters inversely proportional to the enrichment levels allowed by the repository requirements. If the requirements for criticality control result in 1-2% 235U enrichment requirements, then the high level waste volumes will be more than produced by reprocessing (10 to 50 times) and may not produce a repository-durable waste form. Acceptability of the chop-dilute form is unlikely.

Direct disposal in small poisoned canisters is a variation on the direct disposal method proposed for commercial fuel. Because some of the DOE fuel is highly enriched, smaller canisters may be used to reduce the probability of a criticality in the repository. This methodology would reduce the 235U content in a particular area of the repository. A variation of this technique is to increase the size of the canister by including nuclear poisons with the HEU. This methodology may not be adequate in preventing a criticality in the future if it is assumed that the poisons migrate away from the HEU over time.

The Can-in-canister option, which borrows its name from the plutonium immobilization option, would involve placing already canned HEU SNF elements into DWPF canisters and then filling the canisters with HLW glass. This is may be neither a practical nor cost effective option because of the excessively large number of canisters that would be required to dispose of the SNF. Criticality control combined with the space occupied by HLW glass would significantly limit the quantity of 235U per canister. (At three elements per canister, a total of 12,000

canisters would be required). The HLW glass may be significantly cracked and would not provide a barrier against water. Qualification of this waste form for the proposed repository is problematic.

PLANS FOR THE FUTURE

There are several issues that will need to be resolved in order to solidify plans for the future, not the least of which being the issue previously discussed in the Institutional Issues section of this paper regarding "treatment/processing" for disposal versus "reprocessing" for plutonium and highly enriched uranium recovery. At the present time, as discussed previously, there are six treatment/conditioning technologies at various stages of implementation that can potentially be utilized in preparing the multitude of different (>150) DOE SNF types for disposal in the geologic repository or in a Monitored Retrievable Storage facility. The canyons at Savannah River are ready to process any of the aluminum SNF that the decision makers deem appropriate. The research and development of the Electrometallurgical Treatment technology is currently fully funded by the Office of Environmental Management and will continue to receive the necessary funding unless some unforeseen event or change occurs in the future. The funding for the other three technologies (GMODS, Plasma Arc and Chloride Volatility) is currently at a very low level, but these technologies have not been totally discounted. Changes may occur to bring these technologies to the forefront as replacements for currently funded efforts, or to provide viable alternatives so that a degree of flexibility can be retained for future decisions.

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THE REGULATORY APPROACH FOR SPENT FUEL STORAGE AND CONDITIONING FACILITIES - THE HANFORD EXAMPLE

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ABSTRACT

Hearings held before the House Subcommittee on Energy and Mineral Resources in March 1994, requested that officials of federal agencies and other experts explore options for providing regulatory oversight of the U.S. Department of Energy (DOE) facilities and operations. On January 25, 1995, the DOE, supported by the White House Office of Environmental Quality and the Office of Management and Budget, formally initiated an Advisory Committee on External Regulation of DOE Nuclear Safety. In concert with this initiative and public opinion, the DOE Richland Operations Office (DOE-RL) has initiated the K Basin Spent Nuclear Fuel Project - Regulatory Policy (1). The DOE has established a program to move the spent nuclear fuel presently stored in the K Basins to a new storage facility located in the 200 East Area of the Hanford Site. New facilities will be designed and constructed for safe conditioning and interim storage of the fuel. In implementing this Policy, DOE endeavors to achieve in these new facilities "nuclear safety equivalency" to comparable U.S. Nuclear Regulatory Commission (NRC)-licensed facilities. The DOE has established this Policy to take a proactive approach to better align its facilities to the requirements of the NRC, anticipating the future possibility of external regulation. The Policy, supplemented by other DOE rules and directives, form the foundation of an enhanced regulatory program that will be implemented through the DOE K Basin Spent Nuclear Fuel Project (the Project).

INTRODUCTION

The Advisory Committee on External Regulation of DOE Nuclear Safety completed its initial status report in August 1995 (2). The Committee's preliminary findings report widespread dissatisfaction within the DOE Complex with current DOE implementation of nuclear safety. The major concern was excessive redundancy of oversight functions and lack of responsibility delegated to the operating level. The findings conclude that there exists clear support in the DOE community for external regulation with a corresponding reduction in internal regulation. In its report, the Committee emphasizes that regardless as to whether the DOE moves to some form of external regulation, an effective internal system for safety management is presently needed. The preliminary report provides some indication that new DOE facilities would likely be the focus of changes in regulatory oversight, leaving existing facilities, at least temporarily, under existing regulatory arrangements.

In an effort to improve the quality of nuclear safety of new facilities and to align those new facilities with the possibility of future external oversight, the DOE and its contractor, Westinghouse Hanford Company (WHC), have established an enhanced nuclear safety program for the KBasins Spent Nuclear Fuel Project. Central to the program is DOE's Policy to achieve NRC nuclear safety equivalency in the design and construction of the Project's new facilities. This paper describes the Project's regulatory program, shares lessons learned and experiences gained in establishing the program, and provides the current status of the program's implementation. The paper's intent is to make known the existence of the Policy and related documentation developed through its implementation as information to be shared within the DOE Complex.

SPENT NUCLEAR FUEL PROJECT

The Spent Nuclear Fuel Project was established to address resolution of the safety and environmental concerns associated with the deteriorating spent nuclear fuel stored in the K Basins. In February of 1995, the DOE approved the Project's "Path Forward" recommendation for moving the spent fuel in the K Basins away from the Columbia River into the Canister Storage Building (CSB) to be constructed in the 200 East Area of the Hanford Site (3). The initial Path Forward recommendation, was refined following a detailed process alternatives evaluation in the Spent Nuclear Fuel Project Integrated Process Strategy (4). Detailed project plans developed after the Integrated Process Strategy recommendation, plan for the fuel to be removed from the K Basins, packaged in multi-canister overpacks (MCOs), drained and vacuum dried, and shipped to the CSB for interim storage. Additionally, a Conditioning Facility is planned to be constructed near the CSB to further process fuel that requires additional drying to remove chemically bound water and uranium hydride prior to interim storage in the CSB. Figure 1 illustrates the Integrated Process Strategy for the K Basins spent nuclear fuel.

Fig. 1

ENHANCED NUCLEAR SAFETY PROGRAM

Early in the formation of the Project the concept of developing an enhanced regulatory strategy by incorporating NRC nuclear safety equivalency into the design and construction of new facilities was considered. Agreement upon what constituted NRC nuclear safety equivalency was widely debated within DOE and WHC. Many different views existed with regard to organizational authorities, oversight roles, requirements, and the definition and scope of NRC equivalency. To resolve these issues, the DOE established the Policy for implementation of NRC nuclear safety equivalency. The Policy received approval by DOE Headquarters (HQ), as well as by DOE Richland (RL) and the WHC Spent Nuclear Fuel Project.

The Policy defines general terms, identifies key roles and responsibilities, and outlines the overall approach to be followed in implementing NRC nuclear safety equivalency for the Project. The Policy is limited to matters of nuclear safety; and does not apply to environmental, OSHA, chemical accident safety, and other non-nuclear safety issues, as these are covered elsewhere by DOE orders and statutory requirements.

The Policy states that the Project will achieve NRC nuclear safety equivalency by "...applying technical requirements based on those applied by the NRC to comparable licensed facilities and by adopting appropriate features of the NRC licensing process, in addition to applicable DOE orders and requirements."

The Policy was developed for the following purposes:

To achieve a set of requirements that are technically defensible and cost-effective.

To achieve in the design and construction of new Project facilities, a level of nuclear safety comparable to that of NRC licensed commercial facilities.

To enhance public understanding and confidence in the safety of the new facilities by following an enhanced regulatory strategy.

The Policy defines nuclear safety equivalency for the Project as:

Applying technical requirements which meet the nuclear safety objectives of the NRC regulations for fuel conditioning and storage

facilities. These include requirements regarding radiation exposure limits, safety analysis, design, and construction.

Applying administrative requirements which meet the objectives of the major elements of the NRC licensing process. These include formally documented design and safety analyses, independent technical review, and opportunity for public involvement.

Technical requirements, in the context of the Policy, are considered to be the design and construction measures (as opposed to pre-operational or operational measures) that are mandated by the NRC regulations. The DOE orders and directives form the basis of all aspects of the Project. The DOE design and construction requirements are supplemented by the additional NRC requirements of Title 10, Code of Federal Regulations (CFR), Parts 0-199, in order to establish NRC nuclear safety equivalency. NRC guidance and precedents, illustrative of implementation of the regulations, were considered optional rather than mandatory. However, such guidance was reviewed and incorporated, where applicable, during the Project's implementation of the Policy.

The Policy establishes nuclear safety oversight for the Project and defines organizational roles and responsibilities. The relationships between Project oversight organizations is illustrated in Fig. 2. Nuclear safety oversight from DOE-HQ is provided by the offices of Environmental Management (EM) and Environmental, Safety, and Health (EH). To facilitate accelerated safety authorization schedules, the Policy established a Regulatory Requirements Team (RRT), comprised of DOE-HQ, DOE-RL, and WHC to review and approve selected regulatory requirements and to facilitate the safety review and authorization process for the Project's facilities. The RRT is chartered to assist the contractor with the task of evaluating NRC regulations and guidance to determine the supplemental requirements that would apply to the Project. Other specific responsibilities of the RRT include; review and approval of requirements, review of new facility safety analysis reports, and input to the preparation of DOE safety evaluation reports.

The Policy also established an independent review process to ensure that the requirements selected as the basis for Project's activities provide an acceptable level of worker safety, public health and safety, and protection of the environment. The process utilizes reviews by both DOE-EH staff and an Independent Review Panel (IRP). The IRP is comprised of nationally recognized technical experts, not associated with the Project, that provide high-level external oversight of the requirements selection process. Specific responsibilities of the IRP include: overview of the requirements selection process, review of safety analysis documentation, and verification that Project facilities meet DOE's Policy for nuclear safety equivalency.

Fig. 2

PROGRAM INFRASTRUCTURE

Figure 3 illustrates the Project's regulatory document hierarchy that forms the key elements that guide implementation of the Project's enhanced nuclear safety program. DOE rules and directives form the foundation of requirements for the Project. These requirements are supplemented by DOE's Policy for NRC nuclear safety equivalency. From this point, the Project has developed documentation along two paths. One path establishes the management infrastructure of the program (organization charters, program baselines, schedules, etc.). The other

path provides technical direction for further definition and implementation of program requirements.

Fig. 3

The Project Management Plan establishes the management basis for the Project (5). The plan defines the Project boundaries and identifies the plans, organizations, and management systems that are to be utilized on the Project. The plan specifies the development of project and sub-project management plans, like the Project's Regulatory Program Plan and Public Involvement Plan, and cross-project topical plans and reports that focus on technical plans and activities, like the Regulatory Strategy and Safety Management Plan.

The Regulatory Program Plan is the management plan for the Project's nuclear safety program (6). This plan establishes and describes plans for implementing and complying with nuclear safety regulatory requirements applicable to the Project's facilities and activities, including the key objectives and requirements identified in the Policy for NRC nuclear safety equivalency. The document discusses Project requirements for planning, budgeting, and scheduling specific tasks such as preparation, review and approval of Safety Analysis Reports (SARs); establishing a regulatory requirements database and commitment tracking system; and managing regulatory compliance.

In parallel with the establishment of the Policy, the Regulatory Strategy (7) was developed to obtain agreement with DOE on an approach for identifying nuclear safety requirements, performing appropriate safety analyses, preparing safety documentation, and obtaining the necessary approvals and authorizations for each of the Project's major facilities. For the Project's new facilities, DOE Order 5480.23, Nuclear Safety Analysis Reports, and its implementing standard, DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports, will be used to prepare safety analysis documentation. The DOE order and standard will be supplemented by the format and contents guidance of Title 10, CFR, Part 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-level Radioactive Waste and NRC Regulatory Guide 3.48, Standard Format and Content for Safety Analysis Reports for an Independent Spent Fuel Storage Installation (Dry Storage), in order to help demonstrate the NRC nuclear safety equivalency objective. Parallel to the development of the Regulatory Strategy was the creation of official charters for the RRT and IRP to focus their efforts and define their responsibilities and membership.

The Safety Management Plan (8) further defines the requirements identification process and establishes the plan to achieve compliance with requirements through development of appropriate safety analyses and design documentation. The document details plans and schedules for preparation, review, and approval of the safety analysis documentation. The plan also establishes the approach for integrating the safety analysis, safety documentation, and independent safety reviews with the Project's design, construction, and startup activities. The plan clearly defines organizational responsibilities within the Project for safety documentation and independent review requirements.

NRC EQUIVALENCY PROCESS

Consistent with the DOE's Policy, a comprehensive evaluation of Title 10, CFR, Parts 0-199, and potentially relevant NRC guidance, including NRC Regulatory Guides, NRC NUREG and SECY documents, Standard Review Plans,

Inspection and Enforcement Notices and Bulletins, was performed against applicable DOE requirements. An evaluation of Title 10, CFR, Part 72, less subparts A,C,J, and K, against applicable DOE requirements performed by Scientech, Inc., under contract to DOE-HQ, was also included in the evaluation. A review was also performed of a current license renewal application under Title 10, CFR, Part 70, Seimans Power Corporation, Application for Renewal of Special Material License No. SNM-1227 (NRC Docket No. 70-1257), at the direction of the IRP to identify any "revealed regulations" (i.e., NRC requirements imposed during the licensing process that are not called out in the regulations). The results of this review are provided in the Spent Nuclear Fuel Project Path Forward - Nuclear Safety Equivalency to Comparable NRC-Licensed Facilities report (9).

Equivalency was established for the most part by the DOE regulations and orders. In some instances WHC procedures and instructions were used to establish equivalency for the Project. A listing of additional NRC requirements that were deemed applicable to the Project were documented in the following reports Spent Nuclear Fuel Project Path Forward - Additional NRC Requirements (10) and Multi-Canister Overpack - Additional NRC Requirements (11). Completion of this process was a significant step toward achieving NRC equivalency in the design and construction of the Project's Path Forward facilities.

ACCOMPLISHMENTS

The Project has nearly completed development of the documents described above that support implementation of its regulatory program. The NRC nuclear safety equivalency policy was approved by DOE in August 1995. The RRT was established in April 1995. The RRT tentatively meets on a weekly basis via video conference. Video conferencing provides an economical means of expediting, facilitating, and coordinating the safety documentation within several organizations which may be separated by several thousand miles. The IRP's membership was defined in June 1995, and the team has held three meetings, in June, September, and December of 1995. A comparison of NRC to DOE requirements has been completed and is in the final stages of review and approval. The requirements assembled from this process are being incorporated into the project design basis through the development of Functions and Requirements documents and performance specifications for the Project's facilities. The Project's Regulatory Strategy and Program Plan have been established and approved. The Safety Management Plan is currently being drafted, and preparation of new facility safety analyses are underway.

CONCLUSION

The DOE is implementing an enhanced regulatory program for Hanford's KBasins Spent Nuclear Fuel Project to provide an increased level of nuclear safety in the design and construction of planned new facilities. The basis for the approach to provide an enhanced regulatory program is DOE's Policy for achieving nuclear safety equivalency to comparable NRC-licensed commercial facilities. By establishing the Project's nuclear safety program and completing a review of regulations to establish NRC nuclear safety equivalency, DOE has taken the first steps toward achieving greater nuclear safety in the design and construction of the Project's new facilities. Implementation of the Policy for NRC nuclear safety equivalency will better align these facilities for the possibility of future external regulation by the NRC. Documents described in this

paper are available for use by the DOE Complex in the development of similar enhanced regulatory programs related to nuclear safety.

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PREPARING DOE SPENT NUCLEAR FUEL FOR ULTIMATE DISPOSITION: MEETING REQUIREMENTS AND MANAGING TECHNICAL RISK

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ABSTRACT

The Department of Energy (DOE) Office of Spent Fuel Management, within the Office of Environmental Management, has undertaken a program to ensure safe existing storage, provide for interim storage, and to prepare for ultimate disposition of DOE spent nuclear fuel (SNF). In the past, most DOE SNF was chemically processed in order to recover plutonium or uranium for the nuclear weapons program. However, with the phaseout of chemical processing throughout the DOE complex, most of the remaining SNF must now be managed until its ultimate disposition, rather than for the few years of storage originally envisioned.

This paper focuses on the challenges and issues surrounding the development of disposal requirements for DOE SNF. A critical element of successfully preparing DOE SNF for ultimate disposition is understanding the requirements that must be met for such disposition. Geologic disposal is the primary option under consideration for the ultimate disposition of most DOE SNF. The Office of Civilian Radioactive Waste Management (RW) is pursuing the development of a geologic repository for the disposal of SNF and high-level waste (HLW). The Office of Environmental Management (EM) has had intensive interactions with RW which will help ensure preparation of DOE SNF for geologic disposal consistent with RW program constraints and requirements. Interactions between EM and RW will also help balance the risk, effort and costs imposed on each program.

INTRODUCTION

Developing requirements for the geologic disposal of DOE SNF is a relatively new initiative. Prior to 1992, DOE had generally planned to manage its SNF by reprocessing it and vitrifying the resulting HLW for disposal in a geologic repository. In 1992, DOE decided to phase-out reprocessing of its spent fuel and began to consider other options, including the possibility of direct disposal in a geologic repository. Interactions between EM and RW resulted in the identification of a planning base for the geologic disposal of DOE SNF. Under this planning base DOE would dispose of all of its spent fuel, not otherwise dispositioned, in the first repository, subject to a number of constraints such as meeting licensing and disposal fee requirements. Recent interactions between EM and RW have now also established the need for disposal requirements specifically for DOE SNF.

Formal interactions between EM and RW began in August 1994 with the first meeting of the DOE SNF Steering Group. In July 1994, the Director of RW had recommended formation of an EM/RW steering group, and identified a number of key issues a steering group should focus on, including repository schedule impacts, criticality control, canisterization, potential regulatory issues regarding the Resource Conservation and Recovery Act, Safeguards and Accounting issues, and establishment of appropriate Interagency Agreements/Disposal Fees. EM concurred on the need for this Steering Group and further recommended that resolution of these issues be used to help establish disposal requirements for DOE SNF. Later in 1994, RW began to examine existing acceptance requirements in order to determine their applicability to DOE SNF and develop disposal requirements specifically for DOE SNF. A document, "Preliminary Requirements for the Disposition of DOE SNF in a Deep Geologic Repository," hereafter referred to as the Preliminary Requirements, was issued in December 1995. This document, developed under a quality assurance program, would be used by EM to plan quality affecting activities for the preparation of DOE SNF for geologic disposal.

DISCUSSION

RW began developing disposal requirements for DOE SNF by examining existing waste acceptance requirements for commercial SNF and HLW. In many cases these existing waste acceptance requirements were taken directly from the Code of Federal Regulations (CFR), especially 10 CFR 60, "Disposal of High-Level Radioactive Wastes in Geologic Repositories," and 10 CFR 961, "Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste." These requirements were, therefore, also directly applicable to DOE SNF and provided the essential components of the disposal requirements for DOE SNF. Other requirements for HLW and

commercial SNF were derived from information gathered by the RW program to date (e.g., site characterization activities, licensing strategies, interpretation of regulations, production specifications for HLW). However, these requirements often need to be verified because they represent the best knowledge to date and could change. In most cases these requirements were potentially flexible and some latitude may be acceptable as long as regulatory compliance is maintained.

Establishing disposal requirements for DOE SNF based on precedents set by commercial SNF and HLW was sometimes difficult. DOE SNF often does not have the same characteristics as commercial SNF. Different sizes, configurations, enrichments, fuel matrices, fuel claddings, burn-ups, etc., all represented potential complications in the establishment of requirements. These characteristics also represent potential complications in preparing DOE SNF to comply with the requirements. In some cases, standardization requirements developed in order to minimize facility' costs, are difficult to apply to DOE SNF with dissimilar sizes and configurations.

DOE SNF also has characteristics different from those of vitrified HLW. The production specifications for vitrified high-level waste are the most developed technical specifications for any material to be disposed of in a repository. Thus these specifications were initially viewed as potentially useful when developing disposal requirements for DOE SNF. However, it was recognized early in the process that production specifications for a manufactured waste form were not always appropriate for DOE SNF because it is constrained by its existing configuration and characteristics (although treatment technologies for some types of DOE SNF may produce a manufactured waste form). Some tests or verifications performed for vitrified HLW may not be necessary or appropriate for DOE SNF. Thus, the HLW production specifications were only of limited value for development of disposal requirements for DOE SNF.

Development of disposal requirements also is difficult because the bases for some existing waste acceptance requirements are still to be verified. This is due in part to the fact that RW is still receiving, and will continue to receive for a number of years, information from site characterization activities at the candidate repository site. In addition, the RW disposal facility has not yet been designed, built, or licensed. Design changes necessitated by future developments or interactions with regulators cannot be accurately predicted at this point. For example, the thermal design bases for the repository have not been finalized and may result in significant changes to the thermal limits placed on waste packages at the repository. Also, RW's implementation of the Multi-Purpose Canister (MPC) concept is being reconsidered and may cause changes to configuration requirements for the repository. The characteristics of DOE SNF, once repository designers are fully cognizant of them, may also cause modifications in the design of the repository which may require changes to the disposal requirements.

Development of other requirements for DOE SNF was problematic in that it was unclear how to apply systems level requirements to individual DOE SNF forms and/or packages. For example, 10 CFR 60.131(b)(7), found in the section of 10 CFR 60 that outlines design criteria for the geologic repository operations area, reads: "All systems for . . . isolation of radioactive waste shall be designed to ensure that a nuclear criticality accident is not possible unless two unlikely and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality

safety." Although this requirement applies to systems for the isolation of radioactive waste and is not applied directly to the DOE SNF forms or packages themselves, the design and performance of DOE SNF forms and/or packages are critical components of the system being evaluated for compliance with this requirement. It appears appropriate, therefore, to place some form of the requirement on the DOE SNF form and/or package, even though the requirement is mandated at the systems level.

Similarly, the requirements of 10 CFR 60.135, "Criteria for the waste package and its components," are addressed primarily in the context of not compromising "the function of the waste packages or the performance of the underground facility or the geologic setting." The application of these requirements is more complex as they are often applied to the waste package itself, which is in close proximity to the DOE SNF. The requirements of 10 CFR 60.135 also in some cases refer directly to the contents of the waste package and thus have a direct impact on DOE SNF. Still, the impact of the performance of DOE SNF forms and/or packages on the performance of the system must also be evaluated, which makes for a potentially complicated allocation of performance requirements. Future interactions between EM and RW must address the appropriate performance allocation required for DOE SNF.

The draft Preliminary Requirements were completed and issued for review in June 1995. Representatives from the Office of Naval Reactors, the Idaho National Engineering Laboratory, the Savannah River Site, and the Hanford site participated in the review for EM. RW also had a number of participants in the review. Final comments on the document were submitted in July 1995. Comment review and resolution were completed in November 1995 and the document was issued in December 1995.

As part of the comment resolution, EM and RW reached agreement on the purpose and use of the Preliminary Requirements. There were concerns that, although this document was prepared under Quality Assurance procedures, the Preliminary Requirements were not yet incorporated into the official RW technical baseline. Part of the EM and RW agreement established that disposal requirements for DOE SNF will be incorporated into the RW technical baseline in early spring of 1996. Following issuance of the disposal requirements, draft documents demonstrating DOE SNF program compliance will likely be issued within a year, although the exact details of these documents are yet to be determined.

RESULTS

Development of Preliminary Requirements for the geologic disposal of DOE SNF, and the related interactions between EM and RW, resulted in a number of lessons and insights regarding the application of, and compliance with, disposal requirements for DOE SNF.

The RW program baseline has been established based on the characteristics of commercial SNF and, to a lesser degree, vitrified HLW. This program baseline optimizes repository design and other system features around SNF that is low-enriched, in oxide form, having a relatively small number of fuel types, and a relatively large amount of data with a Quality Assurance pedigree. Given the very different SNF types represented in the DOE inventory, a whole new suite of issues has emerged. There are new issues regarding compliance with disposal requirements, interpretation of disposal requirements, and other new unforeseen issues based on RW's experience with commercial SNF.

Probably the chief challenge regarding the evaluation of DOE SNF for geologic disposal is the large number and variety of DOE SNF types. Using

some simplifications, DOE SNF may be considered to fall into one of approximately a dozen fuel types. However, if these simplifications are removed and subcategories of fuels are included, the number of DOE SNF types may exceed 150. In addition, if the primary characteristics of SNF are considered, including enrichment, fuel matrix, fuel cladding, burn-up, size and configuration, DOE SNF spans the complete spectrum. Enrichments vary from virtually depleted uranium to enrichments in the high ninety percentiles. Fuel matrices include a number of oxide forms, uranium-Aluminum alloys, carbides, uranium metal, and many others. Fuel claddings include commercial Zircaloys, stainless steels, and aluminum alloys. Burn-up of DOE SNF ranges from almost fresh fuel to deeply burned fuel. There are also a very large number of sizes and configurations of DOE SNF.

Beyond the challenges posed by the number and variety of DOE SNF types, there is the challenge of gathering reliable data, with the appropriate pedigree, for all the DOE SNF types in order to assess compliance with disposal requirements. Data are needed to accurately assess not only the existing physical characteristics of the DOE SNF, but also to assess the long-term performance of DOE SNF in the repository environment.

A large amount of data exists for DOE SNF. However, unlike commercial SNF, most of which was produced and utilized under Nuclear Regulatory Commission (NRC) approved quality assurance programs, most DOE SNF was not maintained under such programs. Some DOE SNF types, such as the ATR and the TRIGA, have a strong pedigree with adequate records traceability. Furthermore, some DOE SNF has safeguards and accountability data which may provide an extensive record of its characteristics. However, for some DOE SNF, only limited storage records exist, making it difficult to assay its physical characteristics. In some cases these existing records may be inadequate to support the data needs of RW. In other cases, the traceability of the data may call into question its adequacy.

The costs for using even existing data to support disposal of DOE SNF could be quite large. One cost estimate based on data needed to support the License Application Annotated Outline is a minimum of five million dollars for DOE SNF types with good records. The costs of developing information for disposal of some types of DOE SNF, which do not have an adequate data base, could exceed ten million dollars.

The high enrichment of DOE SNF also poses challenges for the EM and RW programs. Maintenance of criticality control for highly enriched SNF is more complicated than criticality control for commercial SNF, which is almost exclusively low enriched. When considered over the long time periods being used to predict repository performance, this demonstration of criticality control becomes even more challenging. The high-enrichment of some DOE SNF may also make it more attractive for diversion and thus may require the introduction of more stringent safeguards and accounting requirements. Furthermore, much of the highly-enriched DOE SNF has relatively low radiological and thermal output which may mean that it will not have some of the self-protecting characteristics of commercial SNF, which has relatively high radiological and thermal output.

Degradation of some portions of the DOE SNF inventory poses a significant technical challenge. For up to 5-10 percent of the DOE SNF inventory, degradation, and/or other undesirable characteristics, introduce substantial uncertainty in preparation for geologic disposal. This has raised the possibility that some DOE SNF may have to be treated prior to disposal. In fact, recent decisions by the Department regarding the DOE

SNF inventory at the Savannah River Site have resulted in a plan to process those DOE SNF types that threaten the environment, safety and health of workers and the public.

Thermal limits anticipated for the repository also pose a challenge to a portion of the DOE SNF inventory. Currently, RW anticipates pursuing a repository design with a thermal limit that may be as high as 400 C for a certain period. Temperatures this high would likely have a negative effect on aluminum-clad SNF, a common fuel type in the DOE SNF inventory. Aluminum cladding is normally subject to deformation at temperatures above 180 degrees C. EM and RW have begun to assess potential solutions to this challenge.

CONCLUSION

The development of disposal requirements for DOE SNF has been an extensive and difficult effort. The existing RW disposal requirements were not designed with DOE SNF in mind, but were created for a repository containing HLW and commercial SNF. In addition, regulatory requirements for HLW and commercial SNF may create compliance difficulties for some types of DOE SNF.

As this program progresses, EM and RW face challenges. Some of these challenges naturally draw the two programs to work together. However, for other challenges there are conflicting objectives which drive each program. It will take leadership and enlightened cooperation on the part of both programs to advance the best solutions for the Department.

Management of programmatic uncertainty and risk are an important part of the cooperative efforts of EM and RW. RW's licensing activities are currently focused on the licensing of commercial SNF for disposal in a geologic repository. Some DOE SNF types pose new issues for the licensing of a repository. Resolution of these issues in a very conservative manner could impose very large costs on the DOE SNF management program, especially when certain packaging and conditioning alternatives are considered. However, less conservative approaches could potentially jeopardize the success licensing.

In summary, EM and RW must work carefully together to develop a program to prepare DOE SNF for disposal and to develop a repository program which will minimize costs to the Department and the American taxpayer. Although EM and RW have made progress in this area with the establishment of a formal interface between the two programs and the establishment of Preliminary Requirements for the disposal of DOE SNF, much more remains to be accomplished.

Session 47 -- PACKAGING & TRANSPORTATION I

Co-chairs: Carole Bentz, EJB&A;
Ted Gado, Foster Wheeler Environmental Corp.

47-1

DEVELOPMENT OF A STANDARDIZED DISPOSAL CONTAINER FOR THE DEPARTMENT OF ENERGY'S LOW LEVEL WASTE

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ABSTRACT

The Department of Energy (DOE) is the largest generator of low level radioactive waste (LLW) in the United States (generating nearly 70% of the total national volume). In 1993, DOE disposed of more than 50,000 cubic meters of LLW.

This waste is packaged in a variety of containers and disposed of via shallow land burial at the Hanford Reservation in Washington state, the Idaho National Engineering Laboratory in Idaho, the Los Alamos National Laboratory in New Mexico, the Nevada Test Site in Nevada, the Oak Ridge Reservation in Tennessee, and the Savannah River Site in South Carolina.

In 1995, the Department initiated an effort to develop a standard container for disposal of DOE's LLW. The objective was to enable DOE to order larger numbers of containers at one time, thereby reducing purchasing costs. Further, DOE anticipated that a standardized container had the potential to reduce costs in two additional ways: 1) by enabling new disposal container fabricators to enter the market, competition would be enhanced and prices reduced; and 2) by enabling the generating and disposal sites to concentrate on a single container, future storage, loading and disposal operations would be simplified, different types of equipment required for these operations would be minimized, and operating costs would be reduced.

To ensure a new container would be used by the different DOE sites, a development team was established. This team included representatives from the major generating and disposal sites. Representatives of the Transportation Management program were also included, since another objective was that the container be a DOT-certified 7A Type A transportation package. A key factor in site acceptance and use of this container was the process of achieving consensus among generators and disposal operations personnel at DOE sites.

This paper describes the process followed by the Department's recycle policy development team, including the container development team, to identify key factors important to ensuring equity in recycling and to ensure that recycling could be pursued in a cost-effective manner. Information is also presented on the objectives and process of development of the standardized container, as well as the specific parameters (fabrication metal, size, handling requirements etc.) of the container.

BACKGROUND

Millions of tons of potentially recoverable materials have accumulated over the years at U.S. Department of Energy (DOE) sites and facilities now undergoing environmental restoration. These materials include thousands of tons of scrap metals that can be recycled into new products, to conserve natural resources and avoid costly disposal. While some recoverable materials can be free-released and possess a significant market value, other materials are contaminated either on the surface or in mass, which limits their reuse or recycle in the open market.

The DOE environmental restoration program is considering a national policy for recycling radioactively contaminated scrap metals (RSM) within the DOE complex. The existing inventory of metals in scrap piles has been variously estimated to include from 150,000 to 400,000 tons of RSM. A large percentage of this is radioactively contaminated carbon steel (RCCS), the focus of the policy analysis. The "Recycle 2000" policy under consideration is investigating the fabrication of RCCS into ingots or

waste containers to provide for better management of DOE wastes. Risks and costs of transportation would be associated with shipping the RCCS from the DOE metal-generating sites to the processing (including fabrication) or disposal sites and shipping the products (i.e., ingots or waste containers) from the processing site(s) to disposal or use sites. The DOE environmental restoration program has initiated an assessment of the risks and costs associated with transporting RCCS and its potential products throughout the DOE complex.

THE RECYCLE 2000 POLICY PROPOSAL

To provide for responsible management of RCCS, the Department proposed the following policy: By the year 2000, 50% of low-level waste disposal containers will be fabricated from DOE-generated RCCS. These disposal containers, to be used one time only, will be used exclusively for disposal of low-level waste (LLW) generated by the Environmental Management (EM) program. If insufficient newly generated RCCS is available to meet the 50% goal, the proposed policy would be to refrain from burying potentially recyclable RCCS (i.e., that material already stored in scrap piles at various sites) and to use all available containers made from RCCS.

AN INNOVATIVE POLICY DEVELOPMENT APPROACH

Historically, DOE developed policies, then talked to stakeholders to explore how to implement the already-selected policy. As DOE's culture has changed, stakeholder involvement in decision making has increased dramatically. For the Recycle 2000 Policy concept, DOE invited stakeholders to identify their concerns prior to DOE's decision to pursue developing this policy. If the Recycle 2000 concept is pursued, it will be the first DOE policy decision incorporating stakeholder values and input prior to establishing the policy.

THE PROCESS

DOE identified a broad range of stakeholders to be involved in evaluating the proposed policy. These included DOE sites, regulators, industry, Public Interest Groups, local and State government (elected officials and business development representatives), and labor representatives.

A small group of these stakeholders reviewed the Recycle 2000 concept in July 1994. They were generally supportive of concept but wanted broader review. In response to this input, DOE invited a larger group of stakeholders to a December 1994 workshop to discuss the proposed policy. The workshop consisted of 42 participants from 26 organizations.

The result of this workshop was that stakeholders were supportive of the recycling concept subject to certain conditions, which included:

- protective of public and worker health and safety;
- developed through an open, credible process;
- economic compared to other viable waste disposal options;
- equitable (takes into account equity among sites and States);
- environmentally responsible, neither compromising cleanup nor adding to existing problems; and
- designed not to preclude further recycle initiatives.

Based on the above workshop conclusions, and a request from workshop participants for more and better information upon which to base policy development, DOE committed to conducting analyses of potential health and safety impacts associated with recycling radioactively contaminated scrap metal, the transportation of this material for purposes of recycling, and the cost of recycling. In addition, in response to a workshop recommendation, DOE also committed to developing a standardized low-level

waste disposal container suitable of being fabricated out of radioactively contaminated scrap metal.

The Recycle 2000 policy options considered in the various analyses mentioned above are as follows:

Option 1: Continuing RCCS disposal operations as currently practiced;

Option 2: Processing RCCS into ingots (volume reduced form) for disposal; and

Option 3: Processing RCCS into disposal containers for one-time use within the EM program.

Although initial discussions did not focus on a particular type of RSM, DOE narrowed the scope to focus exclusively on RCCS because it is abundant across the complex, its low market value limits incentive for decontamination and release, and it is suitable for waste management containers in demand with the EM program.

STANDARD CONTAINER DEVELOPMENT

As stated above, DOE had committed to developing a standardized low-level waste container suitable of being fabricated out of RCCS. Yet DOE had additional incentives for developing a standard container. DOE is the largest generator of low-level radioactive waste (LLW) in the United States (generating nearly 70% of the total national volume). In 1993, DOE disposed of more than 50,000 cubic meters of LLW via shallow land burial at the Hanford Reservation in Washington State, the Idaho National Engineering Laboratory in Idaho, the Los Alamos National Laboratory in New Mexico, the Nevada Test Site in Nevada, the Oak Ridge Reservation in Tennessee, and the Savannah River Site in South Carolina. The DOE LLW generating sites package this waste using various sizes of containers. This results in transportation-related inefficiencies, the need for differing disposal site equipment, and potential for disposal site void space. Even the DOE "B-25-type" container is not standard, as it is modified at most sites in both external dimensions and engineered capacity.

In 1995, the Department initiated the effort to develop a standard container for disposal of DOE's LLW. The objectives for the initiative were:

Design a family of standardized low level waste disposal boxes (M-100 series)

Enhance economies of scale through larger DOE orders of uniformly designed containers

Improve transportation efficiencies by minimizing variety of disposal containers used by DOE waste generators

Minimize void space at disposal sites through use of standard size containers

Reduce uncertainty for vendors of what is a "B-25-type" container

Among the various M-100 container requirements are the following:

The M -100 series containers must be easily fabricated, using standard tooling and nonproprietary parts;

The M -100 series container design must accommodate both RCCS and commercial fabrication paths;

Each M-100 design must use a single gauge or metal thickness for all components (i.e., all 8-gauge or all 12-gauge components); and

The M -100 series containers must meet Waste Acceptance Criteria for all DOE low-level waste disposal sites.

The M-100 series of containers are designed so a fully loaded container can be lifted by forklift or overhead hoist and the container is suitable as a six, 55-gallon drum overpack.

To ensure that a new container would be used by the different DOE sites, a consensus approach to container development was used. A container development task force was established that included representatives from the major generating sites and disposal site operators. This included representatives from Oak Ridge, Savannah River, Idaho, Hanford, and Nevada. Representatives of DOE's Transportation Management Program were also included, since another objective was that the container be a DOT-certified 7A Type A transportation package.

The prototype 12-gauge, 7A-type (M-103/7A/12/90) container has been fabricated, and testing was completed in September 1995. M-100 information is being shared with commercial low-level waste generators and disposal site operators. Follow-on activities include:

Integrating M-100 containers into DOE procurement

Establishing DOE-wide commitment to use M-100 container designs for low level waste disposal

Value-analyzing M-100 designs to minimize cost of manufacture (while retaining performance requirements)

If "Recycle 2000" concept pursued, promote manufacture of M-100 containers from RCCS.

Due to the preliminary success of the standard container initiative, DOE expects to implement the fabrication and use of the standard container regardless of the outcome of the recycle policy decision.

TRANSPORTATION RISK ANALYSIS

A transportation risk analysis was conducted to provide an assessment of potential human health risks and developed unit risks and costs for transporting RCCS scrap between DOE sites. A summary of the report of the risk analysis (1) was presented at the September 1995 Recycle workshop. The report notes that the RCCS may be generated from DOE activities (current or future) or from decommissioning of DOE facilities. The transportation system risk estimates reflect preliminary information regarding the quantities of RCCS at some sites and the spectrum of radioactive contamination in RCCS at various types of DOE facilities. Transportation risks for the three options (shown above) were analyzed. For Options 2 and 3, conceptual system configuration alternatives for processing RCCS at two regional sites or one national site are also evaluated. Risks and costs of transportation would be associated with shipping the RCCS, its products (i.e., ingots or waste containers), and secondary wastes. Specifically, this assessment considers truck or rail transportation of 1) purchased containers to DOE RCCS-generating sites, 2) RCCS in boxes to disposal sites, 3) RCCS for processing into ingots or fabrication into containers, 4) ingots to disposal sites, 5) containers fabricated from RCCS to user sites, and 6) secondary waste to disposal sites. All transportation is assumed to occur by truck and rail services that are available commercially.

Given the current stage of DOE decommissioning operations, the information currently available did not permit a full-scale analysis of transportation risks. Complete RCCS inventory (physical quantity and activity) information for each major DOE site is not available; data on scrap inventories have only been compiled for a limited number of sites and there are no estimates for future scrap generation. Without extensive inventory estimates, it is not possible to determine the number of

shipments required and the associated risk totals for DOE's alternatives. Therefore, the analysis was limited to providing unit risk and cost data elements.

The risk assessment methodology used was consistent with the DOE Environmental Management Programmatic Environmental Impact Statement (EM PEIS). The endpoints analyzed were:

Cancer incidents and fatalities due to external radiological exposure from routine operations;

Cancer incidents and fatalities due to external radiological exposure from accidental release;

Cancer incidents and fatalities due to exposure to vehicle exhaust emissions from routine operations; and

Injuries and fatalities from vehicle accident trauma.

The analysis indicated that total risk is dominated by traffic accident risks. Specifically, radiological transportation risks are a small part (10% or less) of total risk for transportation of RCCS and RCCS-fabricated boxes. Additionally, due to more people being in close proximity to roads than rails, truck transportation resulted in a higher external dose than rail transportation of RCCS and RCCS-fabricated boxes. Risks from shipping the empty fabricated containers are generally lower than for unprocessed RCCS because of the potential removal or immobilization of radioactivity by the metal melting process. Risk factors that include injuries are about a factor of 10 higher than those for fatalities alone. Risk factors including total cancer incidence are about 50% higher than those for latent fatalities. Because trucks travel in close proximity to exposed populations, truck transportation results in slightly higher risks than rail transport.

Unit transportation risk factors for all options (estimated health effects/shipment mile) are on the order of 10-7 - 10-8. These include fatalities, fatal and non fatal cancers, injuries, and severe genetic effects. For both truck and rail transportation, risks varied between the three options by less than a factor of 2, indicating that all three options were roughly equal in terms of risk.

TRANSPORTATION COSTS

The assessment provides fixed and variable unit costs (dollar/shipment-mile). Fixed and variable costs vary by transport mode (truck or rail), by distance traveled, and by the form of RCCS (scrap, ingots, fabricated containers, or secondary wastes). In practice, the costs may be affected by the number of shipments and the time period covered by the contract. Either truck or rail may have lower variable costs, depending on the dimensions and weight of the material being hauled and the shipping distance. Higher costs are assumed for secondary waste transportation because of greater handling and certification costs for this material. In general, variable costs decline as shipping distance increases, and variable costs are higher per ton for fabricated containers (empty) than for scrap haulage.

Transportation costs associated with implementing any of the three options ended up contributing between 2 and 12% of the total option cost. A centralized processing site yielded the highest transportation costs, contributing 10-12% of the total option cost, while a regionalized processing site yielded transportation cost contributions of 7-9%. Continuing present operations yielded the least transportation cost contribution of 2%. Transportation costs associated with processing to ingots for volume reduction ($1.72/ft3 for regionalized processing and

$2.65/ft3 for centralized processing) were slightly less than transportation costs associated with fabricating disposal containers from RCCS ($2.12/ft3 for regionalized processing and $3.26/ft3 for centralized processing).

(For the centralized processing scenario, all RCCS origin sites shipping to a single processing site. A U.S. geographic centroid represents the fictitious centralized destination site. For the regionalized processing site, waste generating sites ship to either of two regional processing centers. The analysis showed that total risk is proportional to mileage, and Option 1 has the lowest estimated risk due to this option resulting in the lowest mileage. The analysis also showed that lower risk results from two regional processing facilities rather than a single, centralized processing facility, due to the RCCS and RCCS-fabricated boxes being transported over fewer miles. Additional processing sites did not provide substantial reductions as compared to two sites.)

CONCLUSION

Based on the transportation risks and costs, as well as other data presented at the workshop, participants encouraged DOE to establish a recycling policy with a 2-3 year demonstration, then reevaluate the success and cost of the policy. The participants felt that a decision based on the limited cost data available so far would result in the selection of either Option 2 or Option 1, and workshop participants clearly believe the "right thing for the environment" is to make disposal containers from RCCS. Participants also encouraged DOE to clearly state in the policy that a box made from RCCS is not waste, but is a product. This distinction will enable any site to use a box made from RCCS from either its own site or any other site without causing disposal site concerns. Participants also encouraged DOE to explore conducting a demonstration of RCCS recycling if it appeared too difficult to establish an EM-wide policy.

Based on the strong support for recycling indicated by workshop participants, DOE staff has developed a draft policy package to be submitted for management approval. This package recommends that the Office of Environmental Management establish a policy that recycling of contaminated metals should be pursued.

Based on the consensus of the container development team, DOE staff is also developing a draft policy package for issuance by HQ management. This package will inform field management of the standardized container and will encourage its use. This package will also enable aggregate purchasing of standardized containers, and may ease the practical aspects of making standardized containers out of volumetrically contaminated carbon steel.

One of the demonstration activities being conducted in advance of formal approval of the recycling policy is the Savannah River Site's contract with Scientific Ecology Group (SEG) in Oak Ridge to accept contaminated lead for decontamination and reuse. By the end of 1995, the SRS had shipped 20 tons of contaminated lead to SEG to begin the demonstration of an effective decontamination technology. It is expected that the lead will be recycled for reuse and a per unit volume cost will be established, which will support future decontamination and recycle activities. Additionally, 20 tons of contaminated HEPA filter frames have been shipped to SEG for treatment through melting and recycle/reuse. The radioactively-contaminated lead items were shipped to SEG as scrap metal for recycling, not as waste. The distinction is important so that the

scrap metal can be recycled and not, instead, disposed of as DOE-generated waste.

DOE will be identifying and implementing additional innovative approaches to ensure recycling is pursued to the maximum extent possible across the complex.

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47-2

IMPACTS OF A LLW TRANSPORTATION ACCIDENT IN AN URBAN AREA OF TEXAS

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ABSTRACT

In response to questions from members of the public regarding the Texas Low-Level Radioactive Waste Disposal Authority's application for license to develop and operate a LLW disposal facility, the effects of a highly improbable transportation accident were analyzed. This paper summarizes the results major activities in the analysis, considers the probabilities of such an accident, and identifies sources of conservatism in the analysis.

INTRODUCTION

Members of the public have commented that the application for a license to develop and operate a low-level radioactive waste (LLW) disposal facility that was submitted by the Texas Low-Level Radioactive Waste Disposal Authority (the Authority) did not adequately address transportation risks. One comment requested that the Authority evaluate the effects of a worst case transportation accident occurring in an urban area.

The Authority is not obligated under applicable Texas laws or regulations to justify that the potential impacts of transportation accidents are within acceptable limits. Nevertheless, the Authority performed and reported the results and conclusions of the requested analyses. The conclusion is that members of the general public would not suffer unacceptable risks if the postulated worst case transportation accident were to occur in a densely populated urban area.

An extreme accident involving the burning of the worst waste streams for external gamma radiation and inhalation exposures was postulated to occur at two separate urban locations. The accident was assumed to involve the collision between a truck transporting LLW and a tanker truck filled with

gasoline or similar fuel. The waste containers were assumed to rupture and their contents to be consumed by the fire over a three hour time. The accident was characterized by a series of extreme assumptions that ensured the projected radiological impacts would not be underestimated. The approach to estimate the radiological impacts of the hypothesized accident involved six major steps:

1) Identify and characterize all waste streams that will be disposed at the Texas LLW disposal facility.
2) Screen waste streams from the respective generators for their potential to produce external gamma radiation and inhalation exposures.
3) Identify likely waste transportation routes from the respective waste generators.
4) Define the worst-case transportation accident scenario.
5) Perform calculations to estimate radiation exposures to the public as a result of the worst-case transportation accident using the RADTRAN4 computer code.
6) Summarize and interpret results of calculations.

This paper focuses on the methodologies used to identify the limiting waste streams to be considered in the analyses. It also addresses the probability of the accident analyzed actually occurring and identifies sources of conservatism in the analyses.

IDENTIFYING THE CRITICAL WASTE STREAMS

As noted above, the first step in the process was to identify all sources of LLW. The Authority had projected waste volumes and characteristics expected for disposal at the proposed disposal facility in earlier publications (1,2). Waste stream characteristics were based on the results presented in those documents. Because the objective of these evaluations was to estimate the effects if the accident were to occur, no information about shipment rates were necessary to conduct the requested analyses.

The premise of screening all waste streams is that the critical waste stream(s), those likely to produce the worst radiation exposures, depend only upon their radionuclide concentrations. That is, the details of the RADTRAN4 calculation are immaterial to determining the potential for producing radiation exposures. This screening approach, described below, implicitly assumes that the entire radioactive inventory of a waste container is available to produce exposures. This assumption is clearly a substantial conservatism that is further discussed below.

The concentration of each radionuclide in each waste stream was multiplied by two dose conversion factors -- one for external gamma exposures and the other for inhalation exposures. The product was further multiplied by factors to render the result in common units. The product is a "figure of merit" that differs from the actual hazard only by some multiplicative factor that accounts for processes and phenomena that do not depend upon waste stream characteristics. Thus, waste streams with larger figures of merit present greater risks of radiation exposure.

Each waste stream from each generator was evaluated for its potential to produce external gamma and inhalation radiation exposures. This process is illustrated in Table I for LLW waste steams generated at the Comanche Peak nuclear power plant. The process was applied identically for waste streams from all other generators, although non-utility wastes were treated as a composite from all non-utility waste generators. Table I demonstrates that, for Comanche Peak waste streams, cobalt-60 is the primary source of potential radiation exposure (considering both external

gamma and inhalation radiation exposures) with plutonium-241 as the secondary source.

Table I

The sum of the products described above is a figure of merit that represents the total potential for radiation exposure for the waste stream. These total across all radionuclides are shown at the bottom of Table I.

As noted above, each waste stream from each generator was screened as depicted in Table I and described in the preceding paragraphs. The screening of waste streams generated at the Comanche Peak nuclear power plant is summarized in Table II as a further illustration of the screening process. In the table, the figures of merit (the sum of all radionuclide products within waste stream) are tabulated for both external gamma and inhalation exposures for each waste stream. Radionuclide characteristics of Class C resins and filters, denoted with bold text in Table II, have the greatest potential of producing radiation exposures.

Table II

As a final illustration of the waste stream screening process, Table III summarizes the results for all generators expected to deliver waste to the proposed Texas LLW disposal facility. The waste stream with the greatest potential to produce radiation exposure is presented for each generator. As shown in the table, Class C resins and filters from the Comanche Peak nuclear power plant have the greatest potential for producing both external gamma and inhalation exposures. The characteristics of this waste stream were used in RADTRAN4 calculations of the impacts of the worst-case transportation accident.

Table III

PROBABILITY OF WORST-CASE TRANSPORTATION ACCIDENT OCCURRING

The very act of selecting such a severe accident ensures that the probability of the accident occurring will be low. Everyday experience tells us that extreme accidents occur rarely. Accident statistics confirm this casual observation. For example, among non-collision accidents, only 2.8 percent also involved fires in 1990. Whereas 35,885 accidents occurred in 1990 involving motor carriers in interstate transport, only about 5 percent of these also involved hazardous materials. Of all hazardous materials shipped, only an extremely small fraction is LLW (5). In 1985, about 250 billion ton-miles of freight were hauled by vehicles regulated by the Interstate Commerce Commission. In that same year, about 3,700 accidents involving trucks occurred that resulted in fatalities. This yields an annual fatal accident rate for trucks of about 1.5 X 10-8 fatalities per ton-mile. If the assumption is made that the accident considered in these analyses results in the death of at least one individual, this accident rate can be used to infer the expected number of accidents involving LLW (6). The amount of waste expected for disposal at the Texas LLW disposal facility over its lifetime is about 170,000 tons (1,2). The typical shipping distance (from Houston to El Paso) is about 750 mile. Given the accident rate stated above, about 2 accidents involving LLW shipments would be expected over the facility's operating life.

By contrast, in 1985, 3,825 motor vehicle accidents occurred in Texas that resulted fatalities (6). Thus, over the minimum 20-year operating life of the Texas LLW disposal facility, about 76,000 traffic accidents severe enough to produce fatalities might be expected. The potential of a

few accidents involving shipments of LLW to the Texas LLW disposal facility cannot add measurably to this potential transportation impact. In the analyses of transportation risks performed in response to public questions regarding the Texas LLW disposal facility license application, the accident was assumed to occur in an urban area. The probability of this occurring is small because of the relative number of miles near urban areas and in rural areas. Over the 750-mile trip to the disposal facility from Houston, less than about 50 miles could be in urban areas typical of that assumed in these analyses (namely near Houston and San Antonio). On the basis of miles alone, the probability that the postulated accident would occur in a rural area where radiological impacts would be much smaller is over 90 percent. Only a 7-percent probability exists that the accident would occur in the postulated urban area.

CONSERVATISMS INHERENT IN PROJECTED RADIATION EXPOSURES

Numerous conservative assumptions were made in estimating impacts of LLW transportation to the Texas LLW disposal facility. Among these include:

Accident occurs in an urban area.

Accident involves a truck transporting gasoline or similar fuel which results in a 3-hour fire.

Contents of all waste containers in the shipment are released and available to be consumed by the fire.

The fire consumes all waste in the shipment, releasing the entire radioactive inventory of the shipment into the atmosphere.

Were the effects of these conservatisms to be quantified, they would amount to several orders of magnitude. It is highly unlikely that the projected impacts of the worst-case LLW transportation accident would be as large as reported by the Authority.

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EXPECTED RESIDENCE TIME MODEL

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ABSTRACT

The Transportation Technology Department of Sandia National Laboratories develops analytical and computational tools for the U.S. Department of Energy to assess the radiological consequences and risks from the transportation of radioactive materials by all modes. When large quantities of materials are to be transported, movements may occur over an extended period of time in what is collectively referred as a "shipping campaign." Since the routes over which the shipments occur often remain the same, cumulative exposure to individuals inhabiting the population zones adjacent to the transport links must be estimated. However, individuals do not remain in the same residences throughout their lifetimes and, in fact, move quite often. To appropriately allocate exposures among populations over extended periods of time, perhaps years, a model has been developed that accounts for three population categories; the original populations residing in areas adjacent to the transport links, and both the individuals moving into and out of any designated areas. The model described herein accounts for these populations and will ultimately be incorporated as a user option in the RADTRAN computer code, used for estimating the consequences and risks associated with the transport of radioactive materials.

The simplified methodology is an extension of previous work that uses available census data to determine the total exposed population strictly as a function of the original population, and is therefore applicable to a wide range of problems. It is important to note the mathematical constructions are predicated upon census data in terms of "households," and as such may not be applicable to individuals. The households are contextually integrated into populations, inclusive of several regional, property-utilization, and density subgroups. The result is an expression specifying the total population (in terms of "households") residing in an area over time t years. Significantly, only the original population and the average time of residence need be known. With respect to the transportation of radioactive materials, the relationship provides an estimate of the total exposed population over an entire shipping campaign while only knowing the initial population density. The method has several strengths including simplicity, dependence on few critical parameters, and a firm foundation in empirical data. The inherent advantages from application of this methodology are twofold: 1) the calculated average dose to members of the public residing near transportation routes will decrease as not all persons remain near a link for the full period of the shipping campaign, and 2) the statistical data are amenable to uncertainty analyses, which in turn yields improved dose-consequence estimates for the population group.

INTRODUCTION

The Transportation Technology Department of Sandia National Laboratories develops analytical and computational tools for the U.S. Department of Energy to assess the radiological consequences and risks from the transportation of radioactive materials by all modes. When large quantities of materials are to be transported, movements may occur over an extended period of time in what is collectively referred as a

"shipping campaign." Since the routes over which the shipments occur often remain the same, cumulative exposure to individuals inhabiting the population zones adjacent to the transport links must be estimated. However, individuals do not remain in the same residences throughout their lifetimes and, in fact move quite often. To appropriately allocate exposures among populations over extended periods of time, perhaps years, requires a model that accounts for three population categories: 1) the original populations residing in the areas adjacent to the transport links, 2) individuals moving out and 3) individuals moving into residences in the designated areas. The model described here accounts for these conditions and will be incorporated as a user option in the RADTRAN computer code for transportation consequence and risk analysis (1). RADTRAN is a computer code for estimating the consequences and risks associated with the transport of radioactive materials.

METHODOLOGY

The most mathematically elegant model would be predicated on an analytical "double exponential," composed of terms describing positive exponential biological ingrowth and negative exponential decay, with the latter made up of terms describing both individuals moving away and mortality. However, the associated coefficients describing these sub-populations must be empirically derived and can restrictively lead to instances in which the model does not apply to specific populations. Therefore, we selected the simpler methodology developed here, which is an extension of previous work (2) that uses available census data and is applicable to a wide range of problems.

As a prerequisite in presentation of the methodology, it is necessary to define the major variables:

P0 = Initial population in designated area
Pin = Population moved into designated area
Pout = Population moved out of designated area
Premaining = Population remaining in designated area
Ptotal = Total exposed population in designated area

The key to solution is application of two initial conditions: 1) over the time periods of interest, population distributions within the relatively small areas being analyzed are modeled as remaining constant, and 2) all established residences are modeled as being occupied. The latter condition means that although individual households may leave an area, other households move in to occupy the vacated residences (100% occupancy). Thus, although individuals in the population Pin are distinct from the individuals of Pout, the population groups are approximately numerically equivalent in magnitude (PinPout).

This simplified procedure allows determination of the total exposed population (Ptotal) strictly as a function of P0. Further, the 100% occupancy condition slightly overestimates the actual population. As some households may move to another residence within the area and since housing-unit occupancy may not be 100%, the calculated total population size will also be an overestimate (unless localized rapid population growth has occurred). The methodology proceeds from determination of the original population P0, through summation of all individual link/bandwidth populations comprising the complete route. Over the time interval of interest (the period of the shipping campaign) household populations Pout and Pin will move out of and into the sample space, respectively. The quantity of interest, the total exposed population Ptotal, consists of the original population P0 plus Pin, comprising the

population of individuals who moved into the sample space during the campaign (Fig. 1, Eq. 1). At the end of the shipping campaign, P0 consists of the populations of households which remained (Premaining) plus those which moved out of the sample space (Pout) (1)(Eq. 2).

It is important to note that the procedure considers populations and that the mathematical constructions are predicated upon census data expressed in terms of "households," and as such may not be applied to individuals. The households are contextually integrated into populations, inclusive of several regional, property-utilization, and density subgroups. Although some of the data subsets are available (Table I) and may be incorporated into the model, only the aggregate of all households is considered here.

Fig. 1

The result is an expression specifying the total populations (in terms of "households") residing in an area over time t years (Fig. 1, Eq. 3). Significantly, only the original population and the average time of residence need to be known. With respect to the transportation of radioactive materials, the relationship allows an estimate of the total population exposed over the entire interval of a shipping campaign to be developed, strictly as a function of the initial population density and duration of the shipping campaign. The derivation of the remaining variable Rt is adapted from published census data analyses (2).

Curve-fit coefficients, shown in Table I, were applied to empirically determine the fraction of households moving into an areas St, and the fraction remaining in their current residences Rt.

St = Fraction of households which moved into current residence t years before survey

Eq. 4

Eq. 5

Eq. 6

Table I

The following example illustrates derivation of an equation to calculate the total exposed population of "all houses" for a multi-year shipping campaign. Values listed within Table I are first used to develop an expression for Rt.

Fig. 2

Substituting the relation for Rt of Eq. 7 into Eq. 3 yields

Eq. 7 into Eq. 3

Eq. 8

In a recent environmental assessment of the impacts of transporting foreign research reactor spent nuclear fuels (3), a total population of approximately 139,403 persons was estimated to be exposed during highway transport along the route from Charleston, SC to the Savannah River Plant near Barnwell, SC. For a shipping campaign lasting 10.0 years, substitution of time t = 10 years and initial population of P0= 139,403 persons into Eq. 8 yields a total potentially exposed population Ptotal = 261,747 persons. Significantly, the total potentially exposed population (Ptotal) is nearly twice the initial population (P0), formerly the only value used in dose-consequences analyses of shipping campaigns.

CONCLUSION

A method has been developed for estimating the total potentially exposed population of persons residing near transportation links, during time intervals required to complete a shipping campaign based upon U.S. Bureau of Census data and analyses (2).

This method has several strengths including simplicity, dependence on few critical parameters, and a firm foundation in empirical data. Conversely, two weaknesses of the method are: 1) one cannot readily account for rapid changes in overall population density (examples being explosive population growth or decline resulting from abrupt socioeconomic changes brought about by extensive construction projects or military facility closures), and 2) it is not possible to account for less than 100% occupancy factors. However, in all cases but those associated with rapid growth, the method yields estimates that are slightly conservative (i.e., overestimates the total exposed population). Should a route segment experiencing rapid and significant growth be identified in a specific application, then additional calculations may be required to improve the population estimate for that link.

The ultimate consequences of application for this methodology are twofold. First, the calculated average dose to members of the public residing near transportation routes will decrease, as not all persons remain near a link for the full time period of the shipping campaign. Secondly, the statistical data presented in Table I are amenable to uncertainty analyses, which in turn yields improved dose-consequence estimates for the population group.

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TRANSPORTATION OF RADIOACTIVE MATERIALS IS ENVIRONMENTALLY BENIGN -- LET'S QUIT ANALYZING IT TO DEATH

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ABSTRACT

The Department of Energy (DOE) and other Federal agencies continue to spend millions of dollars on National Environmental Policy Act (NEPA) and other analysis of transporting radioactive materials. This is a particular problem associated with high-level and transuranic waste and spent nuclear fuel. This paper will show the folly of continuing this expensive and time consuming analysis.

A very high degree of safety has been achieved, both domestically and worldwide, through rigorous implementation of the comprehensive Department of Transportation (DOT) and Nuclear Regulatory Commission (NRC) transportation and packaging regulations. There are no reported deaths or serious injuries due to the radioactive nature of the cargo.

There are few, if any, such large scale industrial activities around the globe with this enviable safety record. DOE, NRC, and other federal agencies have exhaustively analyzed and demonstrated the intrinsic safety of transporting radioactive materials. Further NEPA analysis, is clearly not warranted.

This paper will examine the evidence supporting its title thesis. Specifically, it will document the compelling support represented by:

- Recent (and comprehensive) DOE NEPA analysis
- Environmental impact analysis done by the Nuclear Regulatory Commission (NRC)
- NRC's NEPA Rule contained in 10 CFR 51
- Office of Technology Assessment review of the safety of hazardous materials transport

The sum of this analysis will demonstrate the efficacy of the regulatory framework for transporting radioactive materials, thus eliminating the need for further NEPA analysis.

REGULATORY FRAMEWORK

Recommendations for radiation protection standards come from the International Commission on Radiation Protection (ICRP) and the U.S. National Council on Radiation Protection and Measurements (NCRPM). These organizations are composed of physicians, radiologists, and scientists specialists specializing in the biological effects of radiation. Independent reviews are provided by the U.S. National Academy of Sciences and the United Nations Scientific Committee on the Effects of Atomic Radiation. The International Atomic Energy (IAEA), an agency of the United Nations, issues and updates Basic Standards for Radiation Protection (Safety Series No. 9), which reflects the recommendations made by the ICRP, following review by IAEA member state-designated experts. In turn, IAEA also issues and updates the Regulations for the Safe Transport of Radioactive Material (Safety Series No. 6), which considers the requirements of Safety Series No. 9. Although written in regulatory language, IAEA's Safety Series No. 6 may only legally serve as recommendations. However, the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission have structured their regulations covering U.S. shipments on requirements of IAEA Safety Series Nos. 6 and 9. These standards are based on more than 50 years of research and experience with radioactive materials and their safe transport. No other standards have been so extensively reviewed and agreed upon by international experts, organizations, and nations.

SUPPORTING ANALYSIS

Recent DOE experience in preparing major EISs continues to involve significant analysis of transporting radioactive materials--particularly spent nuclear fuel (SNF) and radioactive waste. Since the inception of NEPA, such analysis has never shown transportation of radioactive materials significantly impacting people or the environment. This analysis is in itself very costly and increases cost by stretching schedules. The following are examples of analysis which provides compelling support for a categorical exclusion.

DOE NEPA ANALYSIS

1. Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Final Environmental Impact Statement (EIS) (April 1995) - Volume I of this EIS evaluates the impacts of managing DOE-owned spent nuclear fuel SNF. It includes the most exhaustive analysis of

transportation risks for SNF by sea, rail, and highway done anywhere. All of the alternatives, including those that could involve up to 9,200 shipments, would have very small environmental consequences. The results are conclusive--this is a very low risk activity. Similarly, the risks and consequences for the Volume 2 transportation activities are very low. The Agency preferred alternative here includes enhancing ongoing spent fuel management, environmental restoration, and waste management activities to meet current and expanded waste management needs. Significant shipments of spent nuclear fuel, transuranic, mixed low-level, and low level wastes were considered in the analysis. The Record of Decision was issued on May 30, 1995.

2. Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel - This Draft EIS was issued in April 1995 and public comments are being received. The most significant component of analysis is on transportation risks and impacts for SNF. Activities covered include sea transport, handling in ports, overland transport by highway and rail, and receipt at DOE sites. Up to 22,700 SNF elements (721 cask loads) could be accepted at one or more of 5 different sites. This is a thorough analysis of a massive domestic and international SNF shipping campaign. Again, the risks and environmental impacts are found to be insignificant.

3. Environmental Assessment (EA) of Urgent - Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (April 1994) - This EA considered acceptance of up to 409 SNF elements from Europe. This document contains an extensive transportation analysis projecting no significant impacts. A Finding of No Significant Impact (FONSI) was issued and the spent fuel was received in September 1994.

4. Draft Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste (PEIS--Predecisional Draft May 12, 1995) - This Draft PEIS is now in internal concurrence with anticipated issue in July. It examines the risks and impacts of alternative future configurations for selected waste management facilities. All waste types are considered including high-level, transuranic, low-level, low-level mixed, and hazardous. Alternative configurations are decentralized, regionalized, and centralized. Each has a transportation component with the centralized alternative which would involve massive shipments of all waste types. The preliminary conclusion is there is insignificant risk due to the radioactive nature of the cargo in transit. As with any commodity in normal transportation commerce, the predominant risk (while still low) is with potential non-radiological highway and rail accidents.

5. Final Environmental Impact Statement--Waste Isolation Pilot Plant (WIPP) (October 1980) - The original EIS in its "Authorized Alternative" examined the impacts of shipping transuranic waste to WIPP. It included both contact and remote handled shipments at a 500 per year level. The analysis was done for both highway and rail shipments with very low impacts calculated. This EIS was supplemented in January 1990 to evaluate the environmental impacts associated with new information and changes in the proposed action. In the supplement, the analysis of transportation activities was greatly expanded including maximizing rail (vs highway) shipments, and an all-highway scenario (20,903 shipments). Impacts from the radiological nature of the shipments are low.

6. Environmental Assessment--Return of Isotope Capsules to the Waste Encapsulation and Storage Facility, Hanford Site, Richland, Washington

(May 1994) - This EA examined the environmental impacts of transporting cesium-137 capsules to Richland from Colorado (309) and Virginia (25). On-site transfer of 33 capsules from Pacific Northwest Laboratory was also included. The cesium inventory analyzed ranged from about 675 to 725 kilocuries per shipment. A FONSI was issued on May 11, 1994 and the Colorado shipments are now complete.

NRC NEPA ANALYSIS

1. Nuclear Regulatory Commission, NUREG-0170, Final Environmental Statement on the Transportation of Radioactive Materials by Air and Other Modes (1977) - This was the definitive NEPA study supporting nationwide movement of radioactive materials for over a decade. It provided a generic method to characterize the potential severity of transportation accidents. The risks and consequences were analyzed and found very low. However, this document has not been updated and both major and minor shipping activities began to be independently evaluated (WIPP for example in 1980). The analysis, however, remains undisputed by more recent NEPA documents -- risks are indeed low.

2. Nuclear Regulatory Commission, NUREG/CR-4829, shipping Container Response to Severe Highway and Railway Accident Conditions (1987) - This is commonly referred to as the "Modal Study". The Modal Study was the result of an initiative taken by the NRC to refine more precisely the analysis presented in NUREG-0170 for spent nuclear fuel shipping casks. Whereas the NUREG-0170 analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the modal study relies on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions. It's accident severity scheme is the basis for consequence analysis in DOE Spent Nuclear Fuel EIS work today.

3. Shipment of Core Rubble from Three Mile Island (TMI) in Pennsylvania to Idaho - A very significant DOE precedent was set on these shipments. DOE determined that the TMI shipments fall within a categorical exclusion. In reaching its conclusion DOE relied on:

NUREG-0170, Final Environmental Statement on the Transportation of Radioactive Materials by Air and Other Modes (1977).

NUREG-0683, Final Programmatic Environmental Impact Statement Related to Decontamination and Decommissioning of Radioactive Waste Resulting from March 28, 1979 Accident: Three Mile Island Nuclear Station, Unit 2 (March 1981).

The Department stood firm on the issue of environmental analysis requirements and completed the highly controversial TMI-2 shipping campaign between July 1986 and April 1990. The all-rail campaign included 49 cask load of damaged nuclear fuel and core rubble in 22 shipments.

NRC NEPA RULE

The NRC's NEPA rule is contained in 10 CFR 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions. These regulations cover their basic licensing activities including the ability to possess and transport radioactive materials. The presumption in 10 CFR 51 is that the licensing of an activity (not the activities conduct or implementation) is the action triggering a potential to impact the environment.

A wide variety of transportation (distribution) and use of radioactive materials are listed as CX in 10 CFR 51.22(14). Included are radiopharmaceutical, generators, sealed sources, gauging devices, instruments, waste, uranium munitions, and other source and by product material. It is estimated there may be as many as 2 million of such shipments yearly. In addition to supporting our transportation proposal, this suggests EH-s B2.6 change is totally unnecessary. A DOE CX for packaging, transporting, and storage of sources (which happens infrequently) is a minor operational activity not worthy of a specific CX. Proposed change B1.29 covering "occasional" transportation appears gratuitous and short sighted in light of NRC's position. Conceivable DOE could avoid its own restrictive NEPA rule by having a licensee make DOE shipments.

One final comment on transportation -- NRC also addresses the "controversy" issue. In section 51.22(b), exceptions to CX eligibility include "special circumstances" which appears to address the "controversy" question in CEQ 1508.27(b)(4). For NRC, these special circumstances center around unresolved conflicts concerning alternatives uses of available resources. There is no indication "controversy" is simply organized activists complaining about environmentally benign activities.

NRC's rule on SNF storage is even more supportive of Jill Lytle's proposal on SNF/HLW storage. NRC provides a CX for SNF storage for 30 years beyond the life of a reactor (10 CFR 51.23). This includes storage in reactor basins and both onsite and offsite independent spent fuel storage installations. This CX strongly suggests DOE should go even farther and make HLW/SNF storage a CX. You may want to take the lead and set up a meeting with EH on the subject.

Section 10 CFR 51.22(c)(11)(ii) allows a CX for activities where "there is no significant increase in individual or cumulative occupational radiation exposure". As an excluded class of action, this would have an immense effect on streamlining NEPA in the Department. It seems inconsistent DOE is moving from self-regulation to considering NRC regulation and not adopt the more aggressive (and fully justified) NEPA rule of an independent regulator. In fact the entire 51.22(c)(11) CX provision is ripe with significant opportunities for DOE to base its own NEPA streamlining on NRC precedent. It is also an excellent opportunity for EH to far exceed its monetary savings goal for streamlining. And -- it takes the "high road" of following the NRC lead.

CONGRESSIONAL REVIEW

1. Office of Technology Assessment (OTA)--Transportation of Hazardous Materials (July 1986) - This document was prepared at the request of Congress as a comprehensive assessment of the regulations, information systems, container safety, and training for emergency response, and enforcement as it deliberated reauthorization of the Hazardous Materials Reauthorization Act. A select and prestigious advisory panel led this definitive review of the "system" responsible for safe transport of over 1 billion tons of hazardous materials per year. For radioactive materials, this report cited areas for improvement and some arguable criticisms. However, concerning certified shipping containers, OTA stated:

"The NRC cask certification process is, of necessity, painstaking and time-consuming. The proven safety record of NRC-certified casks, however, provides a degree of public confidence in casks. OTA finds that technical

evidence and cask performance in service indicate that NRC performance standards yield spent fuel shipping cask design specifications that provide for a very high level of public protection--much greater than that afforded in any other current hazardous materials shipping activity."

AN EXAMPLE OF NEPA EXCESS

The Office of Nuclear Energy is preparing an Environmental Impact Statement for the Medical Isotopes Production Project: Molybdenum-99 and Related Isotopes. A Predecisional draft was issued in September, 1995. The objective of this project is to develop a backup capacity for 10-30% of the U.S. demand over the next 5-10 years. The U.S. currently imports all these isotopes from Canada. The project would also include the capability to supply 100% of U.S. demand if the Canadian reactor is shut down. Molybdenum-99 decays into technetium-99 and is the most widely used medical isotope in the U.S. There are about 36,000 procedures each day using technetium-99.

This project on close examination is a very modest undertaking involving modified use of an existing DOE reactor. Reactors at Los Alamos, Oak Ridge, Sandia, or Idaho could be used. Also included in the project is the capability to fabricate targets, process irradiated targets, and package and ship recovered isotopes. A priori, from decades of past experience, the project would clearly have minimal environmental impacts. It would contribute virtually nothing to cumulative impacts at DOE sites with massive (by comparison) nuclear activities.

The Department could have easily supported a categorical exclusion (CX) for this project based on NRC regulations. The NRC's NEPA rule is contained in its Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions (10 CFR 51). If this project were being done by a NRC licensee, it would have been a modification to a license and thereby a CX. This prior NEPA analysis and conclusion by an independent regulatory authority should have been considered in deciding to do an EIS. In today's environment, a CX might be extreme. However, the NRC experience alone could have been sufficient justification for selecting an Environmental Assessment (EA) as the proper NEPA document for DOE.

The draft EIS contains extensive analysis of transporting radioactive materials. The EIS level of analysis is conspicuously extreme. It is a CX under 10 CFR 51 which should have been invoked. The EIS baseline for analysis is 3,225 shipments per year. By comparison (for this NRC categorically excluded activity), there are an estimated 2,000,000 shipments yearly in the U.S.

In the past, an important precedent was set for invoking a NRC CX. The Department conducted the entire Three Mile Island core rubble shipping campaign (Pennsylvania to Idaho) based on a CX. The DOE relied on prior NRC NEPA analysis in taking the position. This position was never challenged in court since the legal precedent was clear. DOE steadfastly defended this position in spite of congressional objection.

Finally, much is said about "controversy" as a justification for EIS levels of analysis. Its hard to imagine a more controversial shipping campaign than TMI core rubble. Its also hard to imagine something as beneficial as isotopes production being challenged for less than an EIS level of NEPA review.

CONCLUSION

A very high degree of safety has been achieved, both domestically and worldwide, through the rigorous implementation of comprehensive transportation and packaging regulations. There are no reported deaths or serious injuries due to the radioactive nature of the cargo. There are few, if any, such large scale industrial activities around the globe with this enviable safety record. Both DOE and NRC have exhaustively analyzed and demonstrated the intrinsic safety of transporting radioactive materials. This analysis coupled with its underlying (extremely) conservative assumptions, show a categorical exclusion is clearly warranted and will result in significant savings.

Session 48 -- TECHNOLOGY AND REGULATORY INNOVATIONS
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Sonny Goldston, WSRC
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TECHNICAL APPROACH TO FINALIZING SENSIBLE SOIL CLEANUP LEVELS AT THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT
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ABSTRACT
The remedial strategy for addressing contaminated environmental media was recently finalized for the U.S. Department of Energy's (DOE) Fernald Environmental Management Project (FEMP) following almost 10 years of detailed technical analysis. The FEMP represents one of the first major nuclear facilities to successfully complete the Remedial Investigation/Feasibility Study (RI/FS) phase of the environmental restoration process. A critical element of this success was the establishment of sensible cleanup levels for contaminated soil and groundwater both on and off the FEMP property. These cleanup levels were derived based upon a strict application of Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) regulations and guidance, coupled with positive input from the regulatory agencies and the local community regarding projected future land uses for the site. The approach for establishing the cleanup levels was based upon a Feasibility Study (FS) strategy that examined a bounding range of viable future land uses for the site. Within each land use, the cost and technical implications of a range of health-protective cleanup levels for the environmental media were analyzed. Technical considerations in deriving these cleanup levels included: direct exposure routes to viable human receptors; cross-media impacts to air, surface water, and groundwater; technical practicality of attaining the levels; volume of affected media; impact to sensitive environmental receptors or ecosystems; and cost. This paper will discuss the technical approach used to support the finalization of the cleanup levels for the site. The final

cleanup levels provide the last remaining significant piece to the puzzle of establishing a final site-wide remedial strategy for the FEMP, and positions the facility for the expedient completion of site-wide remedial activities.

INTRODUCTION

The Fernald Environmental Management Project (FEMP) is nearing the conclusion of the Remedial Investigation/Feasibility Study (RI/FS) process, with Records of Decision (RODs) for all five operable units at the facility due to be completed by July 1996. With the conclusion of the RI/FS process, the attention of the facility is now being directed to the safe and efficient implementation of remedial actions.

Operable Unit 5 at the FEMP represents all of the environmental media (soil, sediment and groundwater) that have been impacted by past uranium production operations and waste disposal practices at the site. A ROD was recently issued for Operable Unit 5, completing over 10 years of intensive environmental investigations into the conditions at the site. The ROD established final cleanup levels for all of the environmental media and defined a strategy for the permanent disposal of contaminated soil and sediment in an on-property engineered facility. This paper is focused on the approach applied to finalize cleanup levels for soil; however, the methods were similar for the other environmental media. These cleanup levels and the associated waste management approach provide the last component to a comprehensive site-wide remedial strategy for the FEMP.

The strategy for finalizing these cleanup levels involved a process of consensus building with local residents, the Ohio Environmental Protection Agency (OEPA), the U.S. States Environmental Protection Agency (EPA) and DOE, and in marrying the CERCLA decision process with the deliberations of a citizens task force formed to make recommendations on cleanup levels and final land use.

A key objective of the RI/FS decision-making process was to arrive at final cleanup levels that were protective to existing and future human and ecological receptors as well as cost effective and implementable. Characterization data collected during the RI phase of the study revealed that small changes (i.e., reductions) in cleanup levels for the principal contaminants of concern would yield large increases in projected soil excavation volumes. With these large increases in the volume of contaminated media requiring excavation, equally dramatic shifts in remedial costs were predicted. Thus, the stakes were high at the FEMP to arrive at cleanup levels for soil that satisfied stakeholder concerns regarding long-term protectiveness and were economically sensible.

BACKGROUND

The FEMP, formerly known as the Feed Materials Production Center, is a 1050-acre DOE facility located approximately 18 miles northwest of Cincinnati. The FEMP is situated in a rural setting near the village of Fernald, Ohio. The FEMP operated from 1952 until 1989 as a large-scale production facility extracting uranium from ores and ore concentrates to yield high-purity metal products in support of U.S. defense programs. During the 38-year production history of the facility over 500 million pounds of uranium metal products were shipped from the FEMP to other DOE sites across the country. In 1989, with a decline in product demand and increasing environmental concerns, production operations were permanently shut down. In August 1991 the site was officially declared closed and the facility renamed to reflect its new mission.

The topography of the area includes gently rolling uplands with steep hillsides along major streams, such as the Great Miami River. Surface drainage on the FEMP is from east to west and south into Paddys Run, with the exception of the northeast corner which drains east toward the Great Miami River. Groundwater beneath the FEMP is found in two principal geologic units: the glacial overburden (ranging in thickness between zero and 50 feet) and the sand and gravel of the Great Miami Aquifer. Groundwater occurring in the glacial overburden is considered "perched," in that it is contained within silty sand lenses residing within a low-permeability, clay-rich soil. The underlying Great Miami Aquifer is the principal drinking water supply for the region and is regulated as a sole-source aquifer under the Safe Drinking Water Act.

In December 1984 the release of approximately 200 pounds of uranium from a plant dust collector was reported to the National Response Center. This release notification focused nationwide attention on the environmental issues at the facility and produced increased oversight by the DOE, EPA and OEPA. Local residents at the site formed a watchdog group entitled the Fernald Residents for Environment, Safety and Health. The high public and political profile surrounding activities at the FEMP has remained relatively unchanged since this initial release in 1984.

The RI/FS process was initiated at the FEMP under a Federal Facility Compliance Agreement between EPA and the DOE. The work plan for the study, prepared by DOE in 1988, identified 39 site areas for investigation. To enhance implementation of the RI/FS, the 39 areas were grouped into five "operable units" by combining similar waste areas or related environmental concerns. The operable unit concept was incorporated into the April 1990 Consent Agreement between EPA and the DOE. The RI/FS and any required cleanup of specific operable units at the FEMP are guided by the Consent Agreement as amended in September 1991, and associated work plans. These documents provide procedures and schedules to ensure investigations are conducted in compliance with federal and state environmental laws. Due to confirmed contaminant releases to the environment identified during the initial stages of the RI, the FEMP was placed on the National Priorities List in November 1989. Operable Units 1-4 are termed "source" operable units and include the former production area and associated waste management areas that were the initial points of contaminant release to the environment. Operable Unit 5 addresses all environmental media on and off the FEMP property impacted by contaminants released from the facility. Each operable unit is being managed in accordance with the schedules set in the Amended Consent Agreement, with RODs for all operable units due to be completed by July 1996.

To foster community input into the decision process, the DOE chartered the Fernald Citizens Task Force. The Task Force, which is comprised of local government officials and residents, labor leaders, FEMP employees and community leaders, focused on making recommendations to decision makers on preferred cleanup levels, waste disposition strategies and future land uses for the FEMP property. Throughout the development of the Operable Unit 5 FS and the ROD, DOE has attempted to consider the evolving deliberations of the Task Force.

The RI identified widespread contamination of surface soil, sediment and groundwater both on and adjacent to the facility as a legacy of the 38-year production mission. The RI identified over 90 contaminants of concern in the various environmental media and uranium as the predominant

contaminant. The following is a brief discussion of the findings of the RI as it pertains to soil at the site. Additional information on soil contamination and the findings for other media are available within the Remedial Investigation Report for Operable Unit 5.

Contamination of surface and subsurface soils occurs within and beyond the FEMP property boundaries. The highest concentrations of uranium in surface soil were found in the former production area at the location of the scrap metal pile (greater than 8000 parts per million [ppm]). Contamination in subsurface soil appears limited to the FEMP property with levels of uranium, up to a hundred times background levels, found in soil at depths as great as 20 feet. Some of the highest subsurface contaminant levels (greater than 400 ppm of total uranium) were found near the former processing facilities where acidic uranium solutions were handled in large quantities.

Concentrations of approximately 20 ppm of uranium (about five times background) were identified in surface soil samples collected off property immediately adjacent to the eastern and northeastern boundary of the FEMP. Uranium was detected at above-background concentrations (generally less than two times background) in a widespread area off the FEMP property; up to 11 square miles of surface soil are projected to have been impacted at these low concentrations. The source of these low concentrations is emissions of dust particles to the atmosphere from plant stacks over the FEMP's 38-year production history.

Radium, thorium, fission and uranium activation products, and inorganic and organic contaminants were also observed in surface and subsurface soils on the FEMP property. The areas affected by these contaminants are localized, with the highest concentrations typically found in association with areas exhibiting the highest uranium concentrations.

TECHNICAL APPROACH TO ESTABLISHING CLEANUP LEVELS

As is the case at many Superfund sites, remediation at the FEMP requires the removal, treatment, and disposal of hazardous source-area materials and the cleanup of environmental media (soil and groundwater) contaminated by the migration of materials from the source areas. There is little dispute over the need to remove, treat, and/or dispose of the source materials themselves; likewise, there is little dispute over the need to restore the Great Miami Aquifer (a protected sole-source aquifer) to full beneficial use, including use as a drinking water supply. Rather, as noted by the Fernald Citizens Task Force in their deliberations, it is the cleanup of the contaminated soil that poses a difficult management problem because: 1) there are large volumes of contaminated material with associated high costs of cleanup; 2) the risk presented by contaminated soil is real but the harm is seldom imminent; 3) the technology for treating contaminated soil is often imperfect; and 4) the materials that are removed during cleanup must be disposed of somewhere and no place is eager to host them.

At the FEMP, the environmental cleanup question can be summarized as: how much contaminated soil must be removed from the site to make it acceptably safe for persons on or near it? The answer to this question is, in turn, driven by two considerations: protection of the groundwater aquifer under the site, and evaluation of risks to persons in contact with the surface soil.

In this section, the major steps in establishing safe, land-use specific, cost-effective cleanup levels for soil are described. From these levels, estimates of the volumes and areal extent of affected soil are derived

for a range of potential risk levels under consideration. The volumes and areas of affected soil serve as the foundation for the development and evaluation of remedial alternatives. They are used throughout the process to judge the viability of remedial technologies and process options, as well as to size and estimate the cost for specific remedial alternatives. To develop cleanup levels that ultimately would achieve regulatory agency concurrence, DOE employed a multistep process (summarized in Fig. 1) that began with the identification of a range of viable potential future land uses for the site, referred to in the FEMP's FS process as land use objectives. For each respective land use objective, the process began with the development of risk- and receptor-based preliminary remediation goals (PRGs) and ended with the identification of preliminary remediation levels (PRLs). PRLs differ from PRGs in their derivation in that PRLs consider the site-specific, naturally occurring background concentrations of the constituents. PRLs also consider analytical limits that affect the ability to detect the constituent in environmental media, and soil-based applicable or relevant and appropriate requirements (ARARs) and to be considered (TBC) criteria that establish maximum regulation-based concentration levels for the constituents in the environment. These PRLs are then used as the contaminant-specific remediation goals to develop and evaluate remedial alternatives for soil. The PRLs are adopted as legally binding final remediation levels for the selected remedy following public concurrence with the Proposed Plan and the issuance of a signed ROD by EPA and DOE. Each of the specific steps comprising this process is described in the subsections that follow.

Fig. 1

Definition of Land Use Objectives and Associated Receptor Scenarios

A range of potential future land uses was used as the foundation for the identification, initial screening, and detailed evaluation of viable remedial action alternatives. The same potential future uses also provided the framework for identifying risk-based exposure scenarios and the hypothetical reasonable maximally exposed (RME) individuals for which land use-specific remediation levels were established.

The land use objectives were developed to take into consideration the progressive deliberations of the Fernald Citizens Task Force. The prevailing land use of the region, residential farming, was used as the point of departure for establishing the following land use objectives:

Land Use Objective 1 examined the viability of returning the entire on-property area to full unrestricted use following cleanup, including the potential for establishing a hypothetical family farm on any portion of the property. For this and all of the other land use objectives, affected off-property areas were examined only in context of the existing land use in the region, residential farming. A hypothetical resident farmer was, therefore; used as the target receptor for both the on- and off-property affected areas. For this receptor, the exposure pathways considered in the setting of soil cleanup levels included: incidental ingestion; dermal contact; direct radiation; fruit and vegetable products; meat and milk products; inhalation of suspended solids; and leaching to groundwater.

Land Use Objective 2 provided for the establishment of an on-property, consolidated management area for contaminated soil, with unrestricted use of all remaining areas of the property. This land use objective considered the potential for establishing a hypothetical family farm, following cleanup, on any portion of the FEMP property outside the area where the contaminated materials are consolidated. A hypothetical

resident farmer was used as the target receptor for the on- and off-property areas outside the consolidation area. For the consolidation area, a hypothetical trespasser is used as the target receptor. For the trespasser receptor, the exposure pathways considered in the setting of soil cleanup levels include: incidental ingestion; dermal contact; direct radiation; and inhalation of suspended solids.

Land Use Objective 3 also provided for the consolidation of contaminated soil in a central area, but restricted potential uses of the remaining areas of the property through the application of institutional controls. This objective considered the potential for establishing recreational, commercial/industrial, or undeveloped open space on any portion of the FEMP property outside the area where the contaminated materials are consolidated. For the hypothetical receptors that represent these land uses, the exposure pathways considered in the setting of soil cleanup levels included: incidental ingestion; dermal contact; direct radiation; and inhalation of suspended solids. For the area of consolidation, a hypothetical trespasser receptor was used in a manner similar to Land Use Objective 2.

Land Use Objective 4 provided for minimum consolidation of contaminated soil with access and future use of the Fernald property restricted. This land use objective contemplated maintaining the entire 1050-acre property under restricted access for waste management purposes. For this land use, a hypothetical trespasser was used to guide the development of cleanup levels, similar to the use of this target receptor for the consolidation area designated in Land Use Objectives 2 and 3.

By using the land use objectives approach to formulate remedial action alternatives, decision-makers are provided with a comprehensive but manageable array of alternatives. From this array, decision-makers are provided with the required information from which to evaluate technical site constraints, required administrative controls, and the overall cost implications of moving from totally restricted to progressively less restricted land use possibilities.

Identification of Constituents of Concern

The Operable Unit 5 baseline risk assessment evaluated constituents of potential concern (CPCs) and exposure pathways to ascertain their present and potential future impacts on human health. Not all CPCs identified in the baseline risk assessment pose significant health risks, and many need not be considered in future remedial activities. Contaminants of concern (COCs) are those constituents that remain a concern following evaluation in the baseline risk assessment process. Only those contaminants identified as posing a concern at the site need to be considered in the development and evaluation of remedial alternatives. The purpose of restricting the number of COCs is to focus on the contaminants that require implementation of remedial actions to ensure the protection of human health and the environment.

The National Contingency Plan establishes a point of departure for acceptable risk as one in a million (10-6) for carcinogenic compounds, including radionuclides. The acceptable limit for noncarcinogenic effects is a hazard index (HI) of 1.0. A HI of greater than 1.0 is considered indicative of a potential toxic effect. However, because multiple contaminants are considered, the screening point for selection of COCs for the FEMP was set at an incremental lifetime cancer risk (ILCR) of 10-7 and an individual HI of 0.1 to the hypothetical on-property farmer to ensure no significant COCs were ignored. Any contaminant with a risk

level or HI less than this screening point is not considered further. For soil contaminants, this screening point considered both direct exposure to contaminated soil as well as the potential impact to groundwater through cross-media pathways.

Using this screening process, 89 soil-based COCs were identified at the FEMP site. Based on the site's uranium-processing history, uranium was found to be the primary COC with the remaining soil COCs generally falling within the concentration-based contamination envelope represented by uranium.

Risk-Based Preliminary Remediation Goals (PRGs)

For each of the COCs discussed in the previous subsection, land-use-scenario-specific PRGs were calculated for each target receptor, using a target HI of 0.2 for noncarcinogenic effects and/or the selected target risk for carcinogenic effects (ILCRs of 10-6, 10-5, and 10-4). The risk-based PRGs were calculated using the equations and parameters for all exposure pathways as detailed in the site's EPA-approved Risk Assessment Work Plan Addendum. The PRGs that are calculated through the process yield health-based contaminant concentration levels for surface contact-related exposure pathways that are protective at each of the target risk levels considered.

Screening of PRGs to Ensure Protection of Groundwater

For purposes of reducing the number of target risk levels and associated risk-based PRGs requiring consideration in the development of remedial alternatives, a screening process was adopted for affected soil. The premise behind this screening process was to determine the maximum uranium concentration that could reside within the soil and still ensure the continued protection of the Great Miami Aquifer (i.e., for a performance period of up to 1000 years into the future, as required by the federal Uranium Mill Tailings Radiation Control Act). For this screening process a cross-media protectiveness goal was adopted to ensure that groundwater concentrations of uranium resulting from the leaching of soil constituents to groundwater do not exceed drinking water quality requirements following completion of remedial actions.

Using a one-dimensional groundwater solute transport model (ECTran) and average hydrogeologic conditions at the site, a maximum soil total uranium concentration of 154 ppm was calculated as the upper-bound value above which undesirable groundwater impacts would be anticipated. This screening-level cross-media-based PRG (i.e., "CPRG") thus represents the upper limit from which to assess the protectiveness of the risk-based PRGs calculated in the previous step. All PRGs with a higher value than the screening-level CPRG would be dropped from further consideration.

To illustrate the results of risk-based PRG development and CPRG screening, the following Table I summarizes the risk-based soil PRGs for uranium for each of the receptor scenarios under consideration. The land-use specific, risk-based PRGs that fall in the shaded area of the table exceed the screening-level CPRG of 154 ppm, and thus would not be expected to be protective of groundwater at the FEMP site (and are therefore eliminated from further consideration).

Table I

Development of Cross-Media PRGs

Following the initial screening process, a more detailed, location-specific analysis was conducted to further evaluate the potential for cross-media impacts, including impacts to media other than groundwater. Cross-media impacts occur when contaminants from waste or an

environmental medium, such as soil, are transported into another medium and result in the potential for secondary exposure to a receptor. When this occurs, receptors can be exposed to these contaminants by an exposure pathway indirectly related to the contaminant source.

The PRGs that passed the groundwater CPRG screening in the previous step were evaluated further using location-specific modeling that considered actual (rather than average) hydrogeologic conditions present within 125- by 125- foot grids across the 1050-acre FEMP property. The detailed evaluation also considered the location-specific potential for contaminants to enter the air and surface water resources as well as groundwater. Reverse-modeling fate and transport simulations were used to ascertain the concentration in the source medium necessary to yield the critical concentration in the receptor medium over a 1000-year performance period.

The results of the simulations were used to further screen the risk-based PRGs to those that are fully protective through both direct contact and indirect (i.e., cross-media) exposure routes. To facilitate the development and presentation of PRGs for soil which could be implemented in the field as part of a remedial action, the mapping of the common physical attributes of the FEMP property discussed above were simplified into three zones, established on the basis of similarities in the hydrogeologic and geochemical characteristics of the soil. The most restrictive physical and geochemical conditions and the controlling transport pathway within each of the individual zones were applied to the entire zone for each individual COC. The derived CPRGs for each of these zones were then arrayed and the most restrictive value identified for each COC was considered. Finally, for uranium, the simulations also considered the varying leaching potentials of the several geochemical forms of uranium that exist in the FEMP environment. The limiting values derived from the evaluations were then used in the development of modified PRGs that fully consider cross-media impacts to groundwater, surface water, and air.

For uranium, the principal COC at the site, the results of the detailed CPRG evaluations indicated the need to further adjust downward the risk-based PRGs developed in the previous step. The simulations indicated that in those areas where more-leachable uranium species are present (primarily in the 135-acre former processing area at the site), a maximum allowable soil concentration of 20 ppm total uranium is necessary to fully protect the Great Miami Aquifer over the full duration of the 1000-year simulation period. In the remaining areas of the site where less-leachable uranium species are present, a maximum allowable soil concentration of 100 ppm total uranium is necessary to fully protect surface water resources in the site area, and ultimately to protect the aquifer from surface water infiltration. Therefore, the 20 ppm and 100 ppm CPRG values provide thresholds that the risk-based PRGs cannot exceed and remain protective of the aquifer.

Identification of Chemical-Specific ARARS and TBCs

CERCLA does not provide for one set of cleanup criteria for universal application to waste sites, but requires that sites attain, or seek a waiver of, federal and state environmental laws and regulations (i.e., ARAR), and meet the intentions of other pertinent considerations (TBCs). Therefore, in addition to meeting the risk-based remediation levels established for each land use objective, all the viable alternatives must satisfy ARARs specified in federal and state environmental laws and

regulations. Over 100 individual ARARs and TBCs were identified that affect the design and implementation of the cleanup at the FEMP. However, the primary ARARs for soil are:

State of Ohio siting criteria for solid waste disposal facilities

Resource Conservation and Recovery Act requirements for treatment of contaminated media and the design of engineered containment facilities

State of Ohio rules for control of particulate emissions and dust

Uranium Mill Tailings Radiation Control Act regulations regarding the management of materials at inactive uranium processing facilities.

Most of the identified requirements address the design and execution of the remedial alternatives, rather than specifying specific concentration-based cleanup levels for soil. The ARARs also govern the handling of residual materials that may be generated during treatment processes.

Establishment of Modified PRGs

Modified PRGs represent an intermediate product in the derivation of PRLs. They are established for each COC by comparing, for the land use scenario and risk level of interest, the risk-based PRGs with available ARARS/TBCs and the appropriate CPRGs, and then selecting the lowest of the values. At this juncture, the lowest value is termed a modified PRG and is carried forward to the next step.

Establishment of COC Background Levels

For each of the naturally occurring and anthropogenic COCs that are present at the FEMP, the 95th percentile of the background distribution of the COC in environmental media was determined through a statistical analysis of contaminant concentration data gathered as part of the Operable Unit 5 RI. These background concentrations were used in the development of PRLs primarily when the modified PRGs fell below the background concentrations.For uranium, the FEMP's key COC, a 95th percentile background concentration in soil of 3.7 ppm was established.

Analytical Detection Limit Considerations

The final element in the development of PRLs was the establishment of the lowest reasonable and achievable analytical detection limits for the 89 soil COCs. These detection limits were used in the PRL development process for those COCs with modified PRGs that fell below analytical detection limits. The detection thresholds were based on experience at the FEMP regarding actual instrument detection limits reported by subcontract laboratories for requested analyses at analytical support levels C and D. For soil, a 25 percent moisture content was assumed in the detection level development; for sediment, a 60 percent moisture content was assumed.

Development of PRLs

PRLs differ from modified PRGs in that PRLs consider the practicality of obtaining and verifying the attainment of a remediation goal. This differentiation is important to allow the development of cost-effective alternative remedial actions.

PRLs for nonradiological COCs were developed in a two-step process. First, all modified PRGs were reviewed against the routinely achievable analytical detection limits. For PRGs below this limit, the analytical detection limit was substituted as the PRL. Next, the modified PRGs were compared to background concentrations in the local environment. In the event the modified PRG was less than the 95th percentile of the background distribution for that constituent, the PRL was considered indistinguishable from background concentrations and the target PRL was set at the 95th percentile background value.

Based on EPA Region 5 policy, a slightly altered approach to developing PRLs for radiological constituents was adopted. First, the 95th percentile background concentration was added to the modified PRG. This value was then compared to the analytical detection limit and the higher of the two values was adopted as the PRL. In two instances background was not added to the modified PRGs for radiological COCs to derive PRLs: if the modified PRG was based directly on an ARAR/TBC or if the modified PRG was based upon a CPRG derived on the basis of attaining an ARAR/TBC in the aquifer.

Estimation of Excavation Area Footprints and Volumes of Contaminated Soil

In order to estimate the volume of contaminated soil at the FEMP site requiring excavation, a solid block model of the top 30 feet of soil was developed. The model consisted of a three-dimensional representation of the FEMP extending to a depth of 30.5 feet. The total model volume was divided into discrete volumes, or solid blocks. Subsurface blocks represented a volume of soil 125 feet by 125 feet by 1 foot deep. Surface soil blocks were 6-inches deep to support a more refined estimate of contaminated soil at shallower depths where contamination is more prevalent.

The solid block model was based upon the results of soil samples collected from various locations and depths across the FEMP site. These sampling results provided uranium concentrations only at the point from which the samples were collected. A geostatistical analysis technique known as kriging was used to establish contaminant concentrations between sampling locations at the center of each model block.

The kriging program employed an ellipsoidal search, using a distance of 16 feet in the vertical direction and 275 feet in the horizontal direction. In other words, when estimating the concentration of uranium within a block, the model searched 16 feet in the vertical direction and 275 feet in the horizontal direction for sampling points with which to establish a spatial relationship for calculation of the contaminant concentration within a block. If no sampling points were found within the search ellipsoid, no estimate of concentration was made for that block.

The resulting uranium concentrations from kriging the solid block model were used to estimate the soil volumes above the maximum contaminant level that require excavation. Furthermore, since the average concentration in each block was known, the excavated soil could be classified as to its ultimate disposition.

Proposed remediation areas (referred to as footprints) and volumes of affected media were estimated for those actions required to achieve each of the four land use objectives over a range of potentially viable PRLs. The PRLs considered under each of the land use objectives were developed to bound the range of potential cleanup levels deemed practical for the site. Volume estimates were performed for a total of nine cases.

A summary of the relationship between uranium soil concentration and affected soil volume is presented in Fig. 2.

Fig. 2

FORMATION AND EVALUATION OF REMEDIAL ALTERNATIVES

There were many remedial technologies and process options initially considered for the cleanup of each of the affected media at the FEMP site. Arraying these process options together produced in excess of 2000 remedial alternatives that could be applied at the site. Using the four land use objectives as a guide, 10 viable remedial alternatives were identified from the long list for further consideration in the initial

screening step of the FS. The alternatives were first compared with one another to identify meaningful differences and then evaluated with respect to implementability, effectiveness, and cost. Only the alternatives judged as most promising on the basis of these evaluation factors were retained for further consideration and analysis. The screening process resulted in the selection of seven remedial alternatives that were sufficiently distinct, yet potentially implementable and effective. Each of the seven alternatives, along with the no-action alternative, is listed below (the number accompanying the alternative corresponds to its land use objective):

No-Action Alternative This alternative was retained to provide a baseline for comparison in accordance with regulatory requirements.

Alternative 1 Excavation and Off-Site Shipment - Under this alternative, soil with contamination exceeding final remediation levels would be excavated and shipped to an off-site licensed disposal facility. Excavated areas would be regraded to reach a predetermined final surface grade that would allow for use of the property as a family farm. Two differing remediation levels were considered; the first case had as an objective the protection of future receptors (in this case a hypothetical on- and off-property farmer) at an ILCR of 10-6 and a HI of less than 1.0. The second case was designed to provide protection to these same receptors at a 10-5 level and a HI of less than 1.0. This alternative would result in the excavation and off-site disposal of 9.6 million cubic yards of soil (10-6 risk level) at a present worth cost of $4.2 billion, and 2.7 million cubic yards (10-5 risk level) at a present worth cost of $1.1 billion. At the 10-6 risk level, approximately 11 square miles of off-property farmland would be disturbed for remedial purposes, and approximately 1 square mile at the 10-5 risk level.

Alternative 2A Engineered Disposal Facility - Under this alternative, a consolidated waste management area would be established and the remaining areas of the property would be made available for unrestricted use. Contaminated soil exceeding final remediation levels would be excavated and placed in an engineered above-grade disposal facility. The facility would be situated in an on-property area displaying the best available geologic conditions. Contaminated soil not meeting waste acceptance criteria for the facility would be shipped to an off-site licensed disposal facility, unless a more economical technology emerged that was deemed more prudent to apply to this soil to attain the acceptance criteria. As in Alternative 1, two different remediation levels were considered for the area outside the disposal facility and for the off-property area: ILCR levels of 10-6 and 10-5 for a hypothetical on- or off-property farmer, and HI values less than 1.0. For all COCs, the waste acceptance criteria for the disposal facility were set at values that would protect neighboring populations and the drinking water quality of the Great Miami Aquifer for a performance period of up to 1000 years. This alternative would result in the excavation and disposal of 9.6 million cubic yards of soil (10-6 risk level) at a present worth cost of $2.1 billion, and 2.7 million cubic yards (10-5 risk level) at a present worth cost of $560 million. At the 10-6 risk level, approximately 11 square miles of off-property farmland would be disturbed for remedial purposes, and approximately 1 square mile at the 10-5 risk level.

Alternative 2C Consolidation with Off-Site Shipment - Under this alternative, contaminated soil exceeding remediation levels would be excavated and, depending on contaminant concentration levels,

dispositioned either in an on-property earthen-covered, revegetated consolidation area or at an off-site licensed disposal facility. Two risk and cleanup levels, consistent with the receptor scenarios of Alternative 2A, were evaluated for this alternative. The waste acceptance criteria for the consolidation area would be established to ensure protection of neighboring populations and the underlying Great Miami Aquifer, and the consolidation area would be managed as an off-limits area to the public. This alternative allows a direct comparison of the cost of off-site shipment to the cost of on-site disposal in an engineered disposal facility (Alternative 2A). This alternative would result in the excavation and disposal of 9.6 million cubic yards of soil (10-6 risk level) at a present worth cost of $4.2 billion, and 2.7 million cubic yards (10-5 risk level) at a present worth cost of $750 million. At the 10-6 risk level, approximately 11 square miles of off-property farmland would be disturbed for remedial purposes, and approximately 1 square mile at the 10-5 risk level.

Alternative 3A Engineered Disposal Facility - This alternative is identical in concept to Alternative 2A, except the area outside the disposal area footprint is made available for restricted (nonresidential and nonfarming) land use. The alternative considers use of the on-property area for commercial/industrial, developed park, and undeveloped park land uses, and a 10-6 risk level for these on-property, nonfarming land uses was used to guide the analysis of this alternative. For the off-property area, two risk levels were considered: an ILCR of 10-5 for the residential farmer (consistent with Alternatives 1, 2A, and 2C above) and an ILCR of 3.5 x 10-5 for the residential farmer, which corresponds to a HI set at its maximum permissible value of 1.0. This alternative would result in the excavation and disposal of soil ranging from 2.4 million cubic yards (industrial land use paired with a 10-5 ILCR for the off-property residential farmer) at a present worth cost of $530 million, to 1.8 million cubic yards (undeveloped park land use paired with a HI of 1.0 for the off-property residential farmer) at a present worth cost of $420 million. At the 10-5 risk level for the off-property area, approximately 1 square mile of farmland would be disturbed for remedial purposes, and approximately 1 acre or less would be disturbed at the HI=1.0 risk level.

Alternative 3C Consolidation with Off-Site Shipment - This alternative is identical in concept to Alternative 2C, except for the changes in land use and the receptor scenarios described for Alternative 3A. The same quantities of soil would require excavation as in Alternative 3A; however, the costs resulting from the need for off-site disposal in this alternative would range from $720 million (industrial land use example) to $610 million (undeveloped park land use example).

Alternative 4A Engineered Disposal Facility - This alternative is identical in concept to Alternative 2A, except the area outside the disposal area footprint is not made available for productive use following remediation; i.e., the entire 1050-acre site is rendered off-limits to the general public. For this alternative, a trespasser receptor scenario (at an ILCR of 10-6) is used to guide the development of cleanup levels. For the off-property area, the same risk levels for residential farming as described under Alternatives 3A and 3C were used. This alternative would result in the excavation and disposal of soil ranging from 2.2 million cubic yards (trespasser scenario paired with a 10-5 ILCR for the off-property residential farmer) at a present worth cost of $450

million, to 1.8 million cubic yards (trespasser scenario paired with a HI of 1.0 for the off-property residential farmer) at a present worth cost of $420 million.

Alternative 4C Consolidation with Off-Site Shipment - This alternative is identical in concept to Alternative 2C, except for the changes described above for Alternative 4A. The same quantities of soil would require excavation as in Alternative 4A; however, the costs resulting from the need for off-site disposal would range from $640 million (using a 10-5 ILCR for the off-property area) to $620 million (using a HI of 1.0 [3.5 x 10-5 ILCR] for the off-property area).

IDENTIFICATION OF LEADING ALTERNATIVE AND SITE-WIDE RISK ANALYSIS

Of the five operable units at the FEMP, Operable Unit 5 is chronologically the fourth to identify and issue a preferred remedy for the site. Each of the operable units is expected to provide a progressive evaluation of the projected site-wide remedy, using the best available information at the time, to predict the acceptability of post-remediation conditions. This projected site-wide remedy incorporates the selected (identified in a ROD), preferred (identified in a Proposed Plan), or leading remedial alternative for each operable unit, as appropriate. The intent of the analysis is to progressively monitor the interfaces among the operable units to ensure that the final site-wide remedy is well thought out, cost effective, and ensures the long-term protection of human health and the environment. The site-wide risk analysis that accompanies the evaluation, termed a Comprehensive Response Action Risk Evaluation (CRARE), also provides for a comprehensive assessment of the impact of multiple carcinogenic and noncarcinogenic compounds, multiple exposure pathways, and the incremental risks due to background levels of contaminants on human health. To conduct the risk analysis of the adopted site-wide remedy, a hypothetical undeveloped park user was the target on-property receptor.

The results of the risk analysis indicate that the adopted site-wide remedy would result in a 90.7 percent reduction in carcinogenic risk to an undeveloped park user of the Fernald property following remediation. Of the carcinogenic risk projected to remain following remedy implementation, 80 percent is due to the presence of naturally occurring background constituents. The estimated residual carcinogenic risk from all constituents and pathways, inclusive of natural background risk, is estimated to be 2.1 x 10-5 following remediation. Similarly, the risk analysis projects a 96.5 percent reduction in noncarcinogenic health effects (i.e., HI) for the undeveloped park user following implementation of the site-wide remedy. Naturally occurring background constituents will account for approximately 69 percent of this residual noncarcinogenic risk. The residual HI from all constituents and pathways, inclusive of natural background contributions, is estimated to be 0.059.

OVERVIEW OF THE SELECTED REMEDY AND CORRESPONDING CLEANUP LEVELS

In conjunction with the Fernald Citizens Task Force recommendations, DOE, EPA, and OEPA selected Alternative 3A, excavation of contaminated soil and placement in an engineered on-property engineered disposal facility, as the preferred remedy for contaminated soil at the FEMP site. This alternative was selected because it provides a remedy that is reliable over the long term, yields the lowest overall short- term risks, is less costly when compared to the other alternatives, and employs proven technologies which are implementable.

During the solicitation of community and stakeholder input for the remedy decision, it became clear that virtually no stakeholders or members of the public were interested in seeing the on-property area of the FEMP site returned to residential farming following remediation. From this basis, and on the recommendations of the Fernald Citizens Task Force, DOE, EPA, and OEPA collectively agreed to adopt Land Use Objective 3 (i.e., the restricted, nonfarming land use objective) for the setting of on-property cleanup levels. Individual constituent PRG values for the undeveloped park receptor were then set at an ILCR of 10-6 and a HI of 0.2, recognizing that at these target values other nonforming land uses (commercial, industrial, developed park, etc.,) would be possible for the site while meeting the corresponding land use-specific risk range targets (1 x 10-4 to 1x 10-6 ILCR and HI=1) considered acceptable by EPA in the National Contingency Plan. PRLs were therefore developed for the selected remedy from this PRG target risk level, using the sequence of steps outlined in this paper. As indicated by the CRARE evaluation, the individual constituent PRLs are fully health protective when considered collectively from a multiple constituent/multiple exposure pathway perspective. These PRLs also protect the Great Miami Aquifer from cross-media transport pathways.

For the affected off-property area, all parties agreed that a residential farming land use scenario should guide the selection of cleanup levels, as this is the predominant land use in the area. It was agreed that the cleanup levels should not exceed a 10-4 ILCR level or a HI of 1 for any site contaminant present outside the FEMP property boundary. Because uranium is considered to be the only site-related constituent in soil that resides outside the property boundary, the cleanup level was set at 50 ppm (inclusive of background), which corresponds to a HI of 1.0 and an ILCR of 3.5 x 10-5. The most striking consideration in selecting this level was the volume of soil that would require excavation beyond the FEMP property boundary if a 1x 10-6 residential scenario were chosen: a total of 5,200,000 cubic yards of soil would be removed and up to 11 square miles of farmland would be disturbed, with considerable loss of vital topsoil. The tradeoffs to achieve a 10-6 risk level were found by all parties to be disproportionate to the benefits achieved. A key ingredient to the stakeholders' understanding of the tradeoffs and benefits of the various cleanup levels under consideration was the highly successful public-forum deliberations and presentations conducted by the Fernald Citizens Task Force.

Summary of Key Accomplishments

The strategy for establishing health-protective soil cleanup levels, as outlined in this paper, has led to a cost-effective, environmentally sound approach to site remediation at the FEMP. Most notably, through the cross-media impact considerations adopted in this strategy, the site's top environmental priority -- the long-term protection of the Great Miami Aquifer -- will be realized, resulting in the unrestricted availability of groundwater from the aquifer for the foreseeable future following the cessation of remedial operations. Recognition and ultimate achievement of this priority remains absolutely critical to maintaining the outstanding public stakeholder support for the remedy that is currently enjoyed by the FEMP.

By shipping the most contaminated soil off site, and keeping the lightly-contaminated materials on site in an engineered disposal facility, the remedy represents a balanced, fair approach to site remediation. It is

estimated that this element of the remedy, in conjunction with the realistic cleanup levels that were selected, results in a cost savings of over $3.6 billion when compared to the cost impacts of adopting the most stringent cleanup levels (i.e., those corresponding to a 10-6 incremental lifetime cancer risk) and adopting a full offsite shipment and disposal alternative. The selected cleanup levels also eliminate the need for significant physical disturbance to off-property wetlands, habitats, cultural resources, natural vegetative communities and cultivated croplands. Over 11 square miles of off-property disturbance to such resources would be required to achieve a 10-6 incremental lifetime cancer risk, which in the view of the Fernald decision team represents only a marginal improvement in an already acceptable set of off-property risks that exist under current conditions. Removal of soil to the 10-6 level would remove tremendous quantities of topsoil from currently productive agricultural lands.

The soil cleanup levels that were established through the process are each individually health-protective, satisfy ARARs, consider the incremental health risks attributable to naturally occurring background concentration levels, and, when considered collectively through all exposure pathways, fall within the acceptable risk range required for CERCLA sites by EPA's National Contingency Plan regulations. By arriving at the selection of these levels in an open public forum, in concert with the deliberations of the Fernald Citizen's Task Force, citizen trust and understanding of DOE's top cleanup objectives and priorities was gained. DOE cannot be successful at Fernald or anywhere else for that matter without the continuing dialogue and understanding that was displayed among the various stakeholder groups during the Operable Unit 5 remedy selection process.

As the final chapter in the effort, the PRLs that were developed through the Operable Unit 5 FS became legally binding final remediation levels in January, 1996, when the ROD for Operable Unit 5 was signed. This ROD brought to completion the 10-year RI/FS process for addressing environmental impacts at the FEMP site, and set in motion a comprehensive remedial design and construction program to aggressively implement the successful remedy decisions reached collectively and cooperatively by the decision team.

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STREAMLINED SITE CHARACTERIZATION APPROACH IN SUPPORT OF REMEDIATION OF U. S. NUCLEAR REGULATORY COMMISSION LICENSED FACILITIES

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ABSTRACT

Under the U.S. Nuclear Regulatory Commission's (NRC's) requirements for decommissioning, radiological and environmental characteristics provide the basis for preparation of an appropriate decommissioning (remediation) plan for decontamination and decommissioning (D&D). Because of reduced Federal resources, a proposed streamlined approach to review site characterization information has been adopted. NRC's 1992 "Action Plan,"

D&D process, and the streamlined site characterization approach are discussed and summarized.

INTRODUCTION AND PURPOSE

NRC's mission is to ensure adequate protection of the public health and safety, the common defense and security, and the environment in the civilian use of nuclear materials in the United States. The main purpose of site characterization and remediation is to achieve part of the objectives of NRC's mission (i.e., protection of the public health and safety and of the environment).

One way to achieve this purpose is to clean up and remediate (to meet the regulatory standards) the sites with radiological contamination before termination of license. Limited Federal resources are an important driving force in NRC's pursuing a streamlined approach for site characterization. The NRC staff presented this approach to the Commission, on May 19, 1995, and to NRC's Advisory Committee on Nuclear Waste (ACNW), on July 27, 1995. On September 28, 1995, the ACNW wrote to NRC's Chairman, identifying "...the initial definition of the scope and magnitude of the contamination problem..." as one of the two signal milestones in the decommissioning of a site (1). The NRC staff agrees that characterizing a contaminated site is one of the two critical milestones in the decommissioning process (2). Recently, NRC Chairman Dr. Shirley Ann Jackson stressed that, as we move into the 21st century, it is imperative that NRC perform decommissioning reviews (which include site characterization) with a cost-efficient and streamlined approach (3).

SITE CHARACTERIZATION IN NRC'S ACTION PLAN

On April 2, 1992, NRC published the "Action Plan" (4) to ensure timely remediation of sites listed in the Site Decommissioning Management Plan (SDMP). The main purpose of the "Action Plan" was to initiate actions to accelerate the cleanup of SDMP sites, with the overall objective of ensuring timely and effective cleanup.

The key elements of the "Action Plan" are:

1) Cleanup Criteria
2) Finality (NRC's decision to terminate a license will relieve the licensee from any further obligation, to NRC, to conduct additional cleanup, as long as the licensee decommissions the site in full accordance with an approved decommissioning plan.)
3) Timing
4) Site Characterization
5) Procedures to Compel Timely Cleanup

As indicated, Site Characterization is one of the key elements of the "Action Plan" pertaining to the SDMP program. NRC issued in July 1992, and then again in November 1994, the draft Branch Technical Position (BTP) on "Site Characterization for Decommissioning" (5). This BTP provides a detailed explanation of various issues connected with the characterization of a contaminated site. According to NRC's "Action Plan," some decommissioning actions have been delayed because of inadequate site characterization information.

DECONTAMINATION AND DECOMMISSIONING PROCESS

Site characterization, which leads to decontamination and decommissioning (D&D), is an important element of the D&D process. To protect the public health, safety, and the environment, D&D needs to be performed at sites with radiological contamination. NRC defines "decommission" in 10 CFR Parts 30, 40, and 70 as "...to remove (a facility) safely from service

and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license" (see 10 CFR 30.4, 40.4, and 70.4).

The D&D planning process starts as a preplanned activity at the cessation of the operation of a facility or at the moment of discovery of a contaminated site. Typically, the following steps or a combination, including site characterization, may be involved in the general D&D process:

Cessation of operation or discovery of contamination (usually by the licensee or by NRC inspectors)

Assessment of radiological status of the site (by the licensee)

Performance of scoping survey (by the licensee)

Determination of necessity for decommissioning (by comparing scoping survey results with guideline values) (by the licensee)

Preparation of site characterization plan (usually including sampling and analysis plan, radiological control plan, health and safety plan, and quality assurance and quality control program) (by the licensee)

Performance of site characterization (including sampling and analysis) (by the licensee)

Preparation of site characterization report (with emphasis on radiological and environmental characteristics, and including sample results) (by the licensee)

Preparation of the decommissioning plan and, sometimes, an environmental report (by the licensee)

Performance of D&D (remediation) (by the licensee)

Performance of the final termination survey and preparation of a final survey report (by the licensee)

Performance of the confirmatory survey and preparation of a confirmatory survey report (by NRC)

Termination of the license and release of the site (by NRC)

In understanding the appropriateness of site characterization in the D&D process, it is important to look into the background information leading to characterization. Contamination might occur inadvertently, accidentally, or because of normal operating processes. Based on the definition and criteria for decommissioning, the licensee or site owner prepares an assessment of the radiological status of the site either at the cessation of operations or at the time of discovery of contamination. The next step is the performance of a scoping survey to provide the basis for initial estimates of the level of effort required for planning the characterization survey and for decommissioning. This typically consists of limited direct measurements (6) (exposure rates and surface activity levels) and samples (smears, soil, water, and material with induced activity) obtained from site locations. The scoping survey provides a preliminary assessment of site conditions or preliminary characterization, relative to guideline values.

Figure 1 shows the general decommissioning process (7).If remediation is required -- based on the scoping survey results and preliminary characterization -- then comprehensive site characterization information is essential for preparing a decommissioning plan. As an example, for "Source Material" licenses, the following rule applies in preparing a decommissioning plan that includes site characterization data:

Section 40.42 (f)(4)(i) - "A description of the conditions of the site or separate building or outdoor area sufficient to evaluate the acceptability of the (decommissioning) plan."

Similar rules are applicable for "byproduct material," under 10 CFR 30.36; for "production and utilization facilities," under 10 CFR 50.82; for "special nuclear material," under 10 CFR 70.38; and for "spent fuel and high level wastes," under 10 CFR 72.54.

SITE CHARACTERIZATION

Under the NRC's requirements for decommissioning, radiological and environmental characteristics of a site provide the background information and basis for the preparation of an appropriate decommissioning plan. The questions to be asked, based on the site characterization information, are: 1) What is the effect of leaving the contamination as it is on the site (no action)? 2) What remediation alternatives for unrestricted use or restricted use levels are acceptable?

To perform efficient and economical D&D, it is important to evaluate the radiological and environmental characteristics of a site. Further steps in the D&D process depend on the reliability and accuracy of the site characterization data. Characterization is the measurement or sampling and analysis required to gather needed information, usually about the type and quantity of contaminants present in or on a material (8). The purpose of site characterization is to determine, on a site-wide basis, the extent, amount, type, and nature of radiological contamination of: 1) surface and subsurface soils (geologic); 2) surface and ground waters (hydrologic); and 3) buildings and equipment. In addition, if decommissioning includes stabilization onsite of residual radioactivity, another key objective of site characterization is to establish the environmental characteristics of the site sufficiently to assess the long-term fate of the residual radioactivity.

Some licensees and site owners, under the SDMP program, made limited or no progress on site characterization (7). Under previous procedures and policies, NRC staff typically had reviewed site characterization plans and site characterization reports to ensure that licensees had established the extent and type of radiological contamination before developing decommissioning plans (7). In November 1992 and November 1994, NRC conducted public workshops, on the SDMP program, that emphasized the importance of site characterization in successful decommissioning. The final rule on "Timeliness in Decommissioning of Materials Facilities" (59 FR 36026) added a requirement to submit characterization data with the decommissioning plan. This approach implements the "Timeliness Rule," thereby increasing reviewer and licensee efficiency. NRC has clearly established and communicated its expectations, both to the licensees and to other responsible parties, regarding site characterization in support of decommissioning.

Previously, various steps were performed sequentially or in series as shown in Fig. 1. This was to ensure that licensees had established the extent and type(s) of radiological contamination before initiating decommissioning (remediation). In typical cases, NRC staff invested approximately one-half to a full person-month of effort (spread out over several months) in reviewing each site characterization plan and report (8). Figure 2 shows the iterative process used in the conceptual flow diagram of the SDMP site characterization (5). Figure 3 shows a general site characterization procedure.

Based on experience to date in reviewing site characterization plans and reports, and in response to reduced Federal resources, the NRC staff developed a streamlined approach for reviewing site characterization

plans and reports (7). The previous approaches had been costly and had resulted in delaying decommissioning. The NRC staff has changed the procedures by not reviewing site characterization information until reviewing the decommissioning plan. The decommissioning plan is required by the rules promulgated in the Code of Federal Regulations, Title 10. It is sufficient to submit the characterization data along with the decommissioning plan, instead of separately submitting the characterization plan and report before submitting the decommissioning plan. This streamlined approach promotes a more coordinated and focused review of site characterization information, because the reviewers will be compelled to emphasize issues that affect the selection and implementation of a decommissioning approach. This alternative is consistent with NRC regulations, which require characterization data to be submitted with the decommissioning plan (7).

Fig. 1

Fig. 2

Fig. 3

The revised approach should promote parallel resolution of characterization and decommissioning issues that, in many cases, are interdependent. However, delaying the review of characterization data may result in later identification of significant information gaps. NRC will partially compensate for this risk by increasing routine contact with licensees through site visits and meetings, and by paying heightened attention to licensees and responsible parties that have lower levels of performance or complex site characterization issues. Resources thus conserved will then be focused on those sites needing increased staff attention, or on other NRC priorities.

NRC staff discussed these changes with such representatives of industry as the Nuclear Energy Institute (NEI) and the Fuel Cycle Facilities Forum (FCFF) (9). NRC staff agreed with NEI and FCFF that there should be early communication of site characterization plans and issues, to minimize the potential for significant data gaps later in the D&D process.

Implementation of this streamlined approach will reduce the expenditure of NRC resources. In some cases, this approach may delay decommissioning, and increase resource expenditure by the licensee or responsible party. For example, if significant data gaps in the characterization data are identified during the review of the decommissioning plan, additional characterization may take more time and resources, because of remobilization of personnel and equipment, to conduct the investigation. Although, in many cases, NRC is no longer going to review site characterization plans, licensees will still benefit from developing such plans for their own uses. A site characterization plan needs to address the nature and locations of: 1) previous operations involving use of radioactive materials; 2) effluent release points; and 3) buildings and equipment used in the operations. In addition, a site characterization plan should focus on the subsurface environment, with emphasis on groundwater impacts, in those situations where loose radioactive materials may be buried or released into the subsurface.

It is the responsibility of the licensee or responsible party to ensure that adequate expertise and resources are devoted to characterization planning and performance. The NRC staff will work closely with the licensees or responsible parties to ensure that they are aware of existing guidance, and to provide timely informal comments to identify significant data gaps (7).

The ultimate purpose and use of the site characterization information is to develop an appropriate decommissioning plan for the site. According to the "Wall Street Journal" dated April 4, 1995 (10), and the weekly report, "Inside E.P.A." (11), just cleaning up the nation's contaminated former nuclear-weapons facilities will cost $230 billion and will take 75 years. In addition, there are non-nuclear-weapon-related sites with radiological contamination. The draft BTP (5) emphasizes the importance of obtaining accurate and relevant characterization data for a site, to save time and money.

Key elements to be considered in a comprehensive site characterization program are (8):

1) Review historical information.
2) Define characterization objectives.
3) Prepare a sampling and analysis plan to meet objectives
4) Conduct sampling and measurement.
5) Review, analyze, and verify data.

The above elements need not be all-inclusive and serve as general guidelines. Site-specific modifications may be required.

NRC, U.S. Environmental Protection Agency, U.S. Department of Energy, and U.S. Department of Defence are developing a "Multi-Agency Radiation Survey and Site Investigation Manual," (MARSSIM) (12), which contains a section on characterization surveys.

Reducing the effort, time, and resources needed to acquire characterization information, while still ensuring adequacy, is the principle behind NRC's proposed streamlined approach. NRC staff will provide sufficient regulatory guidance for the licensees to understand the requirements of the criteria and rules.

CONCLUSIONS

Under NRC's requirements for decommissioning, radiological and environmental characteristics provide the basis for preparation of an appropriate remediation plan.

Based on experience to date in reviewing site characterization plans and reports, and in response to reduced Federal resources, the NRC staff has developed a streamlined approach for reviewing site characterization plans and reports. The streamlined approach to site characterization identifies all contaminated areas and safety hazards, provides information needed to evaluate the most cost-effective remediation options, reduces the risk of having to re-remediate contaminated areas, and minimizes the overall remediation cost. The purpose of the streamlined approach to site characterization is to determine, on a site-wide basis, the extent and type of radiological or chemical contamination. The ultimate purpose and use of the site characterization information are to prepare an appropriate environmental remediation plan for the site, so as to reduce residual radioactivity to a level that allows release of the property for unrestricted or restricted use. Previous approaches have been costly and have resulted in delaying decommissioning. The staff plans to review site characterization information in conjunction with the decommissioning plan. This approach should promote parallel resolution of characterization and decommissioning issues that, in many cases, are interdependent. NRC hopes that this change in approach will highlight issues that affect the selection and implementation of a decommissioning process. However, this approach may result in later identification of significant data gaps. There should be early and constant communication with licensees about

site characterization plans and issues, to minimize the potential for significant data gaps later in the decommissioning process.

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SUCCESSFUL COMPLETION OF A RCRA CLOSURE FOR THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT

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ABSTRACT

This paper discusses the successful completion of a RCRA (Resource Conservation and Recovery Act) closure of a hydrofluoric acid (HF) Tank Car at the Fernald Environmental Management Project (FEMP). The FEMP is a facility owned by the Department of Energy (DOE) that is on the National

Priorities List of hazardous waste sites and is undergoing CERCLA (Comprehensive Response, Liability Act) remediation. The FEMP is also subject to closure under RCRA and Ohio Hazardous Waste rules. The HF Tank Car Closure was conducted by the Fernald Environmental Management Corporation (FERMCO), the contractor responsible for FEMP facility clean up and closure. Through a combination of sound planning and team work, the HF Tank Car was closed safely and ahead of schedule. During over 22,000 hours of field work required for construction modifications and neutralization of some 9,600 gallons of HF and decontamination rinseates, there were no OSHA recordable incidents. The system design avoided additional costs for constructing a new system and subsequent dismantling for disposal or reuse at another facility by maximizing the use of existing equipment and facilities. The successful closure of the HF Tank Car demonstrated the FEMP's commitment to reducing risks and cleaning up the facility in a manner consistent with objectives of RCRA regulations and the Ohio Environmental Protection Agency (EPA) Hazardous Waste Rules. This in turn, facilitated ongoing negotiations with the Ohio EPA to integrate RCRA closure and the ongoing CERCLA remediation activities. This paper addresses why the unit was clean closed under an approved RCRA Closure Plan. The integration of the EPA regulations for RCRA and CERCLA programs and the DOE-Orders impacting design, construction and operation of an acid neutralization system is also reviewed. The paper concludes with a discussion of lessons learned in the process of preparing the closure plan and through final project close out.

INTRODUCTION

FEMP facility clean up and closure actions must be responsive to existing enforcement actions by both the USEPA and Ohio EPA. In July of 1986, the DOE entered into a Federal Facility Compliance Agreement and initiated a preliminary Remedial Investigation and Feasibility Study. In March of 1986, the Ohio EPA filed a complaint against the DOE which ultimately lead to in a Consent Decree signed in December 1988 by DOE and the Ohio EPA. Because there are no tri-party agreements between USEPA, Ohio EPA and DOE, enforcement actions and responses have been separate and independent and the FEMP has conducted dual compliance programs. The challenge facing the DOE and FERMCO (contracted by DOE in 1993 to oversee site remediation) is to establish an approach to site remediation that integrates RCRA facility closure requirements and the CERCLA requirements for remediation of all contamination.

The success of the HF Tank Car Closure project illustrates how a project team can coordinate and integrate multiple support organizations and regulatory requirements to ensure safe, effective and timely remediation. To achieve successful completion of the project, the project team coordinated engineering, construction and operation activities with support services from multiple organizations (e.g., radiation control, safety and health, quality assurance, and etc.). The closure actions were planned and implemented in a manner that integrated the multiple requirements of DOE Orders and USEPA, OSHA, and Ohio EPA rules and regulations.

OVERVIEW OF THE HF TANK CAR CLOSURE PROJECT

The subject of this paper is the completion of RCRA clean closure of the HF Tank Car. The HF Tank Car was a circa 1940 rubber-lined steel rail car containing an estimated 5,000 gallons of 30 percent hydrofluoric acid (HF). Preliminary evaluation of the HF Tank Car determined that the age of the tank car and the highly corrosive nature of HF posed a significant

threat for release. Although the car was apparently in good condition, the age of the car was a cause for concern. Had the rubber liner failed the HF would have readily corroded through the steel tank car and been released to the environment.

HF is highly corrosive to steel and human tissue and is most corrosive at approximately 35 percent concentration. HF is also toxic, even at relatively low concentrations due to the affinity of free fluoride ions to calcium. Upon exposure to HF, workers experience extreme burning sensation in the eyes and respiratory system. However, the full effect on exposure to the skin, particularly at lower concentrations is often not felt for several hours because it is readily absorbed and migrates toward the calcium rich bones in the body. In the process, significant subcutaneous tissue damage can occur, causing gangrene in severe cases. When HF enters the body it also causes an electrolyte imbalance by removing calcium from the blood which can lead to cardiovascular difficulties.

The FEMP elected to remediate and clean close the HF Tank Car under a RCRA Closure Plan because extensive media contamination was not considered likely. The Ohio EPA had also identified the HF Tank Car as a high priority due to concerns for possible release. By completing clean closure under an approved closure plan, the FEMP increased their credibility with the Ohio EPA concerning RCRA closures which should facilitate ongoing negotiations for RCRA\CERCLA integration.

The original closure plan for the HF Tank Car was submitted in May of 1992. Implementation of closure actions were delayed by repeated cycles of Ohio EPA reviews and submittal of revised closure plans addressing Ohio EPA comments. Historically the FEMP had been reluctant to initiate closure actions prior to Ohio EPA approval. In addition, progress was hindered because the groups responsible for writing plans were not responsible for implementing actions. This created confusion and poor coordination of activities.

In February 1994, FERMCO established an integrated project team with a designated project manager and key contacts from various support organizations and DOE site representatives. The function of the project team was to coordinate, plan and schedule the field work for construction (prior to formal approval of the closure plan), and integrate the various regulatory requirements and FERMCO policies and procedures. The resolution of the final Ohio EPA comments required completion of the bench-scale testing and development of the conceptual design (see discussions below). As a result, the final approval of the Closure Plan was not received until July 1994. The proactive approach of the project team made it possible to complete the HF Tank Car closure within the required 180 days after approval.

Key to Success: Involve and integrate support and performing organizations, including DOE, into a project team focused on how to get work done.

The project team identified and worked through the following seven major project phases.

Phase 1 - Bench-Scale Testing

The first stage was to characterize the acid in the HF Tank Car and evaluate treatment options. Samples of the HF were collected and assayed to confirm the basic chemical properties and characteristics of the solution. Four treatment options were identified and evaluated based on health, safety, technical, and feasibility considerations. Lime slurry

neutralization was selected for further testing and evaluation. A Bench-Scale Test Plan was developed to evaluate and confirm the feasibility of lime slurry neutralization, identify process limits, characterize treated waste streams, and provide data to support the design of a full-scale treatment system. The results confirmed that the best results were obtained by adding HF at a controlled rate to a prepared, agitated neutralization slurry consisting of 10 weight% solid using a mixture of 30% calcium carbonate and 70% lime. In addition, analysis of the resulting neutralized solids and liquid wastes indicated they would no longer be RCRA hazardous. Table I lists the physical and chemical properties of the reagents and neutralization products determined by the bench-scale tests.

Key to Success: Stopped making assumptions and collected hard data to characterize waste for treatability and provide data for system design.

Phase 2 - Conceptual System Design

After confirming that lime slurry neutralization was feasible, three alternatives for neutralization and filtration were identified and evaluated for implementability, time required to implement, regulatory and site limitations, and qualitative cost impacts. Each criteria was assigned a relative ranking between a low of 1 and high of 3 and a cumulative score was calculated. The evaluation and selection of alternatives were reviewed and discussed by the project team. This process allowed early identification of interface and coordination issues (e.g., quality assurance issues and interface requirements between the project and FEMP site facilities). In addition, a preliminary hazard analysis was conducted and recommended safety controls were identified for inclusion into the detailed design.

The alternative selected required new pumps, piping and controls to allow use of an existing 1,400 gallon existing tank equipped with solids feeder unit. The existing tank, located within the same general area as the HF Tank Car, had been installed for HF neutralization but had never been used. The neutralized slurry was to be transferred using portable tanks to an existing on-site operation (referenced as Plant 8) for filtration across a rotating vacuum filter drum. The filtered solids would be collected, drummed and sampled at Plant 8, to determine\confirm disposal requirements, and the filtrate would be collected, tested and discharged to the FEMP wastewater treatment system. The conceptual system design avoided additional costs for constructing a new system and subsequent dismantling for disposal or reuse at another facility by maximizing the use of existing equipment and facilities. Figure 1 is the conceptual process flow schematic.

After the conceptual design and functional design requirements were identified, a Conceptual System Implementation Plan (CSIP) for HF Tank Car Closure was prepared and submitted to DOE. The purpose of the document was to provide an integrated discussion of planning, testing and design activities to support the closure of the HF Tank Car. The CSIP was also used to evaluate and document compliance with Ohio EPA regulations, DOE Orders and FEMP policies and procedures.

Key to Success: Integrate preliminary hazard assessment and safety concerns into conceptual design and functional design requirements.

Table I

Phase 3 - Detailed System Design

The detailed system design defined requirements and specifications necessary for construction and use of the HF neutralization system.

Detailed design included specifications and drawings for the modifications necessary to provide a metering pump, water supply piping and batch controller, piping and valves to transfer HF from the tank car to the neutralization tank, piping and valves to transfer neutralized slurry to portable tanks for transport to the Plant 8 for filtration, and minor modification to the portable tanks to provide agitation and hose connections for neutralized slurry loading and unloading. The final design provided additional process control instruments, with interlocks to the HF transfer pump power, to stop HF addition if the temperature, pH, level, or pressure exceeded process design limitations.
Fig. 1
Design reviews were performed by the project team for the initial and final design packages. An independent review was performed on the final design package. After resolving all comments, a final Certified-for-Construction (CFC) Design package was issued. After CFC, all drawings and specifications were controlled and could only be changed through a formal design change procedure.
After CFC and before major construction activities began, a secondary design review was conducted and a revised CFC package was issued. This design review was conducted to ensure consistency with recent changes to the FERMCO engineering procedures and to ensure that design problems that had been recently encountered on another similar FEMP project were not repeated.
Phase 4 - Construction and System Testing
Construction activities were conducted in accordance with the CFC drawings. Configuration Management was maintained to document construction and confirm design requirements were not compromised. Based on the Safety Assessment, a high level of documentation and quality assurance was required. Test plans for construction acceptance testing and system operability testing were prepared and implemented to confirm that the system was constructed and operated in accordance with the design. Although more problems than expected were encountered in servicing existing equipment and instrumentation, construction and testing was completed without incident.
Key to Success: Proper configuration control precludes making field changes inconsistent with the design and operating requirements.
Phase 5 - Operating Procedures and Operator Training
In order to complete the HF Tank Car closure within the required 180 day regulatory limit, it was not possible to wait until the system construction was completed before preparing the operating procedures and initiating operator training. To compensate, the procedures were initially developed based on the engineering specifications and drawings. A selected working group of project team members conducted a series of joint table top reviews. The operator training was set up in modules designed to maximize the ability to defer training on system components until the construction was completed and a field walk through could be conducted. As part of their training, the operators were tasked to walk through and verify the procedures in the field. They were encouraged to provide comment for improvement of the procedures. When construction was completed, the final procedures were completed and incorporated many of the operators comments. The final procedures were reviewed with the operators and a final field walk through was conducted. The operations manager then qualified each operator by oral exam and field simulation. In addition, the operators were involved in the conduct of the final

system operability testing to increase their understanding of how the system functioned. Because of their involvement in the development of procedures and testing of the system, operator acceptance of the procedures and performance during operations were noticeably above average.

Key to Success: Enhancement of operator acceptance and performance through their involvement in developing procedures and testing and evaluating system operations.

Phase 6 - Operation and Processing

The neutralization of acid and decontamination rinseate was initiated on June 12, 1995 and completed on July 21, 1995. Within days of the system start up, operations personnel identified opportunities for improvement which decreased the time required for slurry preparation. The 9,600 gallons of acid and rinseate neutralized included 1,082 gallons more HF and 2,712 more rinseate than originally estimated. Despite the increased volumes, expedited processing was achieved and neutralization was completed 3 weeks ahead of schedule.

On at least 3 occasions, potential problems were identified and averted by Operations personnel based on system walk downs and inspections prior to initiating HF transfer (as required by the Operation's Standing Orders prepared to comply with DOE Conduct of Operations).

Phase 7 - Project Close Out

There were two main elements for project close out. First, completion of all field work including the utility isolation and shut down of the HF neutralization system, dismantling and staging the decontaminated rail cars, release of the portable tank for other uses in support of site remediation, and staging drummed filter solids for shipment to the DOE Nevada Test Site (NTS) disposal facility. Second, documentation of work completion including certification by an independent, qualified, registered engineer to complete RCRA closure requirements; compilation of final costs, including, final accruals for all labor, subcontractors and materials charges, identify those documents (e.g., standing orders for operation, operating procedures, health and safety plans, and etc.) that are inactive and not applicable to ongoing work, and closure of associated charge numbers.

All processing of HF and decontamination rinse waters were completed by the end of August. The activities for facility isolation and shut down, dismantling of the rail cars, staging of the dismantled rail cars components for a future project to complete recycling or disposal, characterization and staging of drummed filter solids pending shipment to NTS, and release of the decontaminated portable tank for on-site use were completed before the end of September (i.e., within the 1995 Fiscal Year). The RCRA certification report was submitted October 3, 1995. The Ohio EPA issued approval of the closure certification report on November 27, 1996 and closure of applicable charge numbers and site documents have been completed.

SUMMARY AND CONCLUSIONS

The highlighted keys to the HF Tank Car Project success are applicable to most any project and illustrate ways to improve project performance. In summary:

A team approach improves the ability to identify, schedule and address tasks and issues before they become obstacles and cause delays. Keep a focus on the work to be done.

Don't make assumptions, characterize waste for treatability not just for hazardous waste determinations.

Define and verify process chemistry and operational requirements to be addressed in design of treatment system.

Integrate safety requirements into system design, as early as possible.

Once the design has been certified-for-construction, configuration management is essential. Without a detailed understanding of how the design components interact, minor changes can have major impacts.

Involvement of operating personnel in the process of developing procedures and systems testing enhances operator acceptance and performance.

The establishment of a project team focused on how to get the work done represented a fundamental change in approach. Focusing on planning and design based on technical requirements to implement the work is critical. It is all too easy to fall into the trap of trying to design work to fit EPA regulations or DOE Orders. Regulations and orders define what you need to do, not how. Limited progress was made on HF Tank Car project between May 1992 and February 1994. However, once the project team focused on determining the technical requirements the project began to move forward. The most direct indicators of the effectiveness of the project team were the safety record and expedited processing of HF. Without compromising safety, the neutralization of HF and decontamination rinseates was completed 3 weeks ahead of schedule and included processing 1,082 gallons more HF and 2,712 gallons more rinseate than scheduled. There were no OSHA reportable incidents during the 22,000 hours of field work required for construction modifications and neutralization of the 9,600 gallons of HF and decontamination rinseates.

48-4

LACK OF CONSISTENCY IN STATE RCRA PROGRAMS: A THREAT TO REGULATORY REFORM AFFECTING DOE

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ABSTRACT

The Resource Conservation and Recovery Act (RCRA) regulatory reform process is threatened by the failure of the Environmental Protection Agency (EPA), States, and the regulated community to address the contradiction between:

the requirement that RCRA regulations in authorized State programs be consistent with the Federal program and other authorized State programs, and

the prerogative of States to promulgate RCRA regulations that are more stringent and broader in scope than the Federal RCRA regulations. Assigning primacy to stringency, which has been EPA's practice in recent RCRA rules containing regulatory reform elements, appears to ignore the legislative history of RCRA, EPA's own regulations, and relevant court decisions. Failure to address this problem jeopardizes the possibility that certain forms of the Department of Energy's mixed waste may qualify for disposal at radioactive waste disposal facilities.

As a first step, this paper recommends that the contradiction between consistency and stringency be addressed in regulatory preambles and other appropriate forums. If the contradiction is not addressed, RCRA's

ultimate goal, protection of human health and the environment, may be compromised.

LEGISLATIVE BACKGROUND

The Resource Conservation and Recovery Act (RCRA), like other environmental laws, authorizes States to administer their own programs under the statute. Under section 3006(b) of Subtitle C of RCRA (42 U.S.C. 6926), Authorized State Hazardous Waste Programs, State hazardous waste programs must be:

equivalent to the Federal program,

consistent with the Federal program and with State programs applicable in other States, and

amenable to adequate enforcement of compliance.

State programs that fail to exhibit these attributes cannot be authorized. Once authorized, a State program can be withdrawn if it is not administered and enforced in accordance with RCRA Section 3006 requirements.

These provisions of section 3006 date from the original RCRA legislation, Public Law 94-580, passed in the House and Senate in September of 1976. The legislative history of the Act shows that the purposes of the Federal/State partnership were to:

provide uniformity among the States as to how hazardous wastes are regulated,

prevent some States from becoming the dumping ground for hazardous wastes, and

utilize State enforcement personnel to implement the provisions of the Act. (1)

Section 3009 of Subtitle C of RCRA (42 U.S.C. 6929), Retention of State Authority, originally provided that after the Federal RCRA program becomes effective, no State may administer a program less stringent than the Federal program. Neither the statutory language nor the legislative history addressed the question of whether more stringent State requirements were preempted by Section 3006.

Section 3009 was amended in 1980 by Public Law 96-482, which added the following sentence.

> Nothing in this title shall be construed to prohibit any State or political subdivision thereof from imposing any requirements, including those for site selection, which are more stringent than those imposed by such regulations. (1)

The amendment was introduced by Senator Bumpers of Arkansas, whose speech supporting the amendment on the Senate floor referred to:

the incident at Love Canal,

Government Accounting Office (GAO) findings that the hazardous waste regulatory program could not adequately safeguard public health and the environment,

Environmental Protection Agency (EPA) estimates of a 4 to 6 percent annual increase in the amount of hazardous waste being generated,

EPA projections of the siting of increased numbers of commercial hazardous waste treatment and disposal facilities, and

the possible location of a hazardous waste disposal site near a small community in his home state (1).

At the time, there was no discussion or recognition of the possible contradiction between requiring State programs to be consistent with the Federal program and allowing them to be more stringent. Consequently, the Act did not confer primacy on either consistency or stringency.

In the House debate preceding the passage of the Hazardous and Solid Waste Amendments (HSWA) of 1984, the meaning of equivalence was discussed. Congressman Florio of New Jersey explained that equivalence did not mean point-by-point equivalence.

State requirements should be equivalent in overall effect to the Federal program, without the necessity of showing point-by-point equivalence. (1)

In addition, he stressed the importance of bringing State hazardous waste experience to bear in the RCRA Subtitle C program, instead of removing this experience from consideration, and noted that several of the most important RCRA amendments being considered by Congress originated in State programs (i.e., the land disposal restrictions and the regulation of small quantity generators). (1)

Congress believed that it was important for the changes contained in HSWA to take effect as soon as possible. Consequently, section 3006(g) was added to the law to provide that EPA regulations promulgated pursuant to these amendments take effect in authorized State programs on the same date that such requirements take effect in other States. (1)

REGULATORY BACKGROUND

Since 1981, EPA regulations implementing sections 3006 and 3009 of RCRA have addressed the potential contradiction between stringency and consistency by clearly assigning primacy to consistency. To obtain approval, a State program must be consistent with the Federal program and with State programs applicable in other States (40 CFR 271.4). In particular, the following aspects of State programs are deemed to be inconsistent:

  any aspect of the State program which unreasonably restricts, impedes, or operates as a ban on the free movement across the State border of hazardous wastes from or to other States for treatment, storage, or disposal at facilities authorized to operate under the Federal or an approved State program (40 CFR 271.4(a));

  any aspect of the State law or the State program which has no basis in human health or environmental protection and which acts as a prohibition on the treatment, storage, or disposal of hazardous waste in the State (40 CFR 271.4(b)); and

  a manifest system that is inconsistent with the manifest system in the generator standards in 40 CFR 262 (40 CFR 271.4(c)).

Under 40 CFR 271.1(i), States may:

  adopt or enforce requirements that are more stringent or extensive than the Federal requirements, or

  operate a program with a greater scope of coverage than the Federal requirements.

However, 40 CFR 271.1(i) has always required that more stringent, more extensive, and broader requirements be consistent with the Federal program and other authorized State programs. After the passage of HSWA, EPA added paragraph (j) to 40 CFR 271.1. Paragraph (j) lists the HSWA requirements that take effect in authorized State programs on the same date that such requirements take effect in other States.

The implementation of HSWA and other amendments to RCRA in the mid-1980s to early 1990s generally caused RCRA regulations to become more stringent and broader. Most of the deregulatory or less stringent changes of those years, such as the satellite accumulation provisions of 40 CFR 262.34 and the corrective action management unit/temporary unit (CAMU/TU) rule, failed to raise significant consistency issues. During this period, the authorization of State programs was generally focused on whether the

programs were equivalent to, and at least as stringent as, the Federal program.

JUDICIAL BACKGROUND

In June 1987, North Carolina enacted a law that prohibited commercial hazardous waste treatment facilities from discharging waste water into surface water upstream from a public drinking water supply intake unless a dilution factor of at least 1000 at the point of discharge existed. Upon initial review of the law, EPA determined that the law was inconsistent with RCRA and began proceedings to withdraw North Carolina's authorization. After the administrative law judge (ALJ) hearing the case decided that the law was not inconsistent with RCRA, EPA accepted the recommendation of the ALJ and did not withdraw the program.

Laidlaw Environmental Services, Inc. and the Hazardous Waste Treatment Council petitioned the courts for a review of EPA's decision. In 1991, the United States Court of Appeals for the District of Columbia upheld EPA's decision and denied the petition for review. (2)

Among other things, the Court held that

Because the EPA is charged with the administration of RCRA, we defer to its interpretation whenever the statute is silent or ambiguous with respect to a specific issue. So long as the agency's interpretation is reasonable and consistent with the statutory purpose, we must uphold it.... While it is true that RCRA requires 'consistency' between state and federal programs, it does not mandate uniformity; in fact, the statute expressly reserves to States the authority to impose site-selection standards that are 'more stringent' than those imposed by federal authority. Consequently, the Regional Administrator's ruling that the North Carolina statute could be found 'consistent' with federal law...was not in conflict with the federal statutory scheme. (2)

In other cases, however, the provisions of more stringent State laws and regulations that have the potential to threaten the consistency of the national hazardous waste program have been struck down by the courts. Such provisions have involved the imposition of higher waste management fees on wastes generated out of state, or of more stringent permit conditions on facilities that accept waste originating out of state. These challenges have generally been won on the basis that the provisions violate the Commerce Clause of the U.S. Constitution, which denies States the power to regulate interstate commerce.

In one of the most important cases of this kind, the Hazardous Waste Treatment Council took action against the State of South Carolina and the South Carolina Department of Health and Environmental Control for promulgating of regulations that would:

- restrict the transfer of hazardous wastes to South Carolina facilities from States which have not reciprocated by providing treatment and disposal capacity for hazardous wastes within their borders,
- impose quotas on the quantities of such wastes otherwise permitted, and
- create a mandatory preference for wastes generated in South Carolina. (3)

The plaintiffs contended that these laws conflicted with and were preempted by the Federal hazardous waste management regulatory program, thus violating the Supremacy Clause of the U.S. Constitution and federal regulations prohibiting such laws (40 CFR 271.4). (The Supremacy Clause of the Constitution establishes that the laws of the United States take precedence over State laws.) The plaintiffs also raised concerns

regarding the constitutionality of the requirements under the Commerce Clause of the Constitution. (3)

The defendants countered that the laws, Executive Orders, and regulations in question did not conflict with the U.S. Constitution's Commerce Clause because:

South Carolina's hazardous waste management program had received EPA approval, and

Congress' intent was for each State to be permitted to implement its own hazardous waste program. (3)

The defendants cited Section 3009 of RCRA as indicating that States cannot be prohibited from imposing requirements more stringent than the Federal standards. (3)

The U.S. District Court granted injunctive relief based on the challenges raised under the Commerce Clause, and it offered the opinion that the plaintiff could also prevail in possible future actions under the Supremacy Clause of the U.S. Constitution. (3)

RCRA REGULATORY REFORM

In recent years, RCRA implementation efforts have focused on reform and rationalization. The success of these efforts may further advance environmental protection by making specific and general improvements to the RCRA regulatory structure. Specific improvements include changes to remove the impediments to beneficial recycling of industrial hazardous wastes and to improve the voluntary collection of hazardous wastes generated by households. General improvements include changes to improve the allocation of limited social resources, focusing these resources on wastes posing the greatest risks to human health and the environment. Table I lists recent proposed and final RCRA rules that contain regulatory reform elements.

Table I

EPA's recent analyses of the applicability of these regulations in authorized States can be reduced to a formula, presented in Table II.

Table II

According to the formula, authorized States are never required to adopt less stringent rules. More stringent rules must always be adopted by authorized States; they become effective immediately if they are HSWA rules, or when the State amends its RCRA program to incorporate them if they are non-HSWA rules.

The unquestioning application of this formula fails to raise important issues associated with consistency and the potential conflict between consistency and stringency. In all of the regulations listed on Table I with the exception of the proposed munitions rule, no analysis is performed to determine if a State's failure to adopt a Federal RCRA regulation change that is less stringent renders the State program inconsistent with the Federal program and with other authorized State programs.

Other important stringency/consistency issues raised by the proposed and final rules on Table I are listed below.

If a less stringent RCRA rule improves overall environmental protection, should it be considered less stringent?

Should program improvements relating to the elimination of outdated, confusing, and unnecessary language be evaluated in terms of stringency?

Is allowing waste to be treated using an additional treatment method less stringent if the method provides at least equivalent protection of

human health and the environment as the methods currently authorized? What if a new treatment method provides superior protection?

Is it more important to evaluate stringency or consistency when changing a definition, such as the definition of on-site, if the change affects large numbers of RCRA facilities?

If authorized States do not amend their rules to include equivalent and consistent provisions, does the Hazardous Waste Identification Rule address the petition for rulemaking submitted by the Chemical Manufacturers Association, the recommendations made by the Dialogue Committee on Hazardous Waste Identification, and the court decision in Shell Oil v. EPA?

In the munitions rule, EPA recognizes that the failure of authorized States to adopt the rule as proposed would prevent the rule from achieving its intended goal.

Today's proposal raises an issue regarding State authority because Congress clearly expected EPA to develop national standards for waste munitions through the RCRA rulemaking process. Although today's rule would lay out such national standards, States under the standard RCRA approach could enforce their own more stringent standards under their own State programs. This situation, at least in theory, could lead to just the sort of piecemeal approach that the FFCA was intended to avoid (60 FR 56488; November 8, 1995).

The munitions rule also recognizes the Department of Defense's (DOD's) need for national consistency in managing waste munitions, given DOD's national defense mission, nationwide presence, and logistical and operational needs.

Unfortunately, however, the preamble to the proposed munitions rule fails to address the contradiction between consistency and stringency in sections 3006 and 3009 of RCRA, and instead suggests an alternative approach to achieving consistency. (This approach relies on an argument that the scope of the waiver of sovereign immunity in section 6001 of RCRA does not permit a State to impose more stringent requirements than those contained in the Federal regulation if the requirements would result in unfair discrimination against Federal agencies. Because military munitions are unique to the military, EPA argues that more stringent regulations would thus be discriminatory. This paper will not comment on the merits of this alternative approach except to note that if it were adopted, it would reach far beyond military munitions and affect hazardous waste items that are unique to the Department of Energy (DOE) and to other Federal agencies.)

Continued failure to address contradictions between stringency and consistency when new regulations are proposed and promulgated ignores:

the legislative history of RCRA, which shows that Congress intended the law to provide for uniform regulation of hazardous waste, but allowed States to promulgate more stringent requirements because the Federal program was weak and the amount of hazardous waste being generated annually was increasing;

EPA's own regulations, which assign primacy to consistency; and

Court decisions, which encourage EPA to interpret ambiguous aspects of the RCRA statute and find that lack of consistency sometimes violates the Commerce Clause of the U.S. Constitution and may violate the Supremacy Clause as well.

It also threatens the RCRA reform effort because:

States have a powerful disincentive to adopt RCRA reform measures if a consequence of adopting these measures is that the State could become the preferred location for the treatment and/or disposal of some classes of wastes (the "NIMBY" syndrome).

Some reform efforts, such as efforts aimed at streamlining regulation of universal wastes, require national coordination to truly succeed. It is doubtful that States can achieve national coordination on their own. The history of efforts to site commercial low-level radioactive waste disposal facilities absent enforced consistency and national coordination leaves little room for optimism. Under the Low Level Radioactive Waste Policy Act of 1980, each State is responsible for providing disposal capacity, either by itself or in cooperation with other States (compacts), for all of the low-level waste (LLW) generated within its borders.

Limited progress was made until Congress enacted the Low Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA), Pub. L. 99-240, 42 U.S.C. 2021, which established a series of milestones, penalties, and incentives to ensure that the regional compacts and States made adequate progress toward being able to manage their LLW. In particular, the "take title" provision of the Act would have required a State or compact that could not provide disposal capacity for its LLW after January 1, 1996, to take possession of waste from generators notifying that the waste was available for shipment. In 1992, the Supreme Court struck down the constitutionality of this provision of the LLRWPAA, removing the primary incentive for States and compacts to progress with developing new LLW disposal capacity. (4)

No new LLW disposal facilities have opened since 1980, and existing disposal sites have either closed or imposed restrictions on their receipt of LLW. As a result, many LLW generators are faced with making arrangements for expanded on-site storage of their LLW until additional disposal capacity is available. This situation appears to pose a greater threat to human health and the environment than the disposal of LLW in new facilities licensed under Nuclear Regulatory Commission regulations in 10 CFR 61 or equivalent State (i.e., Agreement State) regulations.

RCRA REGULATORY REFORM AND DOE

DOE has submitted two major regulatory reform proposals to the EPA Hazardous Waste Identification Rule docket. These proposals involve:

an exemption from Subtitle C of RCRA for immobilized mixed waste debris disposed of in low-level waste disposal facilities, and

an exemption from Subtitle C of RCRA for vitrified waste forms.

DOE analyses to support these proposals conclude that they are at least as protective of human health and the environment as current RCRA rules and that they could achieve this level of protection at a much lower cost.

From a radiological perspective, however, granting the DOE-proposed exemptions would appear to be more protective of human health and the environment overall than requiring the continued management of the subject waste forms under RCRA. Given that there are no currently operating DOE disposal facilities for mixed waste residuals, granting the exemptions would permit earlier disposal of these wastes. Disposal is more protective of human health and the environment than above-ground storage because it further isolates the radionuclides in these waste forms from human and environmental receptors.

If the contradiction between consistency and stringency is not addressed, EPA's concurrence with DOE's proposals may be irrelevant. Unless the States hosting major DOE LLW disposal facilities (Washington, Nevada, Idaho, New Mexico, Tennessee, and South Carolina) incorporate these changes in their authorized State programs, DOE will not be able to dispose of immobilized mixed waste debris or some vitrified waste forms in radioactive waste disposal facilities.

The preamble to the proposed HWIR rule suggests that State adoption of HWIR reform proposals affecting mixed waste will be a problem and explains that EPA intends to publish a supplemental proposal on mixed waste exit criteria (see 60 FR 66400; December 21, 1995). Hopefully, the proposal will confront the contradiction between consistency and stringency and recognize that the same arguments that support the need for a consistent munitions rule (i.e., DOD's national defense mission, nationwide presence, and logistical and operational needs) apply at least equally to DOE's mixed waste.

CONCLUSIONS/SOLUTIONS

There is often a contradiction between requiring RCRA-authorized State programs to be consistent with the Federal program and allowing them to be more stringent than the Federal program. The contradiction is most evident when determining what should be considered a hazardous waste, and appropriate waste treatment and disposal requirements. Regulatory reform that affects waste identification, treatment, and disposal intensifies this contradiction.

This paper does not present innovative solutions to this problem or advocate that EPA begin withdrawing authorized State programs for reasons of inconsistency. Instead, it makes the modest suggestion that the problem be recognized and discussed in regulatory preambles and other appropriate forums.

In particular, equivalency and consistency should be addressed even when EPA promulgates a less stringent rule. Authorized State programs that fail to adopt equivalent, less stringent regulations should be examined for consistency with the Federal program. Otherwise, States have a strong disincentive to adopt many less stringent changes.

Also, the meaning of "less stringent" and "more stringent" should be evaluated in the context of RCRA reform. Does it mean more rigorous protection of human health and the environment under RCRA alone, or under all environmental laws and programs? If it means more rigorous protection under all environmental laws and programs, some RCRA reform initiatives that are currently considered less stringent would become more stringent.

Failure to address the contradiction between stringency and consistency imperils regulatory reform efforts. In the case of DOE's mixed waste, failure to address the contradiction may result in lesser human health and environmental protection if radioactive wastes that could be safely disposed of now remain in long-term storage until new mixed waste disposal facilities are built and permitted.

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48-5
ZERO ACCIDENTS IN THE REMEDIATION BUSINESS

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ABSTRACT
While many industrial hygiene concerns are encountered at a hazardous wastes site during characterization and remediation activities, experience indicates that these problems can be controlled adequately. Personal exposure to airborne chemicals are effectively controlled primarily because a plan is prepared to review the chemical hazards that are likely to be encountered. Unfortunately, the hazards that are likely to represent a concern are safety issues including slips, trips, and falls.
Consequently, it is injuries to employees that are the principal concern. This presentation describes the elements of a "Zero Accident" program and the improved statistics resulting from its implementation.
At the beginning of 1994, the Midwest Region of OHM Remediation Services Corp. (OHM) implemented a program to supplement the existing industrial hygiene program to reduce accidents at work sites. The management of the Midwest Region made a commitment to promote and support the philosophy of zero accidents and established that goal as part of the Region's business plan. The Regional Safety Management Review Committee was established and identified the key elements that were necessary to accomplish the goal of zero accidents. These were management commitment, training for employees, job safety analysis, accident investigation, and employee recognition program. While each of these elements existed before January 1994, the "Zero Accident Program" emphasized the importance of and the relationship between these elements in eliminating injuries at a job site. The "Zero Accident Program" provided a baseline for supporting the interaction of these elements. The program also provided a vehicle to communicate the ground rules for effective and safe job performance. The program was communicated throughout the Region in a variety of ways including training classes for site supervisors and foremen to review their roles in achieving zero accidents.
The emphasis on zero accidents has paid off already. The current OHM OSHA recordable injuries rate is down by 52% from 1994 and 63% over the last 3 years. This has produced an estimated cost savings of $213,000 for 1994 and $1,274,000 for the last 3 years. The Experience Modification Rate has dropped by 45% from 1994 and is down 72% over the last 3 years. The number of job sites that achieved zero injuries in 1994 was up by 54% from 1993. With the continued implementation of the "Zero Accident Program," this effort will continue to improve each year and clearly demonstrates that a goal of zero accidents for all remediation job sites is realistic and achievable.

INTRODUCTION

Work at a hazardous waste site poses a variety of risks and hazards to the workers assigned to the remediation tasks. There is a potential to be exposed to airborne hazardous chemicals such as volatile organics that may be flammable or poisonous when inhaled. Chemicals at the job sites may be corrosive or absorbed if they come in contact with the skin. Heavy metals or semivolatile chemicals may be encountered in hazardous concentrations. Employees may be required to work in environments where extreme temperatures are encountered and therefore, may experience heat or cold stress. There are several types of controls to limit personal exposures, particularly engineering controls or administrative procedures. Engineering controls may include remotely operated tools or the use of local exhaust mechanical ventilation. Administrative procedures may limit the duration of the work or specify the time of day that work is performed. If the controls are not adequate, the members of the work crew may be required to wear personal protective equipment to limit exposures via the inhalation pathway or limit skin contact. Personal protective equipment is also effective to protect employees during a spill or unexpected release. The use of personal protective equipment also helps to limit the spread of contamination and serves to increase the effectiveness of the decontamination process.

Chemical hazards have been controlled effectively in the past on OHM job sites. Controls to limit exposures are planned in advance and are accompanied by monitoring with direct reading instruments or sampling for the presence of hazardous chemicals. A health and safety plan is prepared in advance to document the controls to be implemented and the conditions requiring additional controls which may arise. Alarms or action levels are established for the sampling systems to warn the workers that conditions are changing and that precautions are necessary. The plan documents the consequences of overexposure and the actions required to limit the symptoms. The plan also prescribes the methods of decontamination that are required and the steps to be followed to limit the spread of contamination. A review of the employee exposure results at most of the OHM job sites and the results of the medical surveillance for OHM employees indicates that the controls are effective. Personal exposures are found to be significantly below the permissible exposure limits established by the Occupational Safety and Health Administration (OSHA). Current technologies to perform monitoring and analysis at the same time are sufficient to provide instant warnings regarding the presence of hazardous chemicals. Chemical hazards are adequately controlled.

Other hazards are encountered at hazardous waste sites which are severe and can result in injuries to the workers performing the tasks. Working around construction equipment or on elevated surfaces presents potential hazards that offer few warning signs and represents a serious risk of injury. Even simple hazards, such as construction debris or lifting bulky loads, often result in injuries to waste site workers. The hazards are exaggerated by the use of personal protective equipment where the field of vision for the employee is reduced or the protective clothing may snag or be trapped. This hazard may result in a worker tripping and falling to the ground. The hazardous waste site is a dangerous place to work if controls are not implemented to limit the construction hazards.

The OHM Midwest Region implemented a program to manage the construction hazards typically found at job sites that result in accidents and injuries. Regional management personnel pledged a commitment to minimize

accidents at job sites and provide the resources necessary to reduce the hazards. The Regional management personnel agreed that all accidents are preventable and as a result a "Zero Accident Program" was designed and implemented. The program was designed to reduce injuries to employees and ultimately reduce the cost of workmen's compensation and the corresponding medical costs. The "Zero Accident Program" also served to promote the idea of performing each task and the entire project without defects. To the extent that the members of the work crew perform without mistakes the project is completed in a quality manner and more closely to the original schedule that was established.

ELEMENTS OF A "ZERO ACCIDENT PROGRAM"

The elements of the "Zero Accident Program" that were adopted by the Midwest Region of OHM are described below. Many of the steps and applications were first described by the Construction Industry Institute (1) (CII) in a study completed in 1993. The study examined many companies who had implemented a formal safety program in their construction projects and were successful in managing the hazards likely to be encountered. The CII published a list of the elements of safety programs that had been found to be effective. OHM reviewed the list of programs and found many similarities to programs already in place. Each of the companies interviewed explained the connection between unsafe practices, accidents, and injuries to employees. It was explained that the rate of injuries to employees could not be reduced until the number of unsafe practices was reduced. As depicted in Fig. 1, it was estimated that there are more than 300 unsafe work practices and non-injury accidents for each major personal injury (2). The poor work practices that contribute to the major injury are the same as those contributing to the non-injury accidents. The severity of the injury depends on chance. Not all accidents result in a personal injury, but the number of accidents needs to be reduced in order to decrease the number of injuries and minimize the chance that a serious injury may occur.

Fig. 1

Management Commitment

Regional management personnel were first required to confirm their commitment to operate in this manner. While the concept seemed to be an easy decision in theory, the resources needed to implement the program are not without certain conflicts. The Region formed a Safety Management Review Committee chaired by the Regional Vice President to evaluate the effectiveness of the program being implemented in the Region. The Committee is also attended by the Health and Safety Director and representatives from Technical Services, Human Resources, and Operations. The Committee reviews the injuries that occurred during the previous month and discusses the root cause of each incident as identified during the accident investigation. The Committee may elect to invite the project manager and/or the site supervisor to discuss the incident and the corrective actions. The Committee is responsible for establishing changes in procedures and work practices in order to avoid the reoccurrence of the incident at a job site anywhere within the Region. Minutes of the meeting are published and distributed for review by the project managers and operations managers. The Committee is a key element in promoting the prevention of accidents at the job sites.

Many job sites form a similar safety committee hosted by the site supervisor or project manager. These project safety committees serve to resolve concerns identified by the employees assigned to a particular

task on that job site. The committees are chaired by the site supervisors and attended by selected project team members, including laborers. The project safety committees perform site safety inspections and serve as the investigation team in the event of an accident. Minutes of each meeting are issued and distributed to the Regional Committee. Issues that are common to other job sites are discussed at each job site and common solutions are implemented.

Training for Employees

All OHM employees are experienced and participate in a variety of training programs. All personnel that work at job sites as well as many others attend the training classes required by OSHA regarding work at a hazardous waste site. Specific training is provided regarding the chemical hazards that can be expected at job sites. Training is also provided to review the correct operation of equipment, including pumps, chainsaws, high pressure water cleaner, or other small tools that may be used at a typical job site. Training is provided for equipment operators to review the correct use of selected heavy equipment, including excavators, loaders, and cranes. Training is provided in the form of classroom sessions accompanied by practical exercises where the employees can practice the criteria that were described in the classroom sessions. Training is also provided at each job site. At the beginning of each shift, the employees review the safety features of a particular piece of equipment or the results of a recent incident or personal injury. These "tailgate safety meetings" are a good forum for employees to express concerns or ask questions regarding the events of the previous days and confirm that work is proceeding as expected. Regardless of the type of training, the information provided to each employee is critical in avoiding an accident or personal injury.

Job Safety Analysis

Training is required for employees before the project begins, but it is important to focus on likely hazards before a specific task begins. It is appropriate to analyze a task to identify the hazards likely to be encountered and the appropriate controls. The job safety analysis (JSA) is performed by the safety professional and the supervisor or foreman to describe the tasks and the equipment to be used. The analysis can be simple or elaborate according to the complexity of the task. Each member of the work crew reviews the JSA before the task begins so that the hazards are clear and the corrective actions are understood. The process of the JSA also serves as a mechanism to plan the task and verify that the correct tools and personnel are ready for use. This approach helps each team member to understand how the task should be performed and helps provides him or her the information needed to identify a change or problem that was not expected. By addressing the problem as it is identified, the supervisor and the work team minimizes the likelihood that an accident may occur and reduces the potential for a personal injury.

Accident Investigation

Despite all of the analysis and training, conditions exist at the job site which may not be recognized and may result in an accident or personal injury. It is appropriate to investigate the root cause of the incident and identify the corrective actions that would reduce the likelihood of the incident being repeated. The purpose of the investigation is to identify how the incident could have been prevented and recommend the corrective action. The investigation should be

conducted as soon after the incident as possible so that all of the witnesses are available to provide information about the conditions that led up to the incident. Where available, the project safety committee serves as the investigation team and prepares a report for review by the Regional Safety Management Review Committee. A team of project personnel are well suited to identify the root cause and the suggested corrective actions. The members of this team are the most familiar with the operation and the equipment being used and they are the same people who must implement the corrective actions.

The Regional Safety Management Review Committee is responsible for reviewing the corrective actions at the job site. The Regional Committee is responsible for reviewing similar circumstances at other job sites and determine if changes in procedures are required. Members of the Regional Committee convey the results of the investigation team to all other job sites. They challenge the members of other teams to eliminate the root causes of the incident and reduce the hazard that could result in a personal injury. The members of the other project teams are encouraged to identify all potential hazards, not just the one condition that previously occurred. The investigation of an injury is one of the few positive things that stems from a personal injury. Any information which helps to eliminate the personal injury in the future is of value.

Employee Recognition Program

An effective program results in a project team who achieves its goal and can complete a project in an efficient, safe manner. It is appropriate to recognize the achievements made by the members of the project team and reward their hard work. A recognition program must emphasize performance of each individual and also encourage the team members to work in a synergistic manner. A team that works together to reduce job site hazards is more effective than employees who are focused only on their specific assignments.

OHM has implemented a program to recognize outstanding safety performance of the project team as well as individuals. The awards vary from monogrammed baseball caps or jackets to dinner out for the crew members. The important feature is that the work crew selects the award that is a priority for its members and establishes a milestone by which the award may be accomplished which may be related to the progress of the project. For example, the project team members may choose to receive a monogrammed jacket after completing the project (2 months in duration) without any injuries recorded on the OSHA 200 log. The project safety committee petitions the Regional Committee to approve the award and track the performance of the project team. The work crew reports problems to the project management personnel and makes sure that they are resolved as soon as practical. If the milestone is accomplished, each member of the crew is given an award as originally described. Project teams have selected a wide variety of awards from clothing, equipment, travel vouchers, or bonus checks.

Awards are also granted for individuals within the Region that complete a year without an OSHA recordable injury. The awards are selected by the Regional Safety Management Review Committee and distributed to eligible individuals. Clothing and equipment are commonly awarded with more expensive awards assigned to individuals who have completed multiple years without a recordable injury. There are many methods to reinforce the success of safe work practices and each method serves to emphasize the importance of the "Zero Accident Program."

RESULTS

The "Zero Accident Program" is showing dividends for the corporation. The rate of OSHA recordable injuries has dropped by more than 63% over the last 3 years (see Fig. 2). Two of the OHM profit centers, Midwest Region and Southeast Region, recorded their last lost work time injury in 1994 and have accumulated over 2.6 million labor hours without a lost time injury. The number of job sites that were able to achieve zero injuries is increased over the last 2 years by 54%.

OHM's Experience Modification Rate, published by the insurance companies to reflect the cost of injuries compared to other companies in the same industry, continues to decline. It is currently 87% below the expected average and it has dropped by 50% each year for the last 2 years. As importantly, many of our clients are recognizing this effort by offering monetary incentives to complete the project without any incidents.

Fig. 2

CONCLUSIONS

1. The chemical hazards encountered at a hazardous waste site can be managed so that exposures to employees are acceptable. The hazards stemming from construction activities represent the major source of personal injury and must be controlled.
2. There is a connection between accidents and personal injuries. In order to reduce personal injuries, a company must reduce accidents at the workplace. It is practical to expect that injuries can be reduced to zero by minimizing accidents and the conditions that contribute to accidents.
3. A "Zero Accident Program" requires the commitment of management to promote the elements and provide adequate resources to correct the discrepancies.
4. The "Zero Accident Program" has been successful for OHM since the spring of 1994. The rate of OSHA recordable injuries has dropped by 63%. The Midwest Region has accumulated over 1 million hours without a lost time injury since the program started.

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ISV TECHNOLOGY TRANSFER FROM BATTELLE/PNNL TO GEOSAFE CORPORATION TO ISV JAPAN

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ABSTRACT

The In Situ Vitrification (ISV) technology was invented by Battelle Pacific Northwest National Laboratory (PNNL) for the U.S. Department of Energy (DOE) in 1980. The technology involves the electric melting of contaminated soil and debris, in situ, for purposes of destroying organic contaminants and permanently immobilizing hazardous and radioactive heavy metals. The ISV technology has been developed and demonstrated on a broad range of contaminants, soil types, and waste materials and debris. Geosafe Corporation (Geosafe) offers ISV remediation and waste treatment services on a large-scale commercial basis, and has recently completed its third major site remediation project.

The ISV technology was transferred from Battelle/PNNL to Geosafe starting in 1988. Subsequently, Geosafe has transferred the technology to a new company, ISV Japan, Limited (ISVJ) starting in 1995. This paper reviews the technology development and commercialization chronologies for the ISV technology, and then discusses the technology transfer processes that were performed during that time. Eight (8) basic elements of the technology transfer process are presented, including: 1) license rights, 2) technical information, 3) laboratory test/demonstration know-how, 4) commercial production technology know-how, 5) provision of experienced staff members, 6) continuing technical assistance, 7) financial/investment support, and 8) shareholder positioning. The two technology transfer events (Battelle/PNNL to Geosafe, and Geosafe to ISVJ) are contrasted relative to these eight elements.

The authors believe that the technology transfer model represented by the 8-elements listed above, is very effective. It is noted that one weakness of the process lies in the area of estimating the time and costs associated with translating a laboratory demonstrated technology into a commercial production technology. It is recognized that technical operating know-how is critical to the success of effective technology transfer; and provisions must be made to ensure that knowledgeable staff are made available to the recipient of a technology transfer, until such time that they possess the needed know-how themselves. In the Battelle/PNNL to Geosafe technology transfer, this need was met by Battelle's provision of full-time experienced technical and business development staff to Geosafe. In the Geosafe to ISVJ technology transfer, this need is being satisfied by provision of knowledgeable Geosafe staff on an as needed basis under a technical assistance contract. Lastly, the paper identifies a continuing financial involvement between the parties as an important element of effective technology transfer.

INTRODUCTION

The purpose of this paper is to review the processes employed in the transfer of the ISV technology originally from the DOE's PNNL to Geosafe, and subsequently from Geosafe to ISVJ. A general technology transfer model is presented, and the two technology transfer experiences are contrasted relative to it. The following summary description of the ISV technology is presented to help the reader relate to the type of technology being transferred, and the type of business Geosafe is involved in.

The ISV technology was an outgrowth of the joule-heated melter vitrification technology developed by PNNL in the 1970s and 80s for the immobilization of high level radioactive waste. At that time, vitrification had been recognized worldwide as the best available means to immobilize heavy metal radionuclides; and PNNL was DOE's reference vitrification laboratory. In 1980, PNNL melter vitrification experts were given DOE discretionary funds to do proof-of-principle testing on the ISV concept. The initial test was so successful that DOE's Hanford Field Office immediately challenged PNNL to develop a means of vitrification that could be applied in situ to soil sites that had been previously contaminated with TRU waste. Since the 1980 invention (and 1983 patent), the ISV technology has been developed to be broadly applicable to site remediation and waste treatment applications involving a full range of hazardous, radioactive, and mixed contaminants present in soil, wastes, other earthen materials, and debris.

The ISV technology involves the in situ electric melting of earthen materials for several possible purposes, including: 1) thermal destruction/removal of hazardous organic and inorganic compounds, 2) permanent immobilization of hazardous and radioactive heavy metals within a high integrity vitrified product form, 3) conversion of waste materials into recycled products, and 4) production of high integrity vitrified masses for various civil engineering applications.

The ISV process involves placement of an array of electrodes (usually 4) a short distance into the media to be treated. A melt is initiated at the surface between the electrodes, and as electrical power is passed through the melt, joule heating occurs, causing the melt to grow outward and downward as long as application of power is continued. Geosafe employs a mobile, large-scale system capable of applying 3.5 MW of power to a melt. This system has been employed to make individual melts as large as 1,400 tons in mass, and approximately 40-ft in diameter by 22-ft deep.

The high operating temperature of the ISV process (1,600-2,000C) results in the pyrolytic destruction of organic contaminants. Heavy metals are typically immobilized by chemical incorporation into the high integrity vitrified product that results from the melting process. Volume reduction on the order of 20-50% is typical for most earthen media applications. The vitrified product typically possesses outstanding physical, weathering, leaching, and biotoxicity properties. These properties for the ISV product are typically superior to those of ex situ melter products because of the effects of fluxant additives which are necessary to lower the melt temperature in ex situ melters.

The ISV process and its attendant vitrified product may be the most technically preferred and cost effective technology alternative in cases where: 1) a combination of organic and heavy metal, or hazardous and radioactive contaminants must be treated, 2) onsite and in situ treatment is desired, 3) the site contains non-homogeneous soil conditions and/or debris, 4) maximum treatment effectiveness and permanence are required, and/or 5) high volume reduction is desired. The reader interested in more information regarding the ISV technology is referred the authors and specifically to the following two references (1)(Thompson, 1995).

Geosafe provides four basic ISV-related products/services to the site remediation and waste treatment markets, including: 1) applicability analyses and cost estimates, 2) treatability testing and demonstrations, 3) technical assistance during remedial design, site preparation, and site restoration, and 4) onsite vitrification services. Figure 1 shows Geosafe's large-scale ISV equipment being operated at the Wasatch Chemical Superfund Site in Salt Lake City.

REVIEW OF ISV TECHNOLOGY DEVELOPMENT AND COMMERCIALIZATION

The ISV technology was invented and proof-of-principle tested in 1980 by PNNL at the Hanford Site. Initial process exploration was performed until 1983, when the program emphasis moved toward a large-scale equipment design and demonstration basis. The initial application interest was DOE's TRU-contaminated soil sites; a successful pilot-scale demonstration on Pu-contaminated soil was performed in 1983.

By 1986, a large-scale (100 ton/day) mobile ISV system had been designed, constructed, and operational acceptance tested. This enabled performance of the first radioactive large-scale test at Hanford's 200 Area during 1986. Approximately 120 tests and demonstrations were performed at engineering-, pilot-, and large-scale between the years 1980 and 1990. Many of these tests demonstrated the applicability of the technology to a

full range of hazardous, radioactive, and mixed contaminants. This range of capabilities strongly interested the non-DOE marketplace. Geosafe was established in 1988, primarily to address the non-DOE markets that were developing in response to CERCLA, RCRA, and TSCA regulations.

Fig. 1

It should be noted that the nature of the DOE ISV technology development program emphasized rapid equipment development and demonstration as a means of quickly evaluating the potential of the technology. While this approach certainly demonstrated the potential of the technology, it did not allow time nor funds for adequate definition of the underlying science and technology needed to support commercial application of the technology. This became obvious after Geosafe attempted to initiate commercial operations at large-scale; and it became necessary for Geosafe to spend substantial time and money on "backfilling" the base science and technology needed to translate the "laboratory-based" experience to the commercial arena. This translation was completed in 1992, and Geosafe initiated its first large-scale commercial work in 1993. Since that time, Geosafe has performed more than 50 large-scale ISV melts, and has demonstrated the broad capabilities and reliability of the technology.

Fig. 2

Figure 2 summarizes the technology development and commercialization chronologies since 1980. The reader interested in more detail in this area is referred to the ISV commercialization case study presented by the Geosafe authors at Waste Management '95 (3).

ISV TECHNOLOGY TRANSFER

Technology transfer involves passing necessary rights, information, and knowledge, regarding a technology, from one party to another so that the receiving party may effectively employ the technology. There are many ways that technology transfer can be accomplished. At one extreme, transfer may be accomplished simply by licensing a patent to a knowledgeable recipient. At the other extreme, effective technology transfer may require a more extensive effort and a long-term relationship between the transferring and receiving parties. As prior employees of Battelle Memorial Institute, the Geosafe authors were involved with technology transfer activities dating back to the 1970s, primarily from the point of view of the transferring party. In that role they have experienced the difficulties involved in locating and generating interest in a potential technology recipient. As Geosafe employees, they have seen the technology transfer process from the technology recipient's point of view. These experiences have led to a technology transfer model that includes eight (8) basic elements, as follows:

1) Provision of license rights to patent(s), technical information, know-how, and future improvements
2) Provision of full technical information regarding the technology, including process and equipment specifications and a historical project database
3) Provision of know-how related to technology test and demonstration capabilities
4) Provision of know-how related to use of the technology in commercial production
5) Provision of experienced technical and business development staff members
6) Provision of continuing technical, business development, and environmental industry assistance

7) Provision of financial and/or investment support
8) Assumption of an ownership interest by the transferring party in the receiving party.

Transfer of the ISV technology has occurred two times, and is in the process of being performed a third time. The original transfer was from DOE's PNNL (operated by Battelle) to Geosafe, starting in 1988. The second major transfer was from Geosafe to ISVJ starting in 1995. Preparations related to transferring the technology to Australia are currently underway. Figure 3 relates the first two technology transfer processes to a time line and to the commercial status of the technology. Each of these technology transfers are discussed individually below, relative to the technology transfer model listed above.

Transfer from DOE/PNNL to Geosafe Corporation Battelle invented the ISV technology for DOE in 1980. A very fundamental U.S. patent was issued in 1983. DOE also filed for and obtained ISV patents in the major industrialized countries of the world. After issuance of the U.S. patent, DOE announced that nonexclusive rights to use the patent were available to interested parties; to Geosafe's knowledge, there were no respondents to that offer. With hindsight, the authors believe that the early development stage of the technology, and the nonexclusivity of available license rights, contributed to the lack of applicants in response to DOE's offer. Subsequently, Battelle applied for and was granted partially exclusive rights to the technology in 1986. The rights were partially exclusive in that they did not include rights to the radioactive field of use. Radioactive rights were granted within a few years, making Battelle's license exclusive in all fields of use, worldwide. In exchange for these rights, Battelle had to commit to preparation and implementation of a strategic business plan leading to commercialization of the ISV technology. This was completed in 1987. Battelle's plan called for the establishment of a separate company for commercial application of the technology; Geosafe was so established in 1988.

As the world's largest contract R&D firm, Battelle held extensive experience at that time in the field of technology transfer. Many technologies were transferred at that time by licensing patent rights, provision of technical information, and provision of technical assistance to licensees as required. Battelle's strategic analysis of the opportunities for the ISV technology, and the nature of the developing site remediation marketplace, caused them to make an extraordinary commitment to the commercialization of ISV, including provision of key technical and business development staff, and initial capitalization of the new company (Geosafe), in addition to exclusive worldwide sublicense rights, technical information, and continuing technical assistance.

Fig. 3

This transfer of technology from Battelle/PNNL to Geosafe resulted in a licensee that had a full and working knowledge of the technology from technical and business development points of view, and had a direct path for advanced technical assistance as needed. Recognizing that the key staff from Battelle did not have significant environmental remediation industry experience, Battelle further increased the chances of success for its offspring by employing a management team that had prior successful experience in the industry.

It is important to recognize that this transfer of technology involved the transfer of a "laboratory developed and demonstrated technology", but not a "commercially viable production technology". It was believed at

that time that it would be but a short step to translate the laboratory technology into a fully functioning production technology. That belief proved to be in error, and Geosafe found it necessary to invest significant time, money, and effort into producing a commercially viable technology. Had Geosafe been unable to obtain additional capitalization during this period, this technology translation need could have been fatal to the technology transfer and commercialization effort. It is recognized that it is very difficult to accurately predict the effort needed to translate a laboratory technology into a production technology. The importance and need for such translation should be a major consideration in technology transfer efforts.

The transfer of technology from Battelle/PNNL was substantially completed by 1992, when Geosafe's capabilities became strong enough in the ISV technology field to minimize the need for further technical assistance from Battelle/PNNL, except for longer-term R&D needs. As would be naturally expected, the child rapidly gained knowledge and experience from its technology translation efforts and large-scale field activities, making it stronger than the parent in the transferred technology. In fact, collaboration between the two organizations today often involves "reverse" technology transfer, as Geosafe provides technical assistance to Battelle/PNNL in support of its ISV Program being performed for various DOE interests.

Transfer from Geosafe to ISV Japan

Geosafe's sublicense involves worldwide rights. Since its inception, Geosafe has received more than 200 inquiries from foreign organizations regarding the possibility of sublicensing the ISV technology. Such inquiries were handled in a manner consistent with the state of the technology, and Geosafe's perceived need to firmly establish ISV within the U.S. marketplace before taking it abroad. Most inquiries were "blind" (i.e., based on little knowledge of the ISV technology and its status), and were motivated by foreign firms searching the U.S. literature for technology acquisition opportunities.

Many such inquiries also came from Japan, which during the past 5 to 7 years, has been developing its environmental remediation and waste treatment markets. One very different line of inquiry came to Geosafe through the Japan Research Institute, Limited (JRI). JRI had assembled a consortium of companies that were interested in exploring ISV's potential for the Japanese marketplace. At that time they neither requested, nor were ready to seek a sublicense to ISV. Rather, they proposed a time of working together to evaluate the technology and its potential application within Japan. That approach held significant interest to Geosafe; and in 1990, cooperative efforts were undertaken. Membership in the Japan ISV Consortium varied with time; typically there were about 8 member companies, including JRI, a major trading company, a major chemical industry firm, and several construction companies (Japan's remedial action contractors).

For the following four years, Geosafe performed exploratory tests related to the Japanese "condition", at its test site in Richland, WA. Testing was performed in several key market application areas (e.g., LLW facility remediation, contaminated soil remediation, industrial waste treatment, concrete D&D waste treatment). In addition to the U.S.-based testing, the annual program involved performance of demonstration tests in Japan to confirm and demonstrate to others that the U.S. test results would be the same for Japanese soil conditions. The Japan ISV Consortium would arrange

for "field days" (typically two consecutive days) in conjunction with the demonstration projects for purposes of informing potential clients and regulators about the ISV technology. Often test melt products (vitrified monoliths) from U.S. testing would be shipped to Japan for display during such demonstration field days. JRI and the Consortium were very effective in developing market and regulatory awareness within Japan in this manner. Television and printed media coverage was also present during the field days.

In addition to the technology side of the exploratory and evaluation efforts performed by the Consortium, they also worked extensively in the regulatory and marketplace areas. They were able to obtain partial financial support from the Japanese Ministry of International Trade and Industry (MITI) to support their efforts. In 1994, MITI contributed about $5 million toward the design, construction, and testing of a stationary batch type ISV treatment system as part of an economic stimulation project in Japan. That system is shown in Fig. 4.

After the several year evaluation process, a majority of the Consortium members decided that an appropriate opportunity existed for ISV in the Japanese marketplace. The Consortium then proceeded to request and negotiate a license from Geosafe for the territory of Japan. That license was finalized, and ISVJ was formally established in Tokyo during May, 1995.

Fig. 4

COMPARISON OF TECHNOLOGY TRANSFER EFFORTS

Significant differences exist between the transfer of ISV technology between Battelle/PNNL and Geosafe, and between Geosafe and ISVJ. Table I presents the elements employed in both transfers, and points out the primary differences. Most notably, the Geosafe-ISVJ transfer involved transfer of a fully developed, commercially proven technology; whereas the Battelle/PNNL-Geosafe transfer involved transfer of a laboratory demonstrated technology. This situation, at the time of the Battelle/PNNL to Geosafe technology transfer, meant that Geosafe had to bear the risk of the time and expense of translating the laboratory technology into a commercial production technology. This was not the case of the transfer between Geosafe and ISVJ. In this case, ISVJ received a fully developed commercial technology and did not have to withstand the risks that Geosafe had to when it received the technology from Battelle/PNNL. This difference provided ISVJ with much greater certainty that the ISV technology could be immediately applied on a commercial basis within Japan.

Another significant difference between the two technology transfer cases lies in the area of technology deployment know-how that exists within key Geosafe staff members. As mentioned above, the Battelle/PNNL to Geosafe transfer involved the employment of experienced Battelle/PNNL staff members as key staff of Geosafe. This type of know-how transfer has not been done in the Geosafe to ISVJ transfer, primarily because two different countries and cultures are involved, and because of the distances involved. It is recognized, however, that the transfer of know-how to the new company is a crucial ingredient of the technology transfer process. Therefore, as part of the license agreement between Geosafe and ISVJ, Geosafe has committed to provide onsite technical support as needed to support the project development, planning/design, staff training, and performance of initial projects, until such time that ISVJ no longer requires such assistance. The agreement requires that Geosafe technical

staff approve all operational plans and equipment designs prior to commencement. The agreement also requires the participation of Geosafe personnel, onsite, at an agreed upon level during initial operations. In this way, Geosafe will ensure that the needed know-how is present during ISV operations.

Table I

The provisions described above also address another very important consideration of technology transfer; that is the need to ensure that the technology is not misapplied. The ISV technology is not a straight forward extension of an existing technology (e.g., joule-heated melters); rather it is fundamentally different in most aspects of its operation. There is a significant amount of underlying, ISV-specific science and engineering knowledge that is necessary to properly apply the technology. And it is possible to misapply it. Therefore, clear establishment of an acceptable operating "envelope" is a crucial element of the transfer process. And of course, it is of utmost importance to both parties that the technology not be misapplied due to the repercussions that could result in both the U.S. and Japanese marketplaces. The license agreement between Geosafe and ISVJ specifically addresses these concerns.

A last significant difference between the two transfer processes also partially relates to the above needs. In the case of the Battelle/PNNL to Geosafe transfer, Battelle became the primary shareholder of the new company. This guaranteed the kind of close, continuing relationship that is very helpful for effective technology transfer. It is noted, however, that Battelle is not in the business of operating subsidiary companies, but rather has invested in Geosafe as part of the technology transfer process. It is expected that Battelle's share in Geosafe will decrease as additional capitalization is obtained from environmental industry sources to support Geosafe's growth. Geosafe has also become a significant shareholder of ISVJ, but not as a primary investor. This investment was not necessary to provide initial capitalization to ISVJ, but rather it provides benefits to both Geosafe and ISVJ. It allows Geosafe to participate in the growth and success of ISVJ; and it guarantees Geosafe's attention to the needs of ISVJ.

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CONSIDERATIONS FOR THE IMPLEMENTATION OF PARTNERING AGREEMENTS FOR TECHNOLOGY TRANSFER RELATED TO DEPARTMENT OF ENERGY PROGRAMS

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ABSTRACT

The Department of Energy (DOE), and the DOE national laboratories in particular, have developed numerous processes and technologies that have the potential to be of practical use to both the government and

commercial market sectors in areas such as environmental remediation, waste management, applied physics, remote handling, etc. For effective commercial application, these technologies must be transferred to private enterprize for further development, tailoring for specific functions and uses, and full development and targeting of potential markets. This transfer will require the skills, interfaces, and business relationships of a consortium of organizations with appropriate specialized experience and expertise.

In order to develop and implement effective technology transfer programs, it is beneficial for the DOE site or laboratory to enter into partnering arrangement with engineering/technology firms, universities and academia, private industry, other government agencies, and specialized business entities to support the identification, development and transfer of emerging technologies developed and/or adapted by a laboratory or research and development (R&D) organization which can be applied to address ongoing or future needs for environmental management.

The concept of partnering represents a long-term commitment among the involved organizations for the purpose of achieving specific business objectives by maximizing the effectiveness of the resources of each participant. This relationship requires change in the traditional business and working relationships to a shared culture without regard to organizational boundaries. This relationship must be based upon trust, dedication to common goals, and understanding of the individual expectations and goals of the partnering team members.

A typical partnering arrangement for DOE technology transfer should accommodate the following goals:

Development and transfer of technology from the DOE operations, laboratory, and/or R&D environment to business entities within the private sector (commercialization);

Implementation of an arrangement to involve (to the extent practicable) entrepreneurial organizations such as small businesses, small disadvantaged businesses, woman-owned businesses, and collegiate universities and organizations;

Identification of areas of potential ongoing and/or follow-on research in support of the DOE and private/commercial sector requirements.

These objectives will be achieved by establishing defined roles and responsibilities of the member organizations of the partnering arrangement, and by emphasizing the strengths and relevant experience of the team as a unit. The partnering relationship will serve as a business venture, and will conduct research, development, and transfer processes in terms of a defined project management and earned value accountability system.

The key elements of the partnering agreement are: long term commitment; equality among partnering members; a mutually supportive relationship; encouragement of innovation; a commitment to continual improvement; insurance of continuity of resources; and the promotion of a `win-win' atmosphere.

The ultimate measure of success will be the demonstrated improvement of the ability of the consortium to identify currently available technologies useful to the private sector, and the expediting of technology transfer available for commercialization.

The technology transfer process must address the interrelationships and organizational dynamics necessary to bring a developmental technology through the development process into a marketable condition. Since the

transfer of DOE technology to the private sector currently is a developmental process, there is no patent guidance for this process. The development of a partnering arrangement between DOE and a consortium of organizations with complementary skills can lead to an orderly and efficient mechanism for introducing DOE technologies to the common industrial market.

OVERVIEW OF REQUIREMENTS

The goals of for a technology transfer program are to apply business principles to technical development in order to increase the return on investment to a higher ratio, avoid unnecessary costs and expenses, expedite the process of technology transfer directly to the common market, and to implement of effective cost and financial controls to demonstrate earned value and value added.

The technology transfer process must implement technology development and demonstration of `lessons learned' to improve the understanding of available and emerging technologies. In order to identify, develop, and transfer technologies to the private sector for potential commercialization, specific process elements and related responsibilities must be addressed by the team, such as basic research, applied research, pilot/bench scale development, technology demonstration, technology transfer, and commercialization.

The process of technology development, implementation and transfer must involve the identification of problem or need and a related candidate remedial technology (or technologies). The technology may be a process, procedure, equipment and/or system that meets a specific need and allows the user to accomplish the desired results in a more efficient, cost effective, and/or safe and environmentally preferential manner than existing methods and/or techniques.

The development of the candidate technology may require basic and/or applied research, testing and implementation of a process or equipment, or the adaptation of existing processes, systems, or methods. To assure that a technology is cost effective and addresses the appropriate application, several stages of analysis and implementation will be conducted. Each stage requires unique talents, capabilities, and experience from the partnering team members. The team must reflect optimum utilization of the appropriate allocation of resources and expertise without regard to the personal preferences of the individual team members.

PARTNERING OBJECTIVES

Partnering is a long term commitment between organizations for the purpose of achieving specific business objectives by maximizing the effectiveness of the resources of each participant. This relationship requires change in the traditional relationships to a shared culture without regard to organizational boundaries. This relationship must be based upon trust, dedication to common goals, and understanding of the individual expectations and goals of the partnering team members.

The partnering arrangement among team organizations should be structured to accommodate the following goals:

Development and transfer of technology from the research/national laboratory environment to business entities with the private sector (commercialization);

Identification of areas of potential ongoing and/or follow-on basic research in support of the Department of Energy (DOE) and private/commercial sector requirements; and

Implementation of an arrangement to involve (to the extent practicable) small, disadvantaged, and/or minority organizations, including Small Businesses (SBs), Small Disadvantaged Businesses (SBDs), and Historically Black Colleges, Universities and Minority Institutions (HBCU/MIs). These objectives can be achieved by establishing defined roles and responsibilities of the member organizations in the partnering arrangement, and by emphasizing the strengths and relevant experience of the team as a unit. The partnering relationship should serve as a business venture, and should be used to conduct research, development, and transfer processes in terms of a defined project management and earned value accountability system (e.g., as per DOE Order 4700.1).

A partnering agreement should not be construed as a contractual agreement. Partnering does not create legally enforceable rights or duties. Partnering does not replace legal contracts; it is a mechanism to implement contracts. Partnering should not be perceived as a `quick fix'; it cannot be implemented without the total commitment of all of the involved parties.

The key elements of the partnering agreement are: long term commitment; equality among partnering members; a mutually supportive relationship; encouragement of innovation; a commitment to continual improvement; insurance of continuity of resources; and the promotion of a `win-win' atmosphere. Success will be based on:

- Business-driven objectives;
- Dedication to total organization as part of the strategic plan;
- Top-down organizational commitment;
- Willingness to accept other/alternate ways and concepts of conduct;
- Agreement to equality among team participants;
- Commitment to invest resources for future returns;
- Ability to relinquish control and back away from specific details;
- Acknowledgement of risks and planning for risk management; and
- Mutual trust in the skills, intentions, and contribution of team members.

The ultimate measure of success will be the demonstrated improvement of the ability of the team to identify currently available technologies useful to the private sector, and the expediting of technology transfer available for commercialization.

IMPLEMENTING A PARTNERING AGREEMENT

Technology transfer and cooperative R&D programs typically have been met with skepticism, disagreement on partnering terms, and with controversies over topics such as ownership of intellectual property. These barriers to must be overcome if a partnering arrangement is to have any opportunity for success. Since the track record so far for commercialization of technologies has been spotty and erratic, precautions must be taken to ensure accomplishment of specified goals.

The partnering agreement is not without some concerns and issues that must be resolved. These issues and concerns include:

- Protection of proprietary information;
- Licensing of products and technology;
- Fair sharing of risks;
- Obtaining and maintaining total commitment to the relationship;
- Creation of dependencies on partnering members;
- Inherent limitations of the competitive market strategy; and
- Difficulties with the integration of organizations with differing cultures.

The primary focus of the arrangement will be for the partnering organizations to assist each other in the development of technology that can be optimally packaged for implementation and transfer to the commercial sector. The team members must have the commitment to identify areas of further development and needs for new basic research as a source of business development for the technology. This cyclic relationship will provide promotion of a continuing and ongoing partnership that will provide not only technology transfer, but will establish the basis for the team to become a long-term contributor to private industry.

PERFORMANCE REQUIREMENTS AND EXPECTATIONS

The development and transfer of technology must focus on the major obstacles to progress that have existed with previous efforts involving national laboratories and R&D organizations. The team must emphasize problem/solution-oriented technology development to support specific client and market sector needs. The partnering program must implement a management structure that involves the team members, customers, regulators, and other stakeholders. This up-front involvement will be instrumental in the establishment of effective communications with internal and external stakeholders, and the conduct of technology development and implementation in an open and interactive fashion. The early buy-in of stakeholders (and potential intervening organizations) will streamline the transfer process, thereby reducing costs and schedules.

The goals of this program must be to apply business principles to technical development in the following areas:

Increasing the return on investment to a higher ratio;

Avoidance of unnecessary costs and expenses;

Expediting the process of technology transfer directly to the common market; and

Implementation of effective cost and financial controls to demonstrate earned value and value added.

Also, partnering is a vehicle for concepts such as Total Quality Management and innovative thinking for the development of technologies. Expected benefits include improved efficiency and funding utilization, and continuous improvement of quality products and services.

The primary team members of a partnering agreement are the supplying organization (e.g., the national laboratory) and the design/engineering firm. Academia will be included to provide additional resources and support for technology development. Small businesses will be represented on a case-by-case basis by companies identified specifically for the selected technology development and transfer application.

THE TECHNOLOGY DEVELOPMENT/TRANSFER PROCESS

The process of technology development, implementation and transfer must involve the identification of problem or need and a related candidate remedial technology (or technologies). The technology may be a process, procedure, equipment and/or system that meets a specific need and allows the user to accomplish the desired results in a more efficient, cost effective, and/or safe and environmentally preferential manner than existing methods and/or techniques.

The development of the candidate technology may require basic and/or applied research, testing and implementation of a process or equipment, or the adaptation of existing processes, systems, or methods. To assure that a technology is cost effective and addresses the appropriate application, several stages of analysis and implementation will be

conducted. Each stage requires unique talents, capabilities, and experience from the partnering team members. The team must reflect optimum utilization of the appropriate allocation of resources and expertise without regard to the personal preferences of the individual team members.

The technology transfer process must implement technology development and demonstration of `lessons learned' to improve the understanding of available and emerging technologies. In order to identify, develop, and transfer technologies to the private sector for potential commercialization, the follow process elements and related responsibilities will be addressed by the team:

Basic Research

This preliminary effort involves the investigation of identified technologies which may not be directed at a specific problem or need, but have potential for application in one or more target areas. The focus in this area will be on the accumulation of knowledge, with an understanding that some concepts of applicability are necessary. The instigation of a relationship with a private engineering organization, along with the connections to private industry, will allow technology researchers to evaluate the implications of results and the potential technology applications. This area is primarily the focus of the national labs; academic institutions may be utilized to supplement laboratory resources.

Applied Research

This effort involves the definition of specific problems or problem types, and the proposal of candidate technologies to solve these problems. Evaluation of the needs of private industry and DOE programs will provide access to field problems to which research can be applied. The engineering firm may support the national laboratory as required in the coordination of investigations into industry and DOE requirements. Academic institutions may play a role in the development of research applications.

Pilot/Bench Scale Development

This is the proof-of-principle phase. The feasibility of a process for application to a specific problem is the basis for this process element. Identification of potential implementation problems and resolution of technical issues typically are accomplished at this stage. The national laboratory and the team engineering firm will participate here in a coordinated effort with academia and a business interested in commercialization.

Pilot/bench scale testing will involve the identification of performance metrics, evaluation of the technology status, benchmarking and baselining of technology, analytical testing, and comparative analysis to define the utility of the candidate process and/or equipment.

Technology Demonstration

At this stage, the candidate technology will be applied to a `real world' problem in the field. Procedures will be developed to document the process, and tested with the expectation that the technology can be fully implemented with the incorporation of the results of the demonstration. This step will involve the preparation of drawings, procedures, and other documents required for consistent and reproducible operation, and may include ancillary issues such as safety, training, quality assurance, and permitting. The engineering firm will have a major role in this area for coordination of activities with the technology developers and the implementing organization (e.g., small businesses).

Technology Transfer

This is the establishment of the ways and means to prepare the technology for introduction into the commercialization process. This will involve documentation of demonstration results and specifications, and extrapolation of potential and possible applications of the technology within the commercial/private industry environment. The laboratory and the engineering firm will coordinate this transitionary phase with the technology developers and implementation organizations, and may include potential end-users.

Commercialization

This step requires that the process or equipment be defined, developed and packaged for specific commercial applications, and that the product meets safety and regulatory standards for commercial. Necessary permits and licenses must be in place to assure compliance. Primary and secondary markets should be identified and developed for potential commercial application of the processes.

It will require a dedicated, well-planned, and cooperative effort to move technology into the private sector. The team engineering firm will be instrumental in working with the partnering members to target commercial industries and to present technologies in the most marketable perspective.

The engineering firm must take a lead role in this process phase, with the support from academia and technology vendors (e.g., small businesses or other organizations that will make the technology available to the public) for commercialization implementation. The partnering team also will be responsible for identifying areas of possible related research, development, and analysis as a feedback loop for future efforts.

ROLES OF THE PARTICIPANTS AND PARTNERING AGREEMENT

The integrated partnership must include defined roles and levels of responsibility for the initiation, evolution, and completion of each technology transfer project. The following sections define a proposed sharing of responsibilities:

National Laboratory/R&D Organization

The national laboratory will contribute a large array of facilities and resources not available in the private sector, a mutidisciplinary staff, and a history of the development of science and engineering technologies that may be developed to serve national needs in areas such as environmental restoration and waste management. The team of dedicated scientists, technologists, and engineers will provide a cooperative source of superior resources for technology development.

The national laboratory would have the primary responsibilities for:

- Conduct of basic research
- Conduct of applied research
- Provision of scientific/technical support
- Participation in pilot/bench-scale testing
- Provision of testing facilities
- Coordination with user facilities

Architect/Engineering Firm

The team A/E firm should bring an established credibility in the DOE complex for design, evaluation, and demonstration of engineered systems and technology. Successful A/E companies add an extensive network of DOE, DoD and private industry contacts, and can promote the conversion of concepts into directly applicable technology for the commercial marketplace. These firms have experience which can provide opportunities

for research and development in addition to access to ongoing remediation projects where new technologies may be implemented, and can identify areas with potential for future research.

Due to the nature of the architect/engineering business, the A/E team member will have historical and current experience in the establishment and deployment of full-service teams to comply with client requirements. The engineering firm can take the lead role in the integration of the team, interfacing of team elements, and the commitment of mutually-beneficial participation for all partnering members.

The primary contributions of an A/E firm to the partnering arrangement will be:

- Conduct of business planning
- Development of financial feasibility analyses
- Integration of team members and elements
- Conduct of market surveys
- Provision of project management
- Conduct of engineering/design support
- Implementation of technical optimization
- Integration of systems configuration
- Conduct of regulatory analysis
- Preparation of permitting and licensing requirements
- Preparation of technical procedures and specifications
- Identification of new needs and opportunities for laboratory programs

Universities and Academic Organizations

Universities and academic organizations should be selected to the partnering team that have a nationally respected reputation and that provide progressive programs in the promotion of engineering and technology development. Universities typically have dedicated faculty and well-developed facilities to support a partnering arrangement. The university also provides access to an engineering student body that can be dedicated to performance, and that is eager to participate in the development and implementation of leading-edge technologies.

The main focus of participation by academia will include:

- Participation in pilot/bench-scale testing
- Preparation of training programs and teaching aids
- Participation in faculty exchange programs
- Provision of co-op/graduate students for research
- Coordination of technology access from other labs
- Investigation of technology transfer from/to other organizations
- Conduct of literature surveys and information searches
- Preparation of technical reports

Small Businesses

Targeted, focused small businesses are the most effective means of making quick decisions on technology transfer, and should always be considered as potential partners. These small businesses provide an entrepreneurial and action-oriented culture that can serve to identify and act on opportunities and to reduce cost and save time through the implementation of innovative ideas. These companies may be equipment vendors, technology implementation specialists, and/or consultants within the DOE culture, as appropriate for the specific technology transfer venture.

The small businesses will be responsible to:

- Support implementation of technology
- Provide needed technology innovations
- Participate in technology demonstrations

Coordinate small business innovative research funding

SUMMARY OF THE PARTNERING PROCESS

The proposed concepts of a partnering agreement combines national laboratories, engineering firms, and academic institutions as the committed long-term team members, with supplemental small and/or specialty businesses selected as appropriate for each technology demonstration and transfer project. The national laboratory efforts will be concentrated in the front end of the process, where technology identification and development are key elements. The academic institution will provide supplementary development and innovation support, and will assist in receiving technology from the laboratory and preparing the technology for commercialization. The team engineering firm will provide the necessary coordination, management, and engineering/design to promote the success of an effective and beneficial partnering relationship.

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ROADMAP PROCESS FOR COMMERCIALIZATION AND USE OF EMERGING CHARACTERIZATION AND MONITORING TECHNOLOGIES

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ABSTRACT

A roadmap process designed to facilitate acceptance of emerging technologies by users and commercial vendors has been developed for use by the U.S. Department of Energy (DOE) Characterization, Monitoring, and Sensor Technology Crosscutting Program (CMST-CP) to transfer its portfolio of technologies. The CMST-CP technology transfer roadmap consists of six main sections: needs assessment, identification of technologies for transfer, a managing-technology-for-development (MTD) process, a technology de-risking process, a technology verification (and/or certification) process, and a "hands-on" technical assistance process. Through the use of this roadmap process, many barriers that impede acceptance of new technologies have been identified and mitigated.

INTRODUCTION

Many contaminated sites exist within the U.S. as a result of nuclear weapons production activities during the past 50 years. According to the 1995 Baseline Environmental Management Report (1), the life-cycle cost estimate to clean up these contaminated sites ranges from $200 to $350 billion in constant 1995 dollars, with a midrange estimate of $230 billion. Over 90% of this total cost will be incurred during the next 40 years, with the remainder lasting to the year 2070 for operation of a few large waste treatment facilities. Because of significant uncertainties, many assumptions were made to derive this estimated range. While the estimate is continually being revised and can be expected to decrease based on newly available data, the nation's contamination problems remain great in scope as well as costliness.

The significant uncertainties cited in the report relate to the lack of characterization of contaminated sites, the technical risk associated with remedial solutions, public and regulatory acceptance of "effective" technologies, and future land use decisions, to name just a few. The

report further states that only one-fourth of the known 10,500 hazardous release sites has been fully characterized. Assumably, substantial reductions in the cost and duration of cleanup operations can be realized by implementing "effective" characterization and monitoring technologies. However, new technology is rarely used in U.S. Department of Energy (DOE) cleanup activities (2). The lack of acceptance of new technologies primarily results from an inadequate amount of technical information, insufficient stakeholder/regulator involvement in decision making, and a lack of teaming between technology developers and users.

The mission of the Characterization, Monitoring, and Sensor Technology Crosscutting Program (CMST-CP), a technology development program within the Office of Science and Technology
(OST) of the DOE Office of Environmental Management (EM), is to provide needed characterization and monitoring solutions in the following key problem areas, hereafter referred to as the Focus Areas:

- contaminant plume containment and remediation;
- high-level waste tank remediation;
- mixed waste characterization, treatment, and disposal;
- landfill stabilization; and
- facility transitioning, decommissioning, and final disposition.

The annual CMST-CP Technology Summary reports, available from 1993-1995 (3,4,5), detail the program's portfolio of technology development activities and their progress.

The Office of Technology Integration, within the OST, is responsible for developing and implementing roadmap processes that lead to the successful use of developed technologies to address environmental problems as well as commercialization for economic benefit. One such roadmap process has been followed for commercialization and use of emerging technologies developed under the auspices of the CMST-CP.

TECHNOLOGY TRANSFER ROADMAP

The CMST-CP technology transfer roadmap, shown in Fig. 1, consists of six main sections: needs assessment, identification of technologies for transfer, a managing-technology-for-development (MTD) process, a technology de-risking process, a technology verification (and/or certification) process, and a "hands-on" technical assistance process. The development of each technology within the CMST-CP portfolio is closely analyzed and scrutinized in accordance with the criteria established within these sections to maximize the probability for successful use and commercialization. A summary description of each section is provided below.

Fig. 1

Needs Assessment

This section encompasses needs identification, validation, and market study. CMST needs were recently identified through site visits conducted by the five Focus Area teams working with individual Site Technology Coordination Groups (STCGs). In addition, CMST needs published in the 1993 edition of the Technology Needs Crosswalk (6) report were validated through listed user contacts and were prioritized according to their assessment of the urgency and cost-savings impact of new technologies.

A market study report describing current and potential markets for chemical sensors and fieldable analytical instrumentation will be completed by February 1996. This report will be used, as the primary informational material for discussion, in the workshop and forum on

chemical sensors for environmental applications to be held in conjunction with the 1996
Pittsburgh Conference. Further market studies are scheduled, and will focus on process monitors and controls and possibly geophysical measurement techniques and data integration tools. In general, these market studies contain several elements: review of commercially available technologies, market estimates and forecasts, and identification of gaps where technology development is needed as well as prioritization of that development.
These market studies along with the customer-focused needs assessment will be used as a tool to assess the value of each technology as perceived by customers (government and private sector). The evaluation of baseline technologies and identification of specific performance characteristics affecting purchase and usage decisions by customers provide an important validity check of the needs assessment results.
Identification of Technologies for Transfer
This section involves coordinating resources and compiling databases to help identify technologies that are ready for transfer. Reports of federally funded R&D activities are becoming increasingly accessible, and many organizations sponsored by federal programs are spearheading efforts to provide on-line access to database information. In the CMST area, these resources include the Technology Summary booklets (3,4,5), the Technology Catalogue, EnviroTRADE and ProTech services, the CMST Technology Catalogue, and others. The CMST Technology Catalogue, an effort headed by Applied Sciences Laboratory, Albuquerque, New Mexico, in collaboration with the CMST Technology Transfer project office at Ames Laboratory, documents information on performance and cost of performance of 58 CMSTs. These informational materials are presented on the Internet at the address: http://cmst.ameslab.gov/cmst/homepage.html.
In addition, two versions of a "Technology Solutions" brochure have been produced. Each contains a different summary list of technologies available for transfer and their potential applications, with one targeting environmental restoration operations and the other waste management operations.
Managing-Technology-for-Development (MTD) Process (7)
The MTD process categorizes technology development into seven maturation stages (or gates): basic research, applied research, exploratory development, advanced development, engineering development, demonstration, and implementation. The criteria and requirements at each stage are clearly defined. Maturation stage information is necessary for a planned and systematic advancement from conceptualization through implementation. It is also required for resource allocation and decision making on levels of private sector involvement, user commitment, and regulatory/stakeholder involvement. This integrated planning tool is designed to address the funding gap or "valley of death" issue that occurs prominently at and after the engineering development stage. By teaming with technology developers, customers, stakeholders, and regulatory bodies, and by clearly defining the commitment required from each at individual maturation stages, an orderly hand-off process can succeed. This model has been adopted for use by the OST.
Technology De-Risking Process
This process involves providing financial support for beta-site testing of "developed" technologies to facilitate their acceptance by customers. It is applicable when a customer has specific technology performance and

cost requirements, and a technology developer can provide the needed solutions. The technology must be beyond the advanced development stage (see MTD process described above) and the prototype must be accompanied by well documented performance data. An implementation plan must be furnished by the customer describing how the technology will be used once it meets performance and cost requirements. This funding is jointly applied for by the developer and customer, and is intended to cover, on a cost-sharing basis, expenses associated with infrastructure support incurred by the customer and the equipment use, testing, and training expenses incurred by the developer. The CMST Technology Transfer project office has developed an application process and acceptance criteria, and has secured funding to support the de-risking of four technologies. This pilot activity is in the early stages and has received three applications which are listed in Table I. Two are being evaluated, and one has been funded and testing has begun.

Table I

Technology Verification (and/or Certification) Process

This process is designed to gain acceptance of a technology by a wide base of customers. Among the many barriers affecting customer acceptance, two formidable ones are: the robustness and completeness of performance data sets, and varying requirements from site to site. Many documented performance data are site-specific and on occasion are obtained under best-controlled conditions. These data sets lack completeness because of the uncertainty of whether they can be reproduced under different test conditions or sample matrices; they lack robustness because of an inadequate level of quality assurance/quality control (QA/QC) data management.

Many state regulatory authorities, federal departments (Department of Defense, Department of Commerce, and DOE), and regional and national Environmental Protection Agency (EPA) offices are teaming to develop working models to address technology performance verification and reciprocity issues. One such model was implemented for the RCRA and Other Heavy Metals in Soils Demonstration. This activity was jointly sponsored by MSE, Inc., the prime contractor for the DOE Western Environmental Technology Office; Sandia National Laboratory, a technology verification entity under contract to the EPA Consortium for Site Characterization Technologies (CSCT); and the Ames CMST-CP and Technology Transfer projects office. Four technology developers participated in the field demonstration activities during the week of September 25, 1995, in Butte, Montana. These developers, their technologies, and the assayed metals are listed in Table II.

Table II

Split samples were archived and sent to two EPA CLP laboratories for confirmatory analyses. The final evaluation report, scheduled to be issued by the CSCT in March 1996, will not only detail the performance (and cost of performance) of demonstrated technologies against individual developer or vendor claims, but also comparatively evaluate these field performance data with results gained from using baseline, laboratory EPA CLP methods. The performance of the four technologies under differing matrices and test conditions will be documented in accordance with a well defined QA/QC data protocol.

The CMST Technology Transfer project is pursuing the opportunity to participate in the recently announced "Rapid Commercialization Initiative," a coordinated effort between the Department of Commerce,

Department of Defense, EPA, and DOE. In addition, verification activities dealing with continuous emissions monitoring technologies and fieldable technologies for the detection of volatile organic compounds (VOCs) and radionuclides in groundwater are being planned. Progress toward these planned activities will be presented at the conference.

"Hands-On" Technical Assistance Process

This process is designed to bring the best-available technological solutions to DOE sites and to assist users with implementation. Such assistance encompasses assessing site problems, researching best available technologies for solving problems, providing on-site technical services, linking site personnel with vendors, helping make equipment/instruments available for testing, devising test plans, and providing training support to site personnel.

Two approaches are used in this process. The first is to market technical assistance services directly to DOE site managers and contractors. A case study showing how the CMST-CP worked with the Westinghouse Defense Waste Processing Facility (DWPF) exemplifies this method. During a visit to the DWPF in Aiken, South Carolina, in July 1995, CMST-CP and DWPF site personnel identified the need for 20 technologies that could potentially reduce the life-cycle cost of high-level waste processing operations. Of the 20 technology needs, eight could be met through further technology development; the remaining 12 needs could be met with existing technologies. To solicit the needed technology development, the Tanks Focus Area and the CMST-CP jointly issued a Call for Proposals. Five proposals were subsequently selected and awarded. In relation to the 12 technology needs with ready solutions, the CMST-CP requested detailed information about functional and operating requirements, waste matrices, and physical requirements (size, weight, power consumption, shape) from DWPF personnel. A technology solution package for each identified need will be created after adequate information is received and assessed.

CMST-CP personnel also visited the Argonne National Laboratory site to investigate the VOC and radionuclide contamination problems at the 317/319 waste sites. Further progress relating to these activities will be presented at the conference.

The second approach is to partner with environmental firms to collaboratively provide hands-on technical assistance to DOE sites. Currently, environmental firms are contracted to perform many of the tasks described above at DOE sites. By closely teaming with these firms and site personnel, the CMST-CP can incorporate its broad knowledge about the merit of emerging technologies into the planning process for developing effective technological solutions, which environmental firms and site personnel can then implement. This approach is in its infancy stage; many follow-up activities are expected to develop.

CONCLUSION

The CMST-CP technology transfer roadmap process identifies and mitigates many barriers that impede acceptance of new technologies. Completeness and robustness of technical information can be achieved by following a well developed QA/QC data management plan as an integral part of a field demonstration guidance plan. The technical performance and costs of new technologies are evaluated and verified against vendor claims and baseline technologies under real-life field conditions by working closely with regulatory bodies and technology developers and users. The resulting field performance and cost information can then be furnished to potential

customers who are responsible for making characterization and remediation decisions.

The "valley-of-death" funding gap is justifiably identified as one of the main barriers to transferring technologies successfully. The CMST-CP approach of pairing technology developers with users as well as providing beta-site testing support and hands-on technical assistance is aimed at mitigating this issue by awarding limited funding support to leverage large resources from other sources including site operations and technology developers and users. Two notable successes include field testing the Sandia robust hydrogen sensor for use in Hanford high-level waste tanks and the teaming activities with the Tank Focus Area and DWPF site personnel. While the funding gap issue remains largely unresolved, a process to use limited resources to achieve the highest return is being developed and practiced.

ACKNOWLEDGMENTS

This work was jointly sponsored by the Office of Technology Integration and the CMST-CP of the Office of Science and Technology within DOE.

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APPLICATION OF COMMERCIAL NUCLEAR PLANT WASTE MINIMIZATION PRACTICES TO DOE FACILITIES

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ABSTRACT

This paper examines the need for, and feasibility of, an aggressive waste minimization and volume reduction program at the Savannah River Site (SRS). Commercial nuclear plants have been wrestling with the high costs

of radioactive waste disposal for several years and have developed proven programs that may be beneficial for DOE nuclear facilities as well. Though the cost of radioactive waste disposal is typically lower at DOE sites than commercial nuclear plants, dwindling budgets for government projects require closer scrutiny of costs.

It was concluded that significant cost savings could be easily achieved by implementing selected commercial nuclear industry's innovative strategies, techniques, and technologies for eliminating radioactive waste. Analysis of the implementation cost of Commercial Strategies indicates a clear cost benefit with implementation costs typically recovered within the first year.

Acknowledging that the private sector has set the standard for waste minimization, SRS is partnering with private enterprise to lead Savannah River and the rest of the DOE Complex to the cutting edge of waste minimization technology and practices.

INTRODUCTION

The end of the Cold War and shrinking Federal budget has led the DOE Complex to look for innovative means to improve the quality of existing programs while cutting costs. Each DOE site's ultimate survival depends on an aggressive, results-oriented approach to getting more done with less, while protecting the health and safety of workers and the public. Eliminating waste must become a state of mind for every site initiative (1).

The challenge of surviving in a global economy has forced the private sector to develop methods for working smarter and producing more with less. The commercial nuclear industry has developed innovative strategies, techniques, and technologies for eliminating and reducing waste. Acknowledging that the commercial industry has set the standard for waste minimization, the Savannah River Site (SRS) has sought out leaders in the waste field to ensure the best practices and technologies are implemented throughout the DOE Complex.

SRS is a 300 square mile multi-facility DOE nuclear site that includes a fuel processing area, several labs, five reactors, two fuel reprocessing canyons, and several disposal facilities. This site is located on the southwestern border of South Carolina adjacent to the Savannah River and has been operated for several decades producing Special Nuclear Materials for the nation's defense programs and rare isotopes for non-defense applications such as research and space programs.

There are numerous DOE sites that have one or more facilities that are similar in nature to SRS' facilities. The waste minimization and volume reduction programs that are cultivated at SRS could be easily implemented at other DOE sites.

ESTABLISHING THE USER GROUP

To begin learning what techniques the commercial industry is using to minimize their waste, several SRS representatives were asked to form a team to tour two commercial nuclear plants, which were Susquehanna Steam & Electric Station and Beaver Valley Power station. The SRS group had a cross section of organizational lines and functions with representatives from Solid Waste, Radiological Controls, major SRS Waste Generating Organizations, and the Department of Energy (DOE). After two days of walkdowns in the commercial plants and discussions on their programs, the SRS representatives came back with many ideas on how to reduce waste and save money.

To implement programs consistently across the SRS site requires careful planning due to the variety of facilities with vastly different radiological environments. It was decided that the original representatives that went to the commercial plants would form a User group to ensure the best waste minimization and volume reduction programs were implemented for the site. Several smaller task teams were formed as necessary to resolve specific implementation details to make the programs work for the entire site.

Several other commercial plants were visited after the User Group was formed to ensure that the programs being developed at SRS included the major waste minimization practices being utilized by commercial plants in the United States.

BRING IN THE TROOPS

While visiting Susquehanna, a consultant was analyzing the plant's waste generation for further improvements in their waste minimization programs. Susquehanna found that contracting a waste minimization expert to review their program brought a different range of experience and insight than the personnel who worked with the programs on a daily basis.

At the time of the SRS visit, Susquehanna was in the process of cost cutting. Susquehanna's waste cost at Susquehanna included both waste processing and disposal costs (with disposal costs being the largest portion by far). Since Susquehanna is a part of the Appalachian Compact, they have a projected disposal cost in excess of $700 per cubic foot. Susquehanna had established three teams to address the waste reduction strategies. The three teams were to address non-radiological waste, wet radiological waste and dry active waste (DAW) reduction respectively. This was the same focus that the SRS visit was investigating.

Susquehanna had talked to other utilities and implemented many of the obvious waste reduction methods by early 1995, but still lacked the focus on details. At this point, the Susquehanna teams arranged to have an EPRI assistance visit to help identify some of the not so obvious minimization and reduction methodologies. The EPRI assistance visit coincided with the SRS visit to Susquehanna and benefited both the Susquehanna and SRS personnel.

Susquehanna received the recommendations from the EPRI team in June and has been implementing these over the past several months. The recommendations have proven to be effective in reducing the DAW generation in the last outage by nearly one half. These efforts helped Susquehanna become competitive in a deregulating utility environment.

SRS had not had a detailed waste minimization assessment performed by anyone from the commercial industry and decided to seek input from a consultant that had analyzed numerous nuclear waste minimization programs. A detailed assessment may seem costly, but if numerous waste minimization techniques can be incorporated to a waste program, it will compensate for the initial assessment cost many times over.

The SRS Solid Waste department contracted Environmental Resources and Services (ERS) Corporation to assist the major Low Level Waste (LLW) generators in evaluating and characterizing their existing LLW generation, minimization, processing, and disposal practices. The contractor was also tasked to identify a wide variety of potential mechanisms for reducing waste volumes and associated costs. Each facility's waste composition was analyzed to determine where the biggest waste reduction efforts should be concentrated. Figure 1 shows one facility's waste composition breakdown. After several facilities were

analyzed, major waste patterns became apparent that affected most facilities across the site. These waste patterns were targeted for the initial reduction effort since they would give the most cost benefit for the site.

The contractor stressed that the best waste programs incorporate the following strategies in the given order: Management Support, Source Reduction, Recycling, Volume Reduction, and Disposal. Each of these strategies have been incorporated at SRS. The contractor provided a detailed report on various programs that would eliminate the highest volume waste patterns at the facilities evaluated (2).

Fig. 1

Several representatives from Susquehanna also visited SRS for similar walkdowns and discussions that had taken place at their plant. It was thought the commercial partnership would be a "parasitic relationship and a one-way exchange of information" (from commercial utility to SRS). However, this turned out not to be the case. In fact, Susquehanna visitors felt that they brought back with them a sample of one of the best waste tracking programs that they had been exposed to and plan to implement a similar program. This reciprocal visit to SRS confirmed to the participants that this is a win-win relationship since both companies' common ground is in source reduction and back end treatment of radioactive waste.

MANAGEMENT SUPPORT

According to the EPRI report that Susquehanna uses, "Solid Low Level Waste Management Guidelines", visible management support for waste minimization is the foundation for any waste reduction effort. By visible, this means visible to the employees so that waste minimization becomes a team effort. In the commercial industry, the companies with the best waste reduction programs also have the most visible management support for the effort (3). This was also stressed in the report, "Low Level Waste Characteristic Study", which was written for SRS after studying our waste programs (4).

To heighten SRS Management's waste awareness and to increase their support for waste minimization, several User Board representatives gave presentations to the senior staff over a period of several weeks. Also, an Incentive Fee milestone for reducing the site's waste by 40% was set with an award fee of $1,285,000. Senior Staff immediately recognized the advantages of a strong waste minimization program and are supporting efforts to meet the Incentive Fee milestone.

The establishment of the User Group also increased the visibility of management support for reducing waste. Several key waste managers and representatives were participating in this group. At the division level, LLW Reduction Committees were formed to implement the waste reduction programs that the User Board determined to be the most cost beneficial. Waste reduction goals were turned in by the divisions to the site and tracked to ensure progress toward the Incentive Fee milestone of a 40% waste reduction.

A waste minimization training package, that included a video, was developed by SRS Site Training group. This training will be given to the majority of the waste generators on site and emphasizes source reduction practices. This training was developed after viewing several waste reduction tapes from specific commercial power plants who were recognized in the industry as having excellent waste management programs.

GOING FOR THE GREEN

In April of 1995, the procedure for controlling materials in radiological areas was revised to more clearly reflect what the DOE Radiological Controls Manual had intended. One significant change allowed equipment/material (non radiological materials) stored or used in Radiological Buffer Areas (RBAs) be released without a Radiological survey if the owner/custodian/user believed there was no potential for contamination. RBAs are maintained at Clean Area limits which are the following: tritium <1000 dpm/100 cm2, Beta-gamma <200 dpm/100 cm2, and alpha <20 dpm/100 cm2.

The site was considering a program for releasing low potential waste from RBAs when the User Group visited Susquehanna and previewed one of their source reduction programs: Green is Clean. At Susquehanna, their Radiological Control Areas are similar to DOE defined RBAs. Susquehanna segregates their potentially clean waste from contaminated waste by placing the clean waste into Green is Clean receptacles. Green is Clean waste is placed on a conveyor where a gamma monitor scans the waste for release. Susquehanna saves an estimated $500,000 per year as a result of implementation of the Green is Clean program.

Since SRS RBAs are maintained "clean", waste that the owner/custodian/user believes has no potential for contamination can be thrown into Green is Clean receptacles or taken directly out of the RBAs and thrown away as sanitary waste. Radiological Controls (RC) personnel perform surveys of the RBA areas and Green is Clean receptacles to ensure Clean Area limits are being maintained. Green is Clean waste that is removed from the specially marked receptacles is shredded before being disposed of as sanitary waste to ensure radiological markings are defaced. A percentage of the waste is surveyed either before or after being shredded by specially trained personnel to ensure that contaminated material has not been accidentally placed in a Green is Clean receptacle. Incoming Green is Clean waste is specifically tagged with the name of the waste packager, date, and facility from which the waste originated. This allows proper tracking if contaminated material is found in the Green is Clean waste. To date, no waste has exceeded the Green is Clean radiological limits.

Currently, Green is Clean has been implemented in Excess Facilities and Reactor Fuel Storage Division which contains eight facilities and four different radiological environments. 330 cu-ft per month of Low Level Waste (LLW) is being re-directed to Green is Clean for a yearly savings of $160,000. The cost savings far exceeds the implementing cost of the program which was < $10,000. SRS will be implementing this program sitewide in the upcoming months for an estimated annual LLW avoidance of 25,000 cu-ft and savings of $1,000,000. The Green is Clean program is the highest priority on the User Group implementation schedule. This is considered a source reduction program and the implementation costs are inconsequential, whereas the cost savings are tremendous.

WASH IT

While touring the commercial power plants, washable waste bags and tarps were observed being used as a recycling technique. Plant waste was being gathered into washable bags and taken to a waste segregating station where the waste bags were then reclaimed by laundering. Tarps were being used for staging high contamination work that use to require plastic coverings. Even though SRS has not decided to initiate a centralized waste segregating station, launderable bags and tarps were immediately recognized as a potential large waste savings since the site uses more

than 100,000 radiological bags and 200,000 square feet of plastic for transporting radioactive material.

To implement the use of launderable bags, various sized bags were designed to transport a large variety of materials, such as: radiological vacuum cleaners, hot tools, and damp launderable mops. Specialized launderable bags were specifically designed for the removal of 5'x 5'x 16' racks used to store irradiated fuel. The racks are typically lifted out of the basin and double wrapped in plastic, but the launderable bag allowed the rack to be placed right into the bag, zippered and ready to go in about ten minutes. The racks were then transferred to another part of the facility for volume reduction and the bag sent to the laundry for reuse. In all, 14 racks were removed for a waste minimization of 76 cu-ft and total waste cost avoidance of $ 3,780, which does not include the savings from the reduced manpower hours required to prepare the racks for transfer. Another type of bag has been designed to transport fuel casks from one facility to another. The waste savings on this project alone will be a minimum of 4000 cu-ft, plus the savings from numerous man-hours being eliminated for crane manipulation of the cask and wrapping.

Tarps are available to replace plastic being used to protect floors during high contamination work or for wrapping materials for transfer. The site's facilities are located several miles apart in some cases and material must be transported by truck or rail. Historically, the items being transferred were double wrapped to prevent wind damage. The plastic was thrown away as LLW once the item arrived at its destination. By using the washable tarps and bags, a waste minimization of 40,000 cu-ft is estimated once the entire site utilizes these launderable items.

Launderable bags and tarps that have been designed for SRS radiological work are featured in a catalog that can be viewed on the site computer network. An inventory is maintained so that launderable bags and tarps can be utilized immediately by the many facilities at the site.

The commercial industry has also converted to washable mops and rags. SRS uses 450,000 atomic swipes per year for general area smears, absorbent material, and decon cleaning which converts to 4,500 cu-ft of waste per year. By utilizing washable rags instead of atomic swipes, a cost savings of $180,000 can be realized. The same can be said for mops which SRS uses to a tune of 13,379 per year and is responsible for 3,000 cu-ft of LLW. By switching to launderable mops, a cost avoidance of $100,000 can be realized. Currently, five facilities have implemented the use of launderable rags and mops and are estimating a waste minimization of 700 cu-ft annually.

If SRS implements a full launderable program, a total of 98,000 cu-ft of waste can be eliminated (4). In one division where the launderable program has been implemented, the implementing costs were <$40,000, with an annual cost savings is $268,074.

Washable protective clothing (PC) is used throughout the industry, however, the quality of the clothing greatly affects the viable lifetime. At the many plants visited, various types of PCs were observed that seemed superior to those used at SRS. At SRS, a poly/cotton PC is used that has had some shrinkage problems and a looser weave than the ones observed in the commercial plants. Currently, several commercial protective clothing vendors have submitted samples of their wares to SRS for testing. A task team will evaluate the results and procure a more durable PC so that unusable PC waste can be reduced. The PC task team is also reviewing the radiological limits that require PCs to be thrown away

instead of being laundered. The SRS radiological restrictions for sending PCs to the laundry seem to be more restrictive than some DOE sites and all commercial nuclear plants, even though many of the plants are using the same laundering company. If the radiological laundry limits are raised, many PCs can be laundered instead of becoming LLW before they are worn out. In two months alone, more than 16,000 sets of protective clothing were discarded in the trash and lost to one facility. At 2 cu-ft and $20 per set, this one example represents $320,000 in replacement costs and another $1,472,000 in disposal costs (4).

REUSE IT

A prefabricated containment program has been implemented at SRS as a source reduction and recycling technique. Glovebags have been used extensively by the Navy and pre-fabricated huts have been used throughout the commercial industry. In the past, huts at SRS were hand-built out of scaffolding, plastic, and tape for each job. Once the job was completed, all the plastic and sometimes the scaffolding, would be disposed of as LLW. When hand-built huts are used outdoors for an extended period of time, the walls and seams must be maintenanced and retaped many times to maintain its integrity. The site shifted toward prefabricated huts because they were more easily installed and dismantled and can be reused many times. The outdoor huts can be reused at least five times (one month intervals) before they are considered too weathered and the indoor huts can be reused indefinitely. All the seams are heat sealed, so the use of tape is eliminated.

The site is moving toward the increased use of glovebags since they contain contamination to a smaller area than huts and allow the workers to wear less protective clothing. By containing the contamination to such a small work area, an insignificant amount of waste is produced when compared to work performed without containments.

In the past, procurement delays caused many projects to forgo using containments. To eliminate this problem, a central issuance station has been established where containments are stocked so that pre-fabricated huts or glovebags can be readily available as radiological work evolves. The time and effort to set up the issuance station was initiated since the cost savings associated with the use of pre-fabricated containments is approximately $10,000 per use when compared to hand-built huts. Approximately $20,000 in man-hours were spent setting up the issuance station and catalog but over $1,000,000 has already been saved by using prefabricated containments from this centralized location.

REDUCE IT

As the commercial power plants face eventual decommissioning and dismantlement, numerous technologies have been developed to volume reduce the radiological waste produced from this effort. SRS has large volumes of legacy waste that will also need to be volume reduced. A centralized location was established having three major decontamination technologies: CO2 Blasting, Vacuum Grit Blasting, and Kelly Vacuuming.

For low to moderate decontamination efforts, CO2 Blasting and Kelly Vacuuming will be used. The CO2 Blaster works by impelling carbon dioxide pellets against the material surface causing the pellets to sublimate and lift the contamination to a high flow air stream and eventually to a HEPA filter. CO2 blasting is very effective on removing transferable contamination and contamination fixed in an oxide substrate or painted surface. CO2 blasting is also very effective on softer surfaces, such as lead bricks or computer key boards.

Kelly Vacuuming uses a contained super heated water stream and a vacuum system. The super heated water flashes to steam when it hits the contaminated surface and lifts the contamination from the substrate to the vacuum system. This method is especially effective on concrete and porous surfaces.

For aggressive decontamination efforts, the Vacuum Grit Blaster is used. The Vacuum Grit Blaster uses metal grit and a contained vacuum system to blast fixed contamination from metals and concrete.

The above methods may not be used in all cases due to impracticality or reduced cost benefit. For volume reduction of items, a large metal shredder and shearing system is being procured for use in the latter part of 1996.

At this time it is not possible to accurately estimate how much LLW will be reduced due to the new technologies since the equipment can be used to: reduce contaminated area square footage, clean individual pieces of equipment for salvage, clean equipment for repair with less waste generated, and shred equipment for better use of disposal space. A 5,000 cu-ft reduction in waste disposal is a conservative estimate for the first two years.

BURN IT

SRS is in the process of starting up an incinerator facility after reviewing the incinerators used by the commercial power plants which have a volume reduction ratio of 99:1 for their DAW waste. SRS has determined incineration of waste will elongate the life of our burial vaults by several years. To ensure that the incinerator will have feed material after startup, SRS is converting many current waste items to incinerable materials. One of the biggest changes has been converting poly-vinyl chloride (PVC) waste bags to polyethylene waste bags. Another big ticket item has been converting PVC plastic used for wrapping waste to materials that can be incinerated. For waste that can be converted to items suitable for incineration, a reduction of 100,000 cu-ft can be saved from taking up space in the on-site burial vaults.

CONCLUSION

At this point it should be noted that if each of the above mentioned programs is fully implemented at SRS, the waste reduction estimates are not additive. To illustrate this point, implementing the Green is Clean program takes away waste that could be replaced with launderables. Implementing the launderable program will eliminate waste to be incinerated. For the different sites within the DOE Complex, implementing a chosen few of the mentioned programs may be more cost effective than implementing all the suggested programs. Programs with low implementation costs should also be considered for initial waste minimization efforts. The programs mentioned here are not the only ones being considered for SRS. Other programs such as Contamination Area Rollback, 100% Recycle Zones and Second Sort programs are also being considered for implementation at a later date but may be better choices for some sites as their initial waste minimization initiative. In almost all cases, source reduction programs should be the initial consideration for each site.

Due to the similar radiological environments and regulations that each DOE site shares, waste programs could benefit by incorporating the SRS strategies that better emphasize Management Support, Source Reduction, Recycling, and Volume Reduction. Even though many of the DOE sites do not have the disposal costs that the commercial industry has, it is obvious

that tremendous savings can be achieved by implementing a few waste minimization techniques and technologies. Through our partnership with the utilities, each DOE site will have the opportunity to become more cost effective by reducing waste generation volumes.

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Session 49 -- MANAGING & MODELING COSTS TO OPTIMIZE SITE CLEANUP
Co-chair: Richard Bastar, Parsons Engineering Science, Inc.

49-1

A NEW TECHNIQUE FOR COVERING THE FINANCIAL RISK OF A WASTE MANAGEMENT PROGRAM

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ABSTRACT

This paper presents a new financial technique developed to assist a nuclear waste agency in its funding strategy.
Such an agency has a major commitment to cover the totality of the costs generated by the waste entrusted to it. Most of the time, however, planning under uncertainty makes this task difficult. Provisions set aside in application of the "(Waste) Producer Pays Principle" may prove insufficient when it comes to completing the necessary technical programs, sometimes in a remote future.
To hedge against the threat of involving future consumers or tax-payers in the payment of past or present liabilities, adequate and robust financial techniques must be provided.
Such an approach is therefore valuable for all parties involved, namely agency, waste producers and general public.
The main lines of the approach are described in this paper. The basic idea is to have each waste producer committing himself to pay an agreed price for the future use of the services or facilities provided by the agency. The calculation of the price uses modern risk theory for financial decisions.
The paper describes in detail both aspects of how to calculate the price charged to the waste producers and how to invest the funds collected herewith.
It shows how the three main sources of uncertainty in a waste management program can be substantially reduced using the described approach:
- systematic reduction of the waste arisings in the course of time;
- wrong assessments of costs due to adverse trend evolutions, fluctuations or technical scenario changes;

funding strategy.

In conclusion, the paper provides information on the practical implementation of this technique by a nuclear waste agency.

GENERAL BACKGROUND

The fundamental principle for financing the Belgian radioactive waste program is that those who produce the waste should pay for all services to be performed ("Producer Pays Principle").

ONDRAF/NIRAS, the Belgian Agency for Radioactive Waste and Enriched Fissile Materials has the mission to collect radioactive waste and to fulfil all necessary steps till its safe final disposal.

According to the above principle, the agency works at cost price with complete financial transparency with respect to the producer. For that purpose the agency has established a financing mechanism on the basis of fees charged on each volume unit of waste delivered, in order to ensure complete financing of all the operations to be performed. For disposal operations the fees are paid into a fund which is interest bearing.

The accurate determination of the fees is not an easy task, considering the large technical and financial uncertainties, particularly for operations to be performed in a remote future.

Good modelling of the waste management system is therefore required to determine accurate fees which limit the risk of insufficient financing becoming a burden for the future community.

A new technique is presently being developed at ONDRAF/NIRAS which will hopefully reach this aim of reduced risk while respecting the fundamental principle of financing. Although in its present form, it is primarily tailored to meet the direct needs of ONDRAF/NIRAS, this approach may be useful for other radioactive waste agencies. In the following sections, we first describe the traditional approach and its drawbacks. We then sketch the new technique and review its practical application.

SUMMARY DESCRIPTION OF THE FINANCING APPROACHES

For a better understanding of the reasons that have induced the agency to review its approach, it is necessary to describe the traditional approach and its drawbacks.

Model of Traditional Actuarial Calculation

The traditional system of financing the various stages in radioactive waste management can be described as follows:

Calculation of the Unit Fees at Cost Price

A reference program is developed. It includes:

The waste removal schedule per year.

For reasons imposed by the concern for financial conservatism, the volume of waste considered does not include waste generated by the decommissioning of nuclear facilities (power plants and others) owing to the major uncertainties affecting the volumes produced. These uncertainties concern both the volumes and the moment on which the waste is collected and hence invoiced by the agency.

This schedule is therefore prudent from the financial point of view, but probably quite realistic as far as participation in the medium term is concerned.

The technical scenarios describing the waste management operations planned.

The calculation comprises the three following steps:

an economic calculation made on the basis of hypotheses on the various cost items expressed in constant francs of the year under consideration.

an actuarial calculation made on the basis of hypotheses of financial expenses and proceeds. This method makes it possible to calculate a unit cost price per m of waste, fixed for each management operation. The receipts and expenditure balance each other at the end of the operation.

sensitivity analyses, made on the basis of the reference program (schedule and technical scenarios) and of the financial hypotheses. These analyses make it possible to calculate the uncertainty margins added to the unit cost prices to form the per-unit-fee applicable to waste quantities.

The assumptions behind this calculation are the following:

a. The costs related to the services can be estimated with sufficient accuracy. For long-term operations like disposal, knowledge of these costs becomes more detailed with time and as industrial experience is accumulated for those services that are not yet operational.

b. The fee income can be estimated with sufficient accuracy on the basis of removal schedules planned in the long term for the whole of the Belgian program. Knowledge of the program also becomes more detailed with time. The increases or decreases in value of the invested assets are added to - or deducted from - the fee income.

c. The uncertainties related to expenditure on the one hand and to the schedules on the other hand can be evaluated satisfactorily by means of sensitivity analyses. An overall contingency margin is obtained.

d. The financial risk is reduced to its reasonable minimum by a judicious choice of the discounting rates. The financial margin takes account of the residual risk. Detailed Asset-Liability-Management-studies are performed to verify the adequacy of investment strategies of provisions and funds.

e. The distribution of costs between producers is based on a technical calculation of keys (schedule + scenarios), not taking into account uncertainty about actual programs.

Contractual Relationships

As far as contractual relations are concerned, every effort is made to negotiate open-ended contracts with (a) liability clauses to be included in the event of unforeseen costs; (b) clauses concerning the annual notification by the producer of any modifications to his production program; (c) clauses concerning fee adaptations (= cost price + contingency margins) made by the agency, which take into account interdependence in time and with regard to the other producers.

Drawbacks

This approach has shown some important drawbacks which call for a major revision. The main drawback relates to the high sensitivity of the unit fee with respect to the underlying assumptions of the reference program. Large fluctuations are the result. Although the frequent review of cost prices entails in principle high flexibility and, as a consequence, a limited risk of insufficient means, practice has shown that things are much more complicated. The basic hypothesis underlying the model is that the market, represented by the waste producers, would accept any price. This proved to be wrong. Indeed, commercial companies cannot easily accept practically unpredictable changes of price. Therefore, the traditional approach is rather fragile and perhaps unable in the long term to reduce the risk of insufficient coverage of future liabilities. This observation has triggered the need for a more robust approach, which the agency calls "reservation of capacity ". It is sketched below.

Model of So-called "Reservation of Capacity"

Calculation of Costs Attributable to the Producers

The differences with regard to the actuarial calculation, which is clearly "financial" in its approach of time, are considerable. This new model is indeed more "economic". It is more static, hence more prudent and robust in its way of anticipating and handling the time-related uncertainties which can be significant for certain stages of radioactive waste management.

This model has major consequences for the economic studies:

as far as the waste removal programs are concerned, the detailed removal schedule becomes purely indicative of an average distribution in time around a center of gravity. The producer firmly announces the total quantities - not revisable downwards but possibly upwards -, the "capacity" scheduled for removal of his waste.

This method makes it possible to distribute the costs in an objective way among the waste producers, and to determine the full "capacity" needed for the waste management operations;

as far as the technical scenarios are concerned, the detailed cost schedule becomes purely indicative of an average distribution around a "center of gravity", in the same way as before.

Replacing detailed schedules for quantities and costs thus avoids the main pitfall of the actuarial approach, described in section 2.1, which is very sensitive to any change in timing.

Moreover, the uncertainties no longer appear a posteriori in the sensitivity calculations, but are taken into account a priori considering systematic and stochastic uncertainties for each cost item.

The calculations are based on the following principles:

a distinction is made between "fixed costs", independent - within certain limits - of the quantities removed, and "variable costs", proportional to the quantities actually removed in the future;

the fixed costs are distributed among producers according to the "reservation of capacity" principle. Each producer makes a binding minimum commitment to cover his part, regardless of the future fluctuations of his actual program. This commitment takes the shape of an irrevocable guarantee on behalf of the producer.

Each cost item, besides being fixed and/or variable, reveals certain specific uncertainties:

systematic uncertainties, often linked to the lack of conservatism of the project engineer who tends to underestimate the cost driftsa, but also arising from differentials with regard to the economic evolution (measured by inflation or PCI), which are evident in certain activity sectors, for instance in the building trade or in the raw materials industryb;

Systematic factors are taken into account in certain cost items, when it clearly appears that the engineer's assessment is an underassessment of realistic costs, or that a drifting growth can be expected

(1) Average of the realistic cost item =

"Engineer's assessment" x systematic drift factor

stochastic uncertainties, linked to the imperfection or inaccuracy of the a priori knowledge of the actual cost at the time of its occurrence. This type of uncertainty can be represented in different ways, according to the type of analysis performed:

Rectangular: a range around the average value given in (1) and for which a sufficient confidence level can be given (90% of the cases are in this range);

Gaussian: the standard deviation of a gaussian distribution the average of which is given by (1);

Distribution: another distribution, possibly multimodal or asymmetrical, to be specified, centered around an average value (1).

Certain uncertainty distributions can be correlated with each other, for instance the cost of engineering and the personnel costs. Ignoring these correlations amounts to adding uncertainties in a too conservative manner. Consequently, sufficient attention is paid to the identification of uncertainties evolving concurrently.

A global uncertainty factor to be assigned to the fixed and variable costs of each operation is derived by combination of basic uncertainties.The aim of the calculation is to determine the adequate fee to cover the costs at a high confidence level of at least 90%.

Contractual Relationship

The contractual relationship is basically different in timing and in scope from the traditional relationship. While in the latter, the duration and extent of services are simple assumptions based on the reference program, the former implies a firm commitment of both parties. The contract stipulates in detail the precise request of the producer regarding waste management, such as waste types, quantities, and operations to be performed.

In exchange, the agency quotes a price valid for a minimum time period of 10 years. It includes the fixed costs with uncertainties, incurred whatever the actual quantities of waste to be handled.

The fixed part of the price represents a contractual guarantee to be paid by the producer regardless of his actual future program. It is interpreted "as share of the use" of resources of the agency, and it is proportional to the quantities agreed on in the contract. The payments are made, as in the traditional approach, by (fixed and variable) unit fees levied at delivery date on each unit of waste. To take into account either interests on invested money or opportunity cost of capital, the fees escalate each year, beyond inflation, by a constant risk free interest rate.

The payments related to the fixed part of the fee come in deduction of the contractual guarantee.

The conditions agreed upon cannot be revised before the term of at least 10 years. Should the producer exceed the originally planned "reservation of capacity", the guarantee is increased accordingly.

At the end of the contractually agreed period, the waste producer is confronted with the decision of reviewing or of terminating the relationship with the agency.

This decision may be either

Decision 1: the producer decides to end his activities. In this case, he completes outstanding payments related to the fixed costs, i.e. the part of the guarantee that has remained unpaid at this time ;

or

Decision 2: the waste producer decides to sign up for a new period. He provides the same information as previously regarding his waste management needs. The agency reassesses the guarantee level, using the information available at the time, and taking into account the new request as well as the past program and payments. To assure the continuity and the most equitable share of fixed costs among all producers, the new request shall in no case be inferior to the total waste program announced in the previous contract.

FINANCIAL RISK HEDGING USING THE NEW APPROACH

The new approach is characterized by a low financial risk taken by the agency, as fixed costs are covered in their totality with a high confidence level. At the same time, it limits the planning uncertainties of the waste producers, as the future payments are completely known or a known function of the actual waste delivery program. The portion not yet paid, is part of the total guarantee which will only become extinct after all commitments have been fulfilled.

In the wording of modern asset valuation theory, the producer buys a "call option" on the price imposed by the agency for handling his waste. This option is implicitly exercised whenever the actual cost exceeds the agreed price.

The agency sells this option and gets the risk premium, which represents the expected value of the excess costsc. Therefore, adequate financial means are available for the agency, whatever the actual waste delivery program will be.

For funds and provisions, excess return beyond the constant risk free escalating rate of the fees provides additional hedging against unforeseen drifts in the costs.

CONCLUSIONS

The new financing approach significantly reduces the risk of insufficient financial means in carrying out the waste management program.

It should be noted that for financing radioactive waste management, Sweden uses a "reservation of capacity" approach, while Finland has implemented a guarantee system for funding means that are not yet available.

In Belgium there is reasonable confidence that waste producers are in favor of a system combining these two techniques. In contrast to the usual net present value approach, it substantially reduces large sources of uncertainties, i.e.,

- systematic increases or fluctuations in costs ;
- changing technical scenarios or revised operation schedules ;
- systematic reductions or fluctuations in waste volumes to be handled by the agency;
- investment policy of funds and provisions.

This increased robustness before large uncertainties, fully benefits to the agency, as well as to the producers themselves and to the public at large. It eliminates unwanted price fluctuations, and the threat of additional payments for past inaccuracies.

Therefore it also fully responds to the requirements of the "Producers Pays Principle" towards the present and future collectivity.

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ON-SITE VS. OFF-SITE MANAGEMENT OF ENVIRONMENTAL RESTORATION WASTE: A COST EFFECTIVENESS ANALYSIS

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ABSTRACT

The Sandia National Laboratories Environmental Restoration Project is expected to generate relatively large volumes of hazardous waste as a result of cleanup operations. These volumes will exceed the Laboratories' existing waste management capacity. This paper presents four options for managing remediation wastes, including three alternatives for on-site waste management utilizing a corrective action management unit (CAMU). Costs are estimated for each of the four options based on current volumetric estimates of hazardous waste. Cost equations are derived for each of the options with the variables being waste volumes, the major unknowns in the analysis. These equations provide a means to update cost estimates as volume estimates change. This approach may be helpful to others facing similar waste management decisions.

BACKGROUND

Sandia National Laboratories (SNL), a Department of Energy (DOE) laboratory located in Albuquerque, New Mexico, is aggressively implementing an Environmental Restoration (ER) Project that is planned to be completed in the year 2000. There are currently 157 solid waste management units (SWMUs) identified as candidates for assessment and remediation at SNL's New Mexico facilities and test areas. These sites include landfills, septic systems and drain fields, firing sites and burn pits, outfalls, surface storage areas, underground tanks, and miscellaneous test areas and discrete sites. The waste materials, or contaminants of concern, include hazardous constituents, low-level radiological constituents, mixed hazardous and low-level radiological constituents, and some Toxic Substances and Control Act (TSCA) regulated constituents, as well as sanitary or non-regulated constituents. The total estimated volume of ER waste likely to be generated is 60,800 cubic yards, including 37,000 cubic yards of hazardous, 2,500 cubic yards of low-level radioactive, 3,700 cubic yards of mixed, 5,700 cubic yards of TSCA, and 11,900 cubic yards of non-regulated.

The SNL ER Project is regulated under a Resource Conservation and Recovery Act (RCRA) Hazardous and Solid Waste Amendments (HSWA) permit. As such, there is a specified schedule for the major assessment and remediation activities associated with each SWMU. The relatively large quantities of waste that are to be generated by the ER Project will exceed the waste management capabilities of SNL, which are sized to support only the day-to-day operations. Consequently, the ER Project must develop an independent strategy for waste management to assure that it can meet the HSWA permit schedule and do so with maximum efficiency. A temporary unit (TU) to store ER-generated waste for one year, with a possibility for a one-year extension, and a corrective action management unit (CAMU) to store, treat, and possibly dispose of ER waste over the duration of the ER Project, are critical elements of the SNL ER strategy. The Environmental Protection Agency (EPA) promulgated the TU/CAMU Rule in 1993 specifically to give greater flexibility to owners of large restoration projects that were being constrained by lack of treatment, storage, and disposal (TSD) facilities capable of meeting land disposal restrictions (LDRs) and minimum technology requirements (MTRs). The TU/CAMU Rule provides the means to excavate ER waste without triggering either LDR or MTR requirements because the waste is not considered to be

"generated" if it goes to a TU/CAMU for treatment, storage, and/or disposal.

Once it was determined that a waste management strategy utilizing both a TU and CAMU might offer reduced treatment, storage, and disposal costs as well as an accelerated cleanup schedule, it was decided that analyses should be performed to verify benefits and quantify the potential savings. This paper summarizes the results of an analysis of costs (and savings) for each of four waste management options. A previous study concluded that the availability and cost of off-site treatment and disposal for the estimated volumes of ER-generated radiological, mixed, TSCA, and non-regulated wastes was such that the CAMU could be used for hazardous wastes only, thus simplifying the permitting process. Therefore, the options analysis summarized below includes only hazardous waste volumes and associated treatment technologies in its underlying assumptions. The results of this analysis will be used to determine the optimal solution for managing ER wastes, in terms of cost, under various waste volume scenarios.

COST ANALYSIS

Four discrete options for the storage, treatment, and disposal of hazardous ER wastes were evaluated to assess their cost effectiveness. Option 1 assumes that all hazardous wastes will be sent off site for treatment and disposal at permitted TSD facilities. The other three options consider various combinations of on-site and off-site storage, treatment, and disposal, and thus require implementation of a CAMU. The second option assumes that the CAMU will be used for interim storage only with off-site treatment and disposal. The third assumes that the CAMU will be used for on-site storage and treatment. Non-regulated materials, i.e., treated materials, will be reused as fill and concentrated materials will be sent off site for disposal. The fourth option is similar to the third except that the treated materials are assumed to be placed into an engineered disposal cell in the CAMU. This option is considered important because until treatability studies are completed on SNL wastes, the actual efficiency of selected treatments for the various waste streams (especially those containing toxic metals) cannot be known with enough certainty to safely assure that the "clean" treatment residues will be below levels of regulatory concern. These options are summarized in Table I.

Table I

Transportation by rail assumes that wastes can be bulk stored and aggregated until there are sufficient quantities to ship. Option 1 does not allow for on-site storage, therefore, truck transportation is assumed. Options 2, 3, and 4 provide for on-site storage, therefore, rail transportation is assumed.

Waste Volume Estimates

The hazardous waste can be subdivided into contaminated soil, debris, and personal protective equipment (PPE) as shown in Table II. Debris and PPE are assumed to be sent off site for disposal. Decontamination water will be disposed in the sanitary sewer, if clean, or evaporated on-site. Thus, contaminated soils, which are the major hazardous waste source, are the basis for this cost analysis.

Table II

The principal organic contaminant is trichloroethylene (TCE). Of the 16,300 cubic yards of soil contaminated with metals, 12,900 cubic yards are in a landfill and are believed to be contaminated with hexavalent

chromium. The remaining 3,400 cubic yards are primarily surface soils contaminated with lead. It is assumed that the lead can be recovered in the field, and thus these soils are assumed not to be treated in the CAMU. Of the 8,400 cubic yards of soil contaminated with both organics and metals, 6,200 cubic yards are contaminated with hexavalent chromium and TCE. The remaining 2,200 cubic yards come from sites that will probably be remediated in situ. Therefore, for purposes of this analysis, the only wastes that are considered relevant for evaluating whether a CAMU is economically feasible are 11,800 cubic yards of soil contaminated with organics, 12,900 cubic yards contaminated with metals, and 6,200 cubic yards contaminated with both organics and metals.

Cost Estimating Methodology

SNL's remedial actions are predominantly at the corrective measures study stage. Conceptual cost estimates were thus prepared to compare the total project cost (TPC) for each of the options. The cost estimates assume that ER waste management activities will be performed by a prime contractor who will subcontract specific work elements to specialty contractors. It is assumed that Sandia will provide oversight of ER waste management operations.

A combination of bottoms-up and parametric estimating techniques was used to prepare cost estimates for each of the options. A work breakdown structure (WBS), based on the DOE Code of Accounts, was developed to encompass the range of activities assumed to be required for the four options. The first three levels of the WBS are shown in Table III; however, much of the estimate was prepared at the fourth level. For example, the level below Other Structures (1.2.7) includes a disposal cell, engineered cap, monitoring wells, etc. The WBS and the cost estimates exclude items common to all four options because these costs will accrue regardless of the option selected. Common items include removal of contaminated materials from the sites and the Temporary Unit. Quantities of labor, material, and equipment were estimated based on current hazardous waste volume estimates, a conceptual site plan for the CAMU (Fig. 1), and from discussions with Sandia project personnel. In general, vendors were consulted for the major cost drivers; i.e., treatment, transportation and disposal. In addition to vendor pricing, published cost estimating manuals were used to obtain costs. The Remedial Action Cost Engineering and Requirements System (RACER) was used to estimate costs for the engineered cap and long-term monitoring in Option 4.

Table III

Fig. 1

Options 1 and 2 assume that wastes are transported off site for treatment and disposal. Option 1 assumes that wastes are transported via truck, whereas in Option 2 transportation by rail is assumed. Soil contaminated with organics (and organics with metals) is transported approximately 800 miles to a RCRA landfill for incineration. Incineration disposal is estimated at $600 per ton (direct cost) based on a vendor quote and includes final disposal of the ash. Soil contaminated with metals is transported approximately 500 miles to a RCRA facility for stabilization and landfill disposal. Stabilization and landfill disposal are estimated at $185 per ton (direct cost), also based on a vendor quote.

Options 3 and 4 assume that wastes are treated on site. Soil contaminated with organics (and organics with metals) is assumed to be treated using low-temperature thermal desorption to remove organic contaminants from

the soil. The variable cost quoted for low-temperature thermal desorption is $66 per ton (direct cost). Soil contaminated with metals, including the residual from the thermal desorption process containing metals, is assumed to be treated using soil washing to remove the metallic contaminants. Metallic contaminants adsorb onto soil fines and clays, and the soil washing process separates these finer, contaminated particles from the coarser, uncontaminated particles. A 6:1 volume reduction is assumed for soil washing. The variable cost for soil washing is estimated at $150 per ton (direct cost); however, there is some uncertainty in this unit cost. One vendor quoted a range of between $7 and $500 per ton depending on soil characteristics and contaminants. Treatability studies will have to be performed to obtain more definitive costs for SNL wastes. In Option 3, treated soils are assumed to be non-regulated, and thus are reused as fill. In Option 4, an engineered disposal cell is estimated for disposal of treated (but not below levels of regulatory concern) soils. Option 4 also includes an engineered cap over the disposal cell, four groundwater monitoring wells, and 30 years of groundwater and vadose zone monitoring. The 1/6 concentrated residual from the soil washing process is sent off site for stabilization and disposal in both Options 3 and 4. Indirect costs include contractor's overhead and profit (OH&P), bond, SNL loading factors, escalation, and contingency. Prime contractor's OH&P was applied at the rate of 10% and prime contractor's markup on subcontractors was applied at the rate of 5%. Performance and payment bonds were estimated at 0.75%. SNL loads were based on fiscal year (FY) 1996 spend plan rates issued by the SNL financial organization. The June 1995 DOE escalation rates for Environmental Restoration were used to estimate escalation. Escalation was applied at the rate of 8.74% assuming a schedule mid-point of July 1998 and a five-year schedule for ER waste operations beginning in January 1996. Contingency was estimated for each WBS element based on an assessment of cost risk within the scope of work assumed for the analysis. In general, contingency was estimated at 15% for low-risk elements, and at 50% for high-risk elements. High-risk elements include operating labor, operating equipment, soil washing, and CAMU closure. The overall applied contingency rates are 15% for Option 1, 16% for Option 2, 31% for Option 3, and 30% for Option 4. According to the DOE Cost Assessment Team (CAT) Cost Estimating Handbook for Environmental Restoration, the allowable contingency range for the remediation phase of an ER Project at the corrective measures study stage is 10% to 60%; therefore, the contingency applied to each of the options is at the low end of the allowable range.

Cost Analysis Results

Table IV compares the TPC for each of the four options. Without considering the time value of money (see Sensitivity Analysis below), Options 1 and 2 are the least cost effective. The cost to transport, treat, and dispose of waste off site (Options 1 and 2) is considerably higher than the cost to treat and manage wastes on site (Options 3 and 4). In addition to the relatively lower unit costs of treating and managing wastes on site, Options 3 and 4 also avoid the associated transportation costs. Between Option 3 and Option 4, Option 3 is more cost effective because the costs associated with the engineered disposal cell and long-term monitoring in Option 4 are avoided.

Table IV

SENSITIVITY ANALYSIS

The major variables affecting cost are the volumes of wastes to be managed. Therefore, it is advantageous to have general formulae for estimating the TPC of each of the four options as waste volume estimates are continually revised. The cost estimates for each of the four options (at the given waste volumes) were segmented into fixed and variable cost components, with respect to volumes, and equations were derived such that:

TCn = a0,n + a1,nV1 + a2,nV2 + a3,nV3, where
TCn = the total cost of Option n, n = 1...4,
V1 = volume of organic waste in cubic yards,
V2 = volume of metals waste in cubic yards,
V3 = volume of organic plus metals waste in cubic yards,
a0,n = fixed costs associated with Option n, and
ai,n = the variable cost associated with Vi for Option n, i = 1...3.

Fixed costs include Pre-Title I design, CAMU capital costs (excluding the disposal cell and engineered cap in Option 4), post-construction startup activities, CAMU operating costs (excluding storage and the variable portion of treatment), and CAMU closure. Variable costs include storage, the variable portion of treatment, transportation, disposal, and in Option 4, the disposal cell and engineered cap. Following are the total cost equations derived for the four
options.

Eq. 1
Eq. 2
Eq. 3
Eq. 4

These expressions provide an effective and efficient means of estimating the cost of each option given a change in the volumetric estimate of waste. It should be noted that these equations are applicable only to SNL because the underlying costs are specifically related to the scope of work at SNL. However, the general method used to derive the model can be used by others. It should also be noted that these relationships are valid within an unspecified range for the variables. For example, at significantly higher or lower volumes, fixed costs may be appreciably different because the scale of operations, equipment capacities, etc. will be different. In other words, fixed costs will likely increase or decrease in a stepwise fashion with changes in volume. Similarly, variable costs quoted by vendors for treatment are a function of quantity. Larger quantities typically imply lower unit costs.

Other analyses that can be performed include break-even analysis and net present value. Break-even analysis can be performed in situations where there is a single contaminant of concern. The volume at which the total cost of two options is identical is the break-even volume and is determined by simultaneously solving two total cost equations. At volumes above or below the break-even volume one or the other option would yield a lower cost. Although not reported here, SNL used break-even analysis to determine the volume at which it would be more economical to send wastes off site assuming metals could be remediated in situ.

Net present value (NPV) analysis recognizes the time value of money to determine the most cost-effective alternative. Future cash flows are discounted to the present using specified discount factors. Annual cash flows are discounted to the present by dividing the cash flows, stated in current dollars, by the discount factor, (1 + i)n , where i is discount rate and n is the year in which the cash flows are realized, beginning

with n = 0 for the current year. NPV is equal to total discounted benefits minus total discounted costs in the case of benefit-cost analysis, and is equal to the total discounted cost in the case of cost-effectiveness analysis. The alternative having the greatest NPV, or the least net present cost, is the preferred alternative. The Office of Management and Budget specifies a discount rate of 7% for use in benefit-cost and cost-effectiveness studies performed for the Federal government. Activities for the four options were scheduled over a 5-year period resulting in the allocation of estimated costs shown in Table V. Table VI gives the discounted cash flows using a 7% discount rate. The discount factors used to discount the values in Table V are given at the bottom of Table VI.

Table V

Table VI

The use of present value analysis does not change the order of the alternatives in terms of cost. Option 3 is still the most cost-effective and Option 2 is still the least. However, this will not always be the case. The longer the schedule and the higher the discount factor, the more dramatic will be the effect of discounting. Also, alternatives that require large expenditures early in the schedule, will be less affected by discounting than those that are weighted more heavily later in the schedule.

In addition to cost considerations, other criteria must be considered in the options selection process. The following section addresses the criteria considered by SNL and gives the results of the options selection process.

OUTCOME

In addition to the cost analysis results, the SNL ER Project considered other, qualitative criteria in reaching a final decision on which of the four options to select. The decision-making process involved SNL and DOE technical staff and managers, the regulators and the local public. Both the TU and CAMU are permitted facilities that must go through a public hearing and comment process. Thus, the early involvement of both the permitting agency and the public is an important means for expediting the resolution of issues and thereby reducing the time to obtain the permits. The qualitative criteria included worker and public health and safety as related to each of the four options. On-site and off-site handling and transportation, CAMU operations with and without treatment and disposal, ecological impacts, aesthetics, etc. were considered and ranked as to their relative importance for the ultimate decision, and finally they were weighted (along with cost) to yield a quantitative relative value. The results of this process are shown in Figure 2. Options 3 and 4, CAMU with storage and treatment, and CAMU with storage, treatment and disposal, respectively, were clearly preferred based on the resulting values. These are also the options having the least total costcost was a dominant factor for each of the groups. (For those who may be interested, separate reports are available on the entire options analysis and on the process used to involve the stakeholders. These reports can be obtained by contacting the authors of this paper.)

Fig. 2

As an outcome of the cost analyses and the stakeholder options evaluation, SNL will pursue permitting a full treatment, storage, and disposal CAMU as defined under Option 4. The disposal cell will not be constructed until it is clear that it will be needed. That need could

result if the treated residues, especially for metals-contaminated soil, contain after-treatment concentrations above levels that would allow unrestricted reuse.

CONCLUSIONS

The least cost-effective solutions for managing ER hazardous wastes, given the constraints and assumptions of this study, are those that involve off-site treatment and disposal (Options 1 and 2). The most cost effective solution is a CAMU combining on-site treatment of hazardous soils with reuse of the cleaned soil as fill (Option 3). A CAMU combining on-site treatment of hazardous soils with on-site disposal of treated soils comes in a close second (Option 4). However, the ultimate decision regarding disposition of environmental restoration waste will depend on stakeholders' perception of risk as well as on cost.

The volume of waste to be treated and managed greatly influences the optimal solution. At a relatively low volume of waste, the capital costs associated with the CAMU may not be justified. However, if current hazardous waste estimates are accurate, then a CAMU can indeed be justified under the assumptions of this analysis. As current hazardous waste estimates change with the incorporation of newly gathered data, the total cost equations can be used to evaluate the continued economic viability of the chosen waste management options.

ACKNOWLEDGEMENT

This paper was prepared by the authors for Sandia National Laboratories, Albuquerque, New Mexico, and Livermore, California, for the U.S. Department of Energy under Contract DE-AC04-94AL85000

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ESTIMATING MANAGEMENT AND DISPOSITION COSTS FOR MATERIALS IN INVENTORY

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ABSTRACT

The Department of Energy conducted the Materials in Inventory (MIN) Initiative to assess the management practices and structures, vulnerabilities, disposition options, and costs associated with ten categories of materials that the Department no longer needs for their original purpose. One of the key findings of the resulting report (Taking Stock: A Look at the Opportunities and Challenges Posed by Inventories from the Cold War Era, (1)) is the Department's lack of sufficient understanding of the costs of managing and dispositioning most of its materials in inventory. Difficulties in estimating these costs arise from a lack of clear organizational structures within the Department to manage materials and a lack of requirements and incentives to promote identification and disposition of materials no longer needed.

The Department is addressing the difficulty in estimating costs by increasing financial and managerial accountability for materials in inventory and expediting disposition of materials.

INTRODUCTION

The Department of Energy (DOE) accumulated an extensive inventory of materials during the past 50 years to support nuclear weapons production and basic energy research. With the end of the Cold War, DOE sites and programs are scaling back and shifting their attention and resources from weapons production to clean-up and restoration. Therefore, the need for certain materials has decreased. The unused materials, if neither managed appropriately nor dispositioned in a timely manner, may cause the Department environmental, worker safety, or public health problems. Such problems could, then, lead to considerable and unnecessary management, and ultimately, disposition costs. It is imperative that the Department re-evaluate its material needs and ensure that appropriate management practices are in place for the smaller and less dynamic material inventories needed for current and future missions.

Several factors contribute to the increased need for the Department to re-evaluate and comprehensively plan its inventories and materials management practices. First, a change in missions has caused slow-down in materials usage, leading to a backlog of materials for which there is no clearly defined purpose. Second, the Department's recent commitment to openness has brought greater scrutiny to environmental, safety and health issues within the Department, allowing more dialogue about vulnerabilities associated with materials. Third, declining budgets are forcing greater attention on all expenditures, including those currently used to manage materials in inventory.

The Department must understand the costs associated with managing and dispositioning materials in inventory because decisions about several of these materials will need to be made during the course of environmental restoration and deactivation activities. Long-term costs for materials disposition may increase unnecessarily if decision makers favor waste disposal over reuse and recycling due to a lack of understanding about current costs.

OVERVIEW OF THE MATERIALS IN INVENTORY INITIATIVE

The Department undertook the Materials in Inventory (MIN) Initiative to improve management, reduce unnecessary costs, and eventually develop disposition options, thereby reducing the amount of materials held in inventory. The Initiative focused on a subset of the Department's materials, referred to as "materials in inventory," which are defined as

materials that are not currently in use, are not designated as waste, and have not been set aside by the Nuclear Weapons Council as strategic reserves (1).

Materials are considered to be "not in use" when they have not been used for a period of one year and are not expected to be used in the coming year. Nuclear materials are considered "not in use" if they are associated with an inactive program as listed in the Nuclear Materials Management Safeguards System (NMMSS).

Hundreds of materials at DOE-owned facilities potentially meet the definition of a material in inventory. The MIN Initiative focused on ten categories of materials as follows:

Nuclear materials
Spent nuclear fuel
Depleted uranium
Natural and enriched uranium
Plutonium and other nuclear materials
Lithium
Non-nuclear materials
Sodium
Scrap metals and equipment
Chemicals
Lead
Weapons components

Teams of Field and Headquarters staff were established for each category of material to gather and analyze information regarding quantities, locations, management practices and systems, vulnerabilities, costs, and disposition options.

The final report of the Materials in Inventory Initiative, Taking Stock: A Look at the Opportunities and Challenges Posed by Inventories from the Cold War Era, was delivered to Deputy Secretary Curtis in December 1995.

SUMMARY RESULTS OF THE MIN INITIATIVE

The Department is currently storing over 820 million kilograms of materials that fall within the scope of the MIN Initiative. It is difficult to assess the total inventory of materials because tracking requirements and management practices for most of these materials vary from site to site and program to program, with no Departmental level of consistency.

Teams evaluated the status of the Department's current plans for disposition including reuse within the government, recycling, sale or transfer, and disposal as waste. Disposition plans for many of these materials have not been made or implemented because barriers exist that inhibit timely decision making. For example, managers face disincentives to declare materials excess and clear responsibilities among programs are lacking. Also, very few of the materials can be considered "assets" and thus sold for profit. In fact, most of the materials will require characterization, processing, decontamination, or repackaging before they can be dispositioned, which, in some cases, will cost as much or more than the material is worth on the market. For example, the Department recently sold 91 million pounds of lithium for about the same amount of money it will cost the Department to prepare the material for sale.

Some savings can be realized from dispositioning materials in inventory. More importantly, some costs, such as re-occurring costs can be avoided by reducing inventories. For example, if the sale of the 91 million pounds of lithium does not proceed as currently planned, the lithium will have to be repackaged in 5-10 years. This repackaging will cost an estimated $20-40 million dollars.

It is essential to move materials in a timely manner, especially as decommissioning schedules accelerate. Some materials, such as chemicals and equipment, lose value as they age. In the case of some chemicals,

safety vulnerabilities increase overtime as the chemicals change form and composition. Some materials become the focus of regulatory attention when they remain on site without a use for long periods of time. Time is also an important factor as sites shut down and must rapidly disposition large amounts of materials at once. For example, the Energy Technology Engineering Laboratory in California is currently faced with getting rid of 30,000 gallons of sodium within 1 year or it may have to be managed as waste which is costly. The site is working with other sites, state regulators, and sodium manufacturers to resolve this issue.

Findings of the MIN Initiative

Taking Stock includes a number of findings, three of which are relevant to cost discussions:

Finding: The Department does not understand sufficiently the costs of managing most materials in inventory, resolving vulnerabilities, and disposal as compared with other management options.

Finding: The Department does not have clear and effective organizational structures for inventory management and disposition for some materials in inventory.

Finding: The Department lacks policies to require or encourage its programs to identify and disposition materials that are no longer needed.

The complete list of findings appears at the end of this paper.

ESTIMATING COSTS

One of the main tasks that evolved during the course of the MIN Initiative was to begin to assess how much the Department is currently, and will likely, spend to store manage and disposition materials in inventory. In most cases, adequate data were not available to make quantitative estimates for materials other than plutonium and spent nuclear fuel. However, important conclusions can be made about the causes and effects of the difficulty of gathering the data.

Methodology and Results

Cost surveys were sent to each site requesting information on management and disposition costs and potential costs to address vulnerabilities for each material. The quality and utility of the cost estimates received in the surveys vary greatly for two reasons. First, the survey responses did not provide the assumptions or calculations used in developing cost estimates, making verification difficult. Second, the survey requested only existing information; sites were not expected to develop new data for this effort. Less than half of the sites were able to respond with existing information, and not all of those sites responded to all questions. A lack of response does not necessarily indicate a lack of costs.

Cost information gathered for the MIN Initiative presents a snapshot in time of overall costs. Nuclear material inventories tend to be static while non-nuclear material inventories tend to be fluid, making cost estimations difficult. For this and other reasons, nuclear materials, such as plutonium, depleted uranium, spent nuclear fuel, and lithium, tend to have more cost information available than non-nuclear materials, such as scrap metal, lead, and weapons components. Plutonium and spent nuclear fuel seem to have the best management cost data because internal and external pressures have increased attention and programs have been established to deal specifically with these materials.

Managing Materials

Management costs of materials in inventory are estimated to be at least $1-2 billion annually. This estimate drops to about $50-60 million when

not factoring in plutonium and spent nuclear fuel costs. This estimate is not all inclusive because some sites reported that costs to store some materials in inventory, such as lead and chemicals, are negligible except when they are actually dispositioned.

The Department spends approximately $1 billion annually on plutonium management activities such as monitoring, safeguards and securities, facility maintenance, performance testing, surveillance, and packaging and storing. This is the largest identified portion of the management costs for materials in inventory. Spent nuclear fuel accounts for the second largest portion of identified costs with approximately $170 million spent annually on activities such as inspections, pool maintenance, safeguards and security, record keeping, and surveillance.

Addressing Vulnerabilities

As with management costs, data associated with addressing environmental, safety, and health vulnerabilities were best for plutonium and spent nuclear fuel. This is due, in large part, to the fact that the Department has conducted extensive studies on the vulnerabilities of these two materials and both have received focused attention from outside the Department.

The Department estimates that it will cost approximately $900 million to address vulnerabilities associated with plutonium. This number includes stabilization activities at Idaho, Oak Ridge, Savannah River, Hanford, Rocky Flats, and Livermore and Los Alamos National Laboratories as recommended by the Defense Nuclear Facilities Safety Board in 1994. The estimate for addressing vulnerabilities associated with spent nuclear fuel is approximately $677 million.

Dispositioning Materials in Inventory

The Department can save money through timely disposition of materials in inventory rather than maintaining these materials for the indefinite future. Over time the Department can reduce overall costs by avoiding the need to build new storage facilities, monitor some materials, and incur routine maintenance costs, though the initial costs for disposition will seem high.

Initial costs will be incurred because many materials will require activities such as decontamination, repackaging, or other services before they can be dispositioned. These activities often result in higher costs than the Department will immediately recover from the sale or removal of the materials from inventory. Because materials in inventory tend to be co-located with other materials, removing them from inventory will not greatly increase immediate cost savings. However, these savings could be realized in the long-term. For example, lead bricks which fit the definition of "materials in inventory" (MIN lead) may be stored in the same building as other shielding materials or may be stored with waste lead that is not considered to be a "material in inventory". If the "MIN" lead is removed, the other lead will still need to be stored and monitored as before.

Difficulties in Estimating Costs

As the teams tried to develop cost estimates for managing and dispositioning materials in inventory, it became evident that the Department does not understand sufficiently the costs associated with most of these materials for a variety of reasons.

Finding: The Department does not have clear and effective organizational structures for inventory management and disposition for some materials in inventory. Various offices within the Department have responsibility for

materials in inventory. In some cases, several offices have responsibility for managing the same material at a site. This sometimes leads to unclear lines of responsibility for managing and dispositioning the materials. At Fernald, six organizations are involved in the management and disposition of uranium. The high number of participants has slowed down the sale of materials and created confusion over which organization has authority over the material.

For some materials it is difficult to say exactly which program office owns them. Sometimes when projects are terminated, materials are "orphaned", meaning that no office is willing to take responsibility for the material. Without one program responsible for the management and disposition of the "orphan" materials, costs are often carried in general site overhead accounts. The lack of clear and effective organizational structures for inventory management and disposition leads to difficulty in estimating costs.

The Department generally does not allocate budget resources by material, but rather does so by activity. Thus, material management costs generally are paid by the sites as an overall maintenance cost, not as part of a project. Because costs usually are not associated with particular projects, programs do not see the costs of maintaining materials. Maintenance activities such as monitoring, storage, moving, and safety precautions are generally paid for out of overhead accounts for the entire site. These circumstances make it difficult to distinguish how much is spent on materials in inventory compared with other materials. Because lines of responsibility are unclear, life-cycle and opportunity costs are often not taken into consideration when making decisions about disposition. Many management costs will stay the same even if the materials go away. Reducing the volume of materials in inventory, however, may make room for other materials and thereby avoiding new storage facility construction.

Even when it is possible to estimate the cost of managing materials, it is often difficult to assign specific costs to materials in inventory. Determining the portion of costs that are attributable to materials in inventory is difficult because materials are often co-located with other materials not covered by this initiative. For example, chemicals meeting the definition of "materials in inventory" (MIN chemicals) are on shelves along with chemicals that are currently in use. When assessing the costs of managing chemicals, it is very difficult to determine what portion of costs to assign specifically to the "MIN" chemicals.

Disincentives to Dispositioning Materials

Finding: The Department lacks policies to require or encourage its programs to identify and disposition materials that are no longer needed. In fact, there are often financial disincentives to disposition materials in inventory. Many materials require sampling, decontamination and repackaging, as well as time and effort to find adequate and appropriate disposition options. Occasionally, preparation of an Environmental Impact Statement or other documentation is required by the National Environmental Policy Act (NEPA) before materials can be dispositioned. Although NEPA processes are necessary to ensure appropriate consideration of potential environmental impacts and stakeholder involvement, they do increase the amount of time and money a program must spend to disposition materials. If a program identifies and begins to disposition an unneeded material, it would bear the burden of the costs of preparing that

material for disposition. These costs could draw funds away from the program's core missions.

Outside factors also may create disincentives for dispositioning materials in inventory. For example, a storage tank emptied during disposition of a material in inventory may contain a "heel" at the bottom of the tank to collect impurities. Under the Resource Conservation Recovery Act, if the tank was emptied, the heel would be designated as a waste. Once designated a "waste" materials often require costly treatment and packaging procedures. However, if the material was left in the tank, the Department would not have to manage the heel as a waste material. Because of the possible redirection of resources from active projects to pay for disposition or preparation of materials for disposition, and the lack of cost impacts on budgets of responsible organizations, Department programs are effectively discouraged from dispositioning materials. Leaving materials in place is often the simpler option with less immediate or fewer obvious budget impacts. This short-term view can lead to greater costs in the long term as aging materials, storage containers, and facilities require additional attention.

IMPLICATIONS FOR THE DEPARTMENT

The findings of the MIN Initiative have important implications across the Department when it is managing materials, decommissioning facilities, and cleaning up sites. For example, if materials are dispositioned as they are removed from their original location, it is easier to certify the contamination and pedigree of these materials. Once materials are placed in storage, often co-located with materials from other locations, facilities, and processes, it is more difficult to certify that a material has not been contaminated in some way. Many of the Department's materials in inventory were placed uncharacterized in storage many years ago. This has lead to higher current management costs than if final disposition plans had been implemented originally. Additionally, it costs less to move a material once than it does to move it twice.

Material characterization is needed for many materials before management and disposition decisions can be implemented. Some materials have not received extensive management attention in the past and have minimal or no records regarding chemical composition or contamination levels. In addition, the Department has experienced a loss of personnel with process knowledge thereby limiting the amount of reliable knowledge available to make decisions.

Many materials are stored in forms and facilities not designed for long-term storage. For example, a majority of spent nuclear fuel is stored in wet storage pools up to 30 years old, that were not designed for long-term storage. Some materials will require new or updated facilities or repackaging if they are left in inventory for long periods of time. For example, scrap metals and lead will both require new storage facilities in the near future if the inventories of these materials are not decreased.

CONCLUSION

The Department is moving ahead to reduce the amount of materials it has in inventory, and thus reduce unnecessary risk and costs. The Department is working to address the findings of the MIN Initiative by taking a number of actions, including the following:

Developing and implementing systems to promote exchange of materials within the Department;

Evaluating options for effective management structures;

Establishing a policy for programs to identify and disposition materials in inventory in a cost-effective and timely manner;

Clarifying Departmental policies which currently inhibit materials disposition due to misinterpretations; and

Developing methodology to identify the costs of managing materials in inventory.

FINDINGS OF THE MATERIALS IN INVENTORY INITIATIVE

Finding 1: The Departments inventories of materials exceed current mission needs.

Finding 2: Inventory information and materials management systems vary greatly across material types and sites.

Finding 3: The Department does not have clear and effective organizational structures for inventory management and disposition for some materials in inventory.

Finding 4: The Department lacks policies to require or encourage its programs to identify and disposition materials that are no longer needed.

Finding 5: Policy and legal barriers hinder disposition of materials in inventory.

Finding 6: The Department does not understand sufficiently the costs of managing most materials in inventory, resolving vulnerabilities, and disposal as compared with other management options.

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ACKNOWLEDGMENTS

This paper draws on the year-long effort of many people and organizations. Special acknowledgment is due to Suzanne Rudzinski, Steven Taub, and Jeneva Craig at the Department of Energy; Project Performance Corporation; MACTEC; and all of the people involved in the Materials in Inventory Initiative from the program offices and field offices.

UNDERSTANDING AND REDUCING SUPPORT COSTS IN THE DEPARTMENT OF ENERGY OFFICE OF ENVIRONMENTAL MANAGEMENT

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ABSTRACT

Support costs make up a significant portion of total costs in the Department of Energy (DOE) Office of Environmental Management (EM), according to preliminary data from the 1996 Baseline Environmental Management Report (BEMR), the 1995 BEMR, and other studies of EM support costs. (DOE uses the term "support costs" to represent costs that would be considered "overhead costs" by private industry.) For that reason, reducing support costs is vital to reducing the overall cost of EM activities.

This paper provides general findings on the nature and significance of support costs from the 1996 BEMR and other relevant studies. This paper

also discusses how EM can reduce support costs by properly identifying and categorizing those costs and by understanding their drivers.

INTRODUCTION

Both government and private industry are under increasing pressure to cut costs. The drive to reduce the Federal deficit and agency budgets has spurred DOE and other agencies to seek more effective ways of increasing productivity and cutting costs. Private industry has focused recently on maximizing profits through increases in efficiency and reductions in costs. Overhead is often the first stop on the road to cost reduction because overhead costs tend to make up such a large percentage of total cost (5). (DOE uses the term "support costs" to represent costs that would be considered "overhead costs" by private industry.) Managers in both DOE and private industry are paying far greater attention to such costs than in the past and are developing cost systems that more accurately represent them.

This paper discusses overhead and support costs in general, presents findings from the 1996 Baseline Environmental Management Report (BEMR)a and other recent studies on overhead costs, and suggests more effective ways of both understanding and reducing those costs. The paper begins by defining overhead and support costs in private industry and in EM and discusses the traditional view of overhead and support costs and the problems with that view. The paper then looks at ways of improving our understanding those costs, examines support cost drivers in the EM program, and draws lessons from private sector best practices for reducing the costs of EM program management. Finally, the paper summarizes the results of recent studies of overhead and support costs.

OVERHEAD COSTS IN PRIVATE INDUSTRY

In the private sector, overhead costs are defined as costs not directly related to production or the provision of a service. For example, in manufacturing overhead costs include the direct and allocated costs other than labor and the purchase of materials. Areas generally considered to be overhead in manufacturing include indirect labor, general and administrative expenses, facilities and equipment costs, engineering costs, and materials overhead costs (5). In the hospital industry, to take another example, overhead costs include dietary staff; cafeteria; laundry and linen; plant; housekeeping; accounting; communications; patient accounts; data processing; admitting; hospital administration; public relations; personnel; auxiliary groups; chaplain services; and medical records (Noreen and Soderstrom 1994). These activities are vital to the functioning of the hospital but are not directly associated with the provision of medical services.

Over the last two decades in manufacturing, overhead costs have increased as a percentage of total costs, and "are replacing the direct costs of touch-laborers and purchased materials" in industry as the largest component of total cost (2). This shift is due primarily to the increased use of automation, which decreases the need for direct labor and increases the amount of overhead associated with maintaining and running equipment (5). The concepts of just in time, total quality management, group technology, and flexible manufacturing systems have also reduced the need for direct labor (4). Currently, overhead costs represent between 35 and 40 percent of total manufacturing costs (3,5). Overhead accounts for 70-75 percent of value added in the American electronics and machinery industry (5). The same article indicates that overhead rates of 1000 percent occur in some firms.

EM SUPPORT COSTS

EM defines support costs as direct and indirect costs incurred for activities other than cleanup. Cleanup costs include treatment, storage, disposal, characterization, retrieval, assessment, surveillance and maintenance, remedial action, decommissioning, deactivation, and stabilization.For BEMR, EM categorized support costs into 6 general areas and 29 more specific areas. This categorization was based upon an analysis of current EM budget documents: Activity Data Sheets; Allocable Cost Report; and site budget documents:

1) Management:
 Executive direction
 Management and operating (M&O) contractor oversight
 Management/award fee
 Program management
 Quality assurance

2) Finance and Administrative Services:
 Administrative support
 Chief Financial Officer
 Human resources
 Information services
 Legal
 Training

3) Environment, safety, and health:
 Environment, safety, and health
 Monitoring
 Laboratory support
 Pollution prevention

4) Infrastructure:
 Facilities management, engineering and maintenance
 Logistics support
 Procurement
 Utilities

5) Safeguards and security:
 Safeguards and security

6) Stakeholder, regulatory, and other:
 Agreements Economic development
 Lab-directed research and development
 Media/communications
 Regulatory compliance
 Stakeholder-related outreach
 Taxes
 Technology development
 Other

Recent studies show that support costs make up between 30 and 77 percent of total EM costs:

I. The 1995 DOE Support Cost Review (1995) found that support costs make up approximately 45 percent of total operating costs.

II. The 1995 BEMR found that support costs are approximately 50-65 percent of total EM costs.

III. Data collected for recent EM "workouts" held between EM Headquarters and sites indicate that support costs make up from 30-70 percent of site costs.

IV. Preliminary data from the 1996 BEMR are consistent with findings from these other studies of the EM support costs (Table I).

Although the exact definition of support activities varied somewhat from estimate to estimate, it is clear that support costs are a large component of total cost in EM.

Table I

PROBLEMS WITH THE TRADITIONAL VIEW OF OVERHEAD AND SUPPORT COSTS

Traditional cost accounting methods, used both in EM and in private industry, base estimates of overhead costs upon volume-related measures such as direct labor hours or machine hours. Overhead costs are assumed to vary in strict proportion to direct labor or machine hours. In other words, overhead costs are assumed to increase by 100 percent if direct labor hours are increased by 100 percent. Traditionally, accountants have even viewed direct labor hours as the cause or driver of overhead costs:

[Traditional] efforts base overhead burden rates [costs] on direct labor, materials, or machine hours. The problem with this approach is that the driving force behind most overhead costs is not unit output or direct labor. Overhead costs do usually correlate with unit output, but that does not mean that unit outputs "cause" overhead costs. In fact, acting as though they were causally related leads managers to concentrate on output measures or on direct labor rather than on the structural activities that determine overhead costs. (5)

Recent analysis has shown that there are two major problems with the above assumptions for estimating and understanding overhead costs:

There is only a weak relationship between overhead costs and direct activity. Noreen and Soderstrom (1994) found that a 100 percent increase in direct activity, such as labor or machine hours, would only cause a 25 percent increase in overhead costs. Data from the 1995 and 1996 BEMR support that analysis. 1995 BEMR data indicated that a 100 percent increase in treatment, storage, and disposal costs would cause less than a 15 percent increase in support costs. Preliminary 1996 BEMR data shows that a 100 percent increase in treatment, storage, and disposal costs will cause only a 20 percent increase in environment, safety, and health (ES&H) costs. Thus, estimates of overhead costs cannot accurately be based on direct activity levels.

More importantly for this paper, even if there was a relationship between overhead costs and direct activity level, direct activities do not cause overhead costs. For that reason, the common assumption that the way to reduce overhead costs is to reduce direct costs is false. Understanding the real cause or causes of the need for an overhead activity is the first step towards reducing its costs. The next several sections discuss ways of more accurately identifying those causes and more effectively reducing overhead costs.

IMPROVING OUR UNDERSTANDING OF OVERHEAD AND SUPPORT COSTS

The principles of activity-based costing (ABC) and Total Cost Management (TCM) shed some light on overhead costs. ABC, an extension of traditional cost accounting, uses multiple cost drivers to assign costs to products, recognizing that labor-based cost systems often do not capture differences in resource consumption by products. The basic premise of ABC is:

Demand for a product leads to activities of production.

Activities of production consume resources.

The amount of resources consumed is determined by the levels of the drivers.

TCM, as described by Ostrenga (8), is "a business philosophy of managing all company resources and the activities that consume those resources. Managing costs in a TCM environment means focusing on activities and the events, circumstances, or conditions that cause or 'drive' these cost-consuming activities."

Thus, ABC and TCM advocate, first, understanding the cause or driver of all costs and then reducing costs by working to change these drivers. For BEMR, EM has collected information on the drivers of support activities. Other recent EM studies and reviews have gone a step further, assessing ways to reduce support costs based upon a growing understanding of cost drivers. The remainder of the paper will provide results from these recent studies and suggest ways to reduce support costs.

EVALUATING EM SUPPORT COST DRIVERS

Several EM studies, the Support Cost Review and the Program Management Analytical Team, have recently examined the causal factors driving support costs. The Support Cost Review indicates that a large portion of support cost are driven by two factors:

DOE orders and requirements; and

The current guidelines laid out for contractor management.

From a detailed analysis of these drivers, the Support Cost Review concluded that significant costs savings could be achieved through the consolidation of DOE and contractor management practices and modification of low value DOE orders. In the former case, the review found that the separation of contractor and DOE planning and budgeting activities complicated the ES&H management process. Removing that separation would result in significant cost savings as ES&H costs comprise 11 percent to 38 percent of total cost (Table I).

The Support Cost Review attempted to evaluate the relative costs and benefits of current DOE ES&H orders and identified three examples of the cost savings achievable by reexamining and modifying current DOE requirements.

National Environmental Policy Act (NEPA): A categorical exclusion is effectively an exemption from the NEPA process granted to certain activities that, by their very nature, have no impact on the environment. Current orders require that categorical exclusion determination for NEPA be conducted at headquarters. The Support Cost Review suggests that this function be delegated to the Operations Office level and conducted by M&O contractors. Such a change would significantly reduce the number of reports required to confirm that an action is not subject to NEPA compliance.

Radiological Control Manual (RCM): Although acknowledged to provide excellent guidance for the safe handling and control of radiological material, RCM guidelines place a burden that brings little benefit on sites with low risk of radiological exposure. Support costs could be saved by revising these guidelines to exempt low risk sites.

Safety Analysis Reports (SAR): Finally, the Support Cost Review found that SAR preparation often places an undue burden on sites where reports were irrelevant, and that a graded approach acknowledging differences in risk levels would be more appropriate.

Fundamentally, the review found that the basic intention of these DOE orders, achieving an acceptable level of worker safety, is not being achieved efficiently. Modifying these orders would allow sites to achieve

these goals more cost effective. As discussed earlier, BEMR found that ES&H costs do not change much as treatment, storage, and disposal costs change. This may be due to the fixed nature of the compliance burden. If so, the removal of some DOE orders should have a significant impact on ES&H costs.

DETERMINING THE COST EFFECTIVENESS OF SUPPORT COSTS

Overhead costs in the private sector are often reviewed by potential customers, suppliers, or partners as a check on the real cost of producing a product or service (4). This interaction leads firms to adopt industry best practices. In the absence of such interactions, the Office of Environmental Restoration Program Management Analytical Team (1995) conducted a benchmarking study comparing program management of EM activities to program management for similar projects in the private sector. The intent of the study was improve the process for conducting program management, an essential support function, by identifying and adopting private sector best practices.

The Team found that private sector program management costs were approximately 40-80 percent lower than those in EM program management which range from 6 percent to 21 percent of total costs (Table I). The reason for this difference, the team concluded, was that program management in the private sector focused on work directly related to the speedy completion of projects and that EM program management did not. Thus, the study shows both that EM program management can be improved by greater focus on those activities that speed the completion of projects, and that benchmarking can lead both to a better understanding of cost drivers and to lower cost.

CONCLUSION

In summary, recent studies on overhead and support costs in EM, other federal agencies, and private firms show four things:

I. Overhead costs are the single largest component of total cost in nearly all industries and government agencies, accounting for more than 35 percent of total cost. Thus, understanding and controlling these costs is key to controlling overall costs.

II. Historically, little attention has been focused on overhead costs. It is only recently that the "hidden factory" of overhead costs has been scrutinized.

III. ABC and TCM indicate that understanding and managing cost drivers is a key to controlling costs. It is clear that there are many overhead cost drivers other than direct activity level.

IV. EM has begun to perform several analyses that allow a better understanding of overhead costs. We believe that similar analyses should be continued and that management based upon these findings could lead to future cost reductions.

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LIFE CYCLE COST AND RISK ESTIMATION OF ENVIRONMENTAL MANAGEMENT OPTIONS

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ABSTRACT

The Lockheed Idaho Technologies Company (LITCO) Economic and Systems Analysis program has developed a life cycle cost and risk estimation process to evaluate Environmental Management options for the United States Department of Energy. The evaluation process is demonstrated in this paper through the comparative analysis of two alternative scenarios that have been identified for the management of the alpha-contaminated mixed low-level waste that is currently stored at the Idaho National Engineering Laboratory (INEL). The two scenarios that are evaluated are a Base Case and a Delay Case. The two scenarios are realistic and based on actual data, but are not intended to exactly match actual plans currently being developed at the INEL. The paper includes a general description of each scenario, along with major assumptions that were made to support the analytical process.

Life cycle cost estimates were developed for both scenarios with the use of the System Cost Model. The resulting costs are presented and compared. Life cycle costs are shown as a function of time and also aggregated by pretreatment, treatment, storage, and disposal activities. Although there are some short-term cost savings associated with the Delay Case, the cumulative life cycle costs are shown to eventually become much higher than the costs for the Base Case over the same period of time, due mainly to the storage and repackaging necessary to accommodate the longer Delay Case schedule. Similarly, life cycle risk estimates were prepared using a relatively new risk analysis methodology that was developed for the INEL Environmental Management Integration task and has been adapted to the System Cost Model architecture for automated, systematic cost/risk applications. Relative risk summaries are presented for both scenarios as a function of time and also aggregated by pretreatment, treatment, storage, and disposal activities. The relative risk associated with the Delay Case is shown to be higher than that of the Base Case. Finally, risk and cost results are combined to show how the collective information can be used to help identify opportunities for risk or cost reduction, and highlight areas where risk reduction can be achieved most economically.

INTRODUCTION

A life cycle cost and risk estimation process has been developed to evaluate various Environmental Management options for the United States Department of Energy (DOE). The Economic and Systems Analysis program, based at the Idaho National Engineering Laboratory, has been sponsored by

the Office of Waste Management, Office of Planning and Analysis (EM-35) and the Office of Science and Technology, Office of Technology Systems (EM-53).

Lockheed Idaho Technologies Company's (LITCO's) Economic and Systems Analysis (ESA) program has developed a systems engineering process for analysis of waste management problems. The process is based on the engineering analysis that LITCO has provided on the Waste Management Programmatic Environmental Impact Statement, Site Treatment Plans (STPs) as required under the Federal Facility Compliance Act, and support for the Baseline Environmental Management Report (BEMR). A risk methodology was also developed by the Idaho National Engineering Laboratory Environmental Management Integration Program. The combination of cost analysis and risk analysis capabilities has allowed the ESA program to address EM alternatives from new perspectives.

LITCO's Economic and Systems Analysis program has developed an extensive knowledge in waste management facility cost development, engineering model design, and risk applications. This knowledge, which was initially developed to support EM-30, has now been additionally focused on EM-50 initiatives to identify cost-effective and reduced-risk alternatives. The ESA studies will be used by EM to gain a greater understanding of the opportunities for cost reductions and provide a quantitative means for comparison of DOE policy options.

The benefits to EM-50 have included:

Providing a baseline for comparison of Technology Development alternatives,

Providing a method to communicate results on new technologies to EM-30,

Providing a bridge between an "Average Site's" approach and site-specific applicability,

Helping to identify needs for technology development,

Providing a basis for prioritizing opportunities for risk or cost reduction.

The benefits to EM-30 have included:

Better focused Technology Development supporting key EM-30 policy options,

Better understanding of the costs and benefits of advanced technology options,

Better integration of a strategic planning basis,

Consistent methodology enabling comparative analysis of waste management alternatives,

Better understanding of cost implications for various complex-wide configuration options.

The need for cost/risk integration has been identified by the DOE. The DOE considers risk and life cycle costs in establishing program priorities. The ESA program has developed a tool called the System Cost Model (SCM) which has facilitated the cost/risk analysis of complex EM alternatives. The SCM allows analysis of various technology processing options for mixed low-level waste (MLLW), low-level waste (LLW), and transuranic waste (TRUW)--both mixed and non-mixed. The use of the SCM has helped LITCO integrate the requirements of EM-30 and EM-50. The two models which have been developed to support cost/risk analysis are described as follows:

System Cost Model (SCM) - The SCM was initially developed for EM-35 to support sensitivity analysis of waste management costs for the BEMR. The SCM produces complex-wide life cycle costs for treatment, storage,

disposal, and transportation of MLLW, LLW, and TRUW. The SCM also includes a database of site-specific waste management information including: waste inventory volumes and generation rates, treatment processing schemes, existing and planned facilities, site-specific cost factors and labor rates, and schedules.

System Cost Model - Risk (SCM-R) - The SCM-R has been developed to a conceptual level for EM-35 to support cost/risk evaluations. The SCM-R is an add-on to the base SCM. The fundamental risk methodology is based on the work done by the Environmental Management Integration Program (EMIP) at the INEL. Approximately 1000 simplified risk assessments were produced for the EMIP. The SCM-R will produce simplified relative risk assessments to show baseline life cycle risk, worker and public risk, waste disposal risk, and waste transportation risk. All categories of waste, from spent fuel to low-level waste, can be accommodated, as can all types of waste (radioactive, hazardous, and mixed). The method is based on the fundamental equations of risk (e.g., as used in CERCLA risk assessments). The risk calculations are based on the product of probability and consequences. The equations are broken down into risk elements, e.g., inventory quantities, toxicities, confinement barriers. Look-up tables provide values to be used for each risk element.

COST/RISK STUDY

This paper includes a specific cost/risk study demonstrating the use of the SCM and SCM-R tools. The study provides a life cycle cost/risk evaluation of the trade-offs of using long-term storage prior to treatment versus treating with existing technologies and minimizing storage. The study is based on actual INEL waste stream data and can be considered representative of the type of analysis that could be performed at any large DOE site. However, since some of the assumptions used are hypothetical, this study is not intended to accurately reflect current INEL plans. Rather, the study is meant to demonstrate a unique cost/risk analysis capability using realistic input parameters.

The purpose of the study was to compare the magnitude of the costs and risks for long-term storage versus the current planning basis. These options show the costs and risks associated with delaying treatment until new technologies are available. The study also shows how both costs and risks can be evaluated in one analysis. The remainder of this paper includes a description of the alternatives, assumptions, cost and risk results, and key study conclusions.

Alternative Descriptions

Two alternatives were defined for this cost/risk study:

1) Base Case Scenario - The Base Case Scenario is comprised of the INEL BEMR treatment, storage, and disposal configuration and the 1995 Mixed Waste Inventory Report (MWIR) waste stream data. The Base Case scenario used a treatment schedule based on the STP.

The waste is retrieved from earthen-covered storage. Retrieval will be followed by receipt and inspection of the waste at the Stored Waste Examination Pilot Plant (SWEPP), which includes an open, dump, and sort module to determine which treatment the waste form will receive. Also included are pre-treatment handling and storage. All necessary pre-treatment facilities are assumed to be in existence at the INEL; therefore, no construction costs are included until the year 2025, when existing facilities are assumed to become obsolete and new storage is constructed.

This case utilizes incineration followed by grouting for the particulates, sludges, and some of the debris. However, most of the debris will be treated by first shredding the waste, which is then treated by thermal desorption followed by grouting. Other solids and particulates will be treated by a polymer stabilization. Disposal is assumed to take place in an onsite engineered disposal facility.

2) Delay Case Scenario - The Base Case Scenario was revised to show the effects of long-term pre-treatment waste storage.

Treatment and disposal for this case are identical to the Base Case with the only change being that the treatment and subsequent disposal occur fifty years later. The treatment and disposal facilities required for the Delay Case are considered to be nonexistent and will require construction. However, no post-treatment storage is required for this case since the treated waste goes directly to disposal.

In addition to pre-treatment storage, the waste will be overpacked as it is received and inspected based on the assumption that current containers are not adequate to support another 50 years of storage. SCM does not have an "overpack" module, however, in order to provide costs to adequately reflect the overpacking, the receiving and certification module costs were artificially increased to account for the activity. After the waste is retrieved (complete in the year 2015) and overpacked (complete in the year 2039), it is stored until treatment is available in the year 2047.

General Assumptions:

The specific waste management system defined for this study was the INEL alpha MLLW. For the sake of simplicity in modeling the INEL alpha MLLW, the initial risk analysis was limited to five waste matrix categoriesa. The total alpha MLLW inventory at the INEL is characterized by 12 waste matrix codes. However, the five waste codes chosen for this demonstration model represent 97.7% of the volume of waste to be treated at the INEL. The five waste codes chosen for this model are: S 3110, S 3120, S 5110, S 5300, and S 5400.

The costs reflect DOE-built and operated facilities required for the alpha MLLW inventory. Since treatment facilities for this waste do not currently exist at the INEL, new facilities will be required. The alpha MLLW will be treated based on the requirements of the Resource Conservation and Recovery Act. The treatment window for the inventory is assumed to be 19 years in duration (same for both cases).

These cases assume that the alpha MLLW inventory is disposed of onsite in an above-ground engineered disposal facility.

Scheduling Assumptions for the Base Case

For the Base case, all construction of treatment facilities starts in 1996, with treatment to commence in 1999. The SCM was allowed to build all treatment required.

Scheduling Assumptions for the Delay Case

Retrieval of the waste will occur over 19 years. After the waste is retrieved, it is received, inspected and assayed at the SWEPP facility over a 19-year duration that ends in 2017. As the waste is assayed it is overpacked and then placed in storage. Overpacking operations will be complete in 2039. Existing storage capacity for the retrieved and overpacked waste is not sufficient, so SCM will be allowed to construct storage facilities as required.

Waste storage continues until 2065. SCM will construct new facilities and decontaminate and decommission (D&D) the old facilities as required during the extended storage period.

Construction (for three years) is followed by treatment, which commences in 2047 and completes in 2065.

COST RESULTS

Life cycle cost estimates were calculated for the two INEL alpha MLLW cases using the SCM and based on the assumptions and case descriptions outlined above. The total life cycle costb for the Base Case was estimated to be $1.25 billion (B), while the total life cycle cost for the Delay Case was estimated to be $2.79 B.

Figure 1 shows a comparison of the cumulative costs for the two cases over time. The Base Case includes higher up-front costs because all needed treatment, storage, and disposal facilities are constructed immediately. However, the cumulative costs for the Base Case level out after the last facility is decommissioned in the year 2024. Because of the much longer operational time frame, the cumulative costs for the Delay Case surpass those for the Base Case in the year 2032 and are ultimately over twice as high.

Fig. 1

Figure 2 shows a comparison of the costs for the two cases broken down by waste management function. The four waste management functions included in the SCM estimates are pre-treatment (including pre-treatment storage), treatment, post-treatment storage, and disposal. Most of the cost difference between the two cases shows up in the pre-treatment category. This is due to the increased cost of pre-treatment storage required for the Delay Case. The higher costs associated with the Delay Case are due to the construction and operations of the necessary storage facilities. The Delay Case costs also include overpacking activities and D&D of two existing storage facilities, neither of which are necessary for the Base Case.

Fig. 2

RISK RESULTS

The relative risk was calculated for both the Base Case and the Delay Case. The annual relative risk for each case is presented in Fig. 3. These risk profiles should be considered preliminary because the basis for the annualized risk calculations is still under development. The output depicts the annual relative risk change as the alternative is implemented. The output represents the total risk of all steps involved in managing the waste, from initial storage through retrieval, handling, treatment, and disposal. The beneficial effects of treatment on the annual risk are evident by the decreasing risk. These effects are primarily attributed to placing the waste in a less-mobile physical form.

Fig. 3

The relative risk was further defined by summary-level treatment, storage, and disposal risk statesc. The illustration in Fig. 4 indicates that the process steps contributing the most to the total time-integrated risk are pre-treatment storage and long-term disposal (>500 years).

Fig. 4

A refinement of the output shown in the previous figure is the breakdown of the relative risk by state (equivalent to module). This output discriminates which specific processes are contributing the most risk for the alternative. For the Base Case alternative, the relative risk is greatest in the front-end module for storage. In treatment, the majority

of the risk is dominated by the open, dump, and sort and the incineration module. The disposal risk is driven by the failure of the engineered barriers in the period beyond 2499 in the module.

COST/RISK ANALYSIS

The individual cost and risk results provide insights to the highest cost and risk modules. This information is valuable to determine general areas of emphasis. In addition to these results, the costs and risks were integrated to provide new insights into cost-effective risk reduction opportunities. Two techniques were developed to analyze the cost/risk results:

1) Normalized Product technique - which can be used to guide technology development prioritization and risk mitigation activities,

2) Marginal Alternative Comparison technique - which discriminates risk/cost performance between alternatives to support further system optimization.

These innovative techniques were developed for the purpose of leveraging the cost and risk data to gain insights into areas of greatest cost savings and risk reduction potential. A description of the methodology for the techniques is included below. The costs and risks from the Base Case and the Delay Case scenarios were also analyzed using these techniques.

1) Normalized Product (NP) technique - The NP technique was used to evaluate the system components to determine the greatest opportunities for combined risk and cost reduction. The NP places cost and risk on a normalized scale so that a unit of risk is related to a unit of costd. The higher values on the NP scale are indicative of the modules/states with the largest relative risks and costs as compared to the processes with the smallest products of cost and risk. The technique helps prioritize where funding should be directed to affect the largest potential risk reduction and cost savings. This technique was applied to the Base Case and Delay Case scenarios and the results are summarized in Fig. 5.

Fig. 5

Results using the Normalized Product technique: The results indicate that the greatest opportunities for risk and cost reduction are in the Delay Case. The marginal changes for cost and risk were highest for pre-treatment. Proportionately, the marginal cost decrement was almost five times that of the risk. This would indicate that the Delay Case has significantly higher costs and some associated increase in relative risk. The results also show an opportunity for improvement in both cases for disposal.

Risk mitigation could include (but is not limited to) reducing the number of years of pre-treatment storage, producing a less-mobile waste form, destroying organics, reducing worker exposure in characterization and packaging processes, and improved final waste form.

2) Marginal Alternative Comparison (MAC) technique - This technique is used to discriminate differences in risk and cost between alternatives. The MAC technique helps an analyst understand trade-offs and sensitivities between risk and costs6 from different waste management options (technologies, scheduling, etc.). This technique can be used to help answer questions like:

- Does risk decrease proportionately with increased costs?
- How much does risk increase if costs are cut?

The technique provides a comparison of the normalized cost and risk data from the alternatives. The fractional change in normalized risk (of one alternative compared to another) is compared to the fractional change of normalized cost (between alternatives). The resulting marginal risk per unit of cost is a measure of the potential effectiveness of an improved alternative in reducing risks and costs. This technique can be used to analyze technology and operational effectiveness for pre-treatment operations, treatment, storage and disposal. A comparison of the Delay Case to the Base Case using the MAC technique is presented in Fig. 6.
Fig. 6

Results using the MAC technique: The results indicate that long term pre-treatment storage (including additional characterization and overpacking) causes significant cost degradation and additional risk. Post-treatment storage costs are marginally improved if treatment is delayed due to improved throughput between treatment and disposal operations. Treatment and disposal costs and risks show no significant marginal differences.

System Level impacts can also be assessed using these techniques. For example, if an improved new technology can be developed in 10 years (resulting in increased pre-treatment storage) how much better would the technology need to perform to break even with the additional costs and risk from the added storage? The system assessment could also evaluate the marginal differences of using a new treatment technology which produces an improved final waste form for disposal.

CONCLUSIONS

The cost and risks of two alternatives were analyzed using a systems-based life cycle cost and risk estimation process. Techniques were applied to integrate the results from the individual cost and risk studies. The techniques helped to provide insights into areas to maximize effectiveness while reducing risk. The techniques can be used to support many initiatives for EM:

- Define incentives for investment in R&D
- Identify and prioritize Technology Development projects
- Maximize operational effectiveness (sizing of facilities, years of operation)
- Optimize schedule
- Optimize facility siting and configuration

The ESA studies will be used by EM to gain a greater understanding of the opportunities for cost reductions and to provide a quantitative means for comparison of DOE policy options.

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SLASHING COSTS AND CHOPPING SCHEDULES: RESTRUCTURING THE CERCLA PROCESS FOR A DOE SUPERFUND SITE*

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ABSTRACT
Regulatory document preparation at hazardous waste sites has proved to be costly when complying with the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA). At Lawrence Livermore National Laboratory (LLNL) in Livermore, California, we are devising a methodology that will save millions of dollars and shorten the time to initiation of cleanup while meeting the intent of CERCLA to remediate contaminated sites. In 1994, we completed a Site-Wide Remedial Investigation report that describes the characterization and risk assessment of LLNL's 7000-acre Site 300 experimental test site. The next step in the CERCLA process would have required us to prepare complete Feasibility Studies, Proposed Plans, Records of Decision (RODs), and Remedial Action Implementation Plans for three Operable Units (OUs) containing contaminated ground water. Such an effort would require several years and millions of taxpayer's dollars to accomplish. In close consultation with the State and Federal regulatory agencies, we are creating a model strategy to 1) conduct Removal Actions (landfill capping, buried drum removal, surface drainage control, etc.) in the OUs to remove the threat of future releases to the environment, 2) prepare ground water monitoring plans to monitor the contaminant plumes, 3) prepare contingency plans to address the contamination should it migrate and pose an unacceptable risk, 4) move the ground water monitoring tasks to a pre-existing Site-Wide OU that addresses the site as a whole, and 5) incorporate the Removal Actions into the Site-Wide ROD as completed remedies for the OUs. This process will substitute relatively short documents describing Removal Actions, monitoring, and contingencies for the much longer studies and plans currently required. With this approach, potential sources of future releases will be remediated, low risk ground water contamination will be monitored, human health and the environment will remain protected, and regulatory requirements for RODs will be met. Assuming that our negotiations continue successfully, we will save the taxpayers millions of dollars, expedite the cleanup of the site, and bring the entire process to closure.
INTRODUCTION
Regulatory document preparation for hazardous waste sites has proved to be costly and time consuming when complying with the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) of 1980. At Lawrence Livermore National Laboratory (LLNL) in Livermore, California, we are developing a methodology that we expect will save millions of dollars and shorten the time to initiation of cleanup while meeting the intent of CERCLA to remediate contaminated sites and protect human health and the environment.
Description of LLNL Site 300
LLNL is a nuclear weapons research and development laboratory owned by the U.S. Department of Energy (DOE) and operated for DOE by the University of California. LLNL's Site 300 facility, in existence since 1955, supports DOE's defense mission in research, development, and testing associated with non-nuclear components of nuclear weapons. Site 300 is a 7000-acre Experimental Test Site located about 60 miles east of San Francisco, California, in the sparsely populated Altamont Hills of the Diablo Range (Fig. 1). This Superfund site encompasses approximately

11 square miles of former ranch land; surrounding land is used primarily for sheep and cattle grazing, wind-powered generation of electricity, and off-road vehicle recreation. The closest population center is the San Joaquin County town of Tracy, 8.5 miles northeast. Most of the site is undeveloped, with the developed portion consisting of machine and craft shops, high explosives (HE) firing tables and bunkers, buildings that accommodate HE formulation and machining as well as static and dynamic weapons-component testing, administration buildings, landfill disposal pits, roads, and other associated infrastructure. The site is characterized by rugged terrain consisting of rolling hills and steep northwest-trending valleys.

Fig. 1

Pre-CERCLA Environmental Restoration at LLNL Site 300

Since 1981, LLNL has conducted investigation and restoration of chemical and radioactive contamination at Site 300. The investigations have focused primarily on delineating ground water contamination, but have also addressed contamination of soil and rock, surface water, biota, and air. Prior to August 1990, investigations and remediation were conducted voluntarily under the oversight of the California Central Valley Regional Water Quality Control Board (RWQCB), now a part of the California Environmental Protection Agency. In August 1990, as a result of the ground water contamination, Site 300 was named to the CERCLA National Priorities List, also known as Superfund. In 1992, a Federal Facility Agreement (FFA) was negotiated between DOE, the U.S. Environmental Protection Agency (EPA), the California Department of Toxic Substances Control (DTSC), and the California RWQCB. The FFA defines the scope of work required, coordinates efforts among the FFA parties, standardizes requirements, establishes due dates for deliverables, and ensures compliance. Since late 1990, all environmental investigation and remediation activities have been conducted under the oversight of the three environmental regulatory agencies in accordance with CERCLA and State laws and regulations.

Work Conducted Under CERCLA

An important element of the FFA was the establishment of six Operable Units (OU) for Site 300. The OUs were created based on considerations of geography, hydrogeology, types of contaminants present, and nature of the environmental release. Following CERCLA guidance, the regulatory agencies, DOE, and LLNL jointly established for each OU an enforceable schedule for submittal of CERCLA documents from Feasibility Studies (FS) through Records of Decision (ROD). Figure 2 shows the initial schedule incorporated into the FFA in 1992.

LLNL had conducted initial investigations of the site between 1981 and 1992 that identified ground water contaminated with volatile organic compounds (primarily trichloroethylene [TCE]), tritium, and the HE compounds RDX (cyclotrimethylene-trinitramine) and HMX (cyclotetramethylene-tetranitramine). At the request of the regulatory agencies, we compiled this information into the large, comprehensive Site-Wide Remedial Investigation (SWRI) report, which was completed in 1994. This report also presents the results of a base-line public health assessment. The succeeding FS/Proposed Plan (PP)/ROD documents for each OU were to be based on the remedial investigation results reported in the SWRI.

Fig. 2

As work progressed under the FFA schedule of deliverables, it became apparent that a major amount of DOE/LLNL's restoration program personnel, financial, and time resources were being consumed in preparing lengthy and sometimes repetitive and unnecessary documents. For each OU, we proceeded or were prepared to pass through a cycle of 1) preparing and submitting draft documents to the regulatory agencies, 2) responding to regulatory agencies' comments on the document and incorporating their comments into a draft final document, 3) submitting the draft final document to the agencies, 4) responding to the agencies' comments on the draft final document and incorporating their comments into a final document, and finally, 5) submitting the final document to the agencies for final acceptance. This process was followed to completion for the FS, PP, and ROD documents for one OU, and to date has been completed part way through the cycle for two other OUs. To complete this process, we recently were spending annual budgets of $6-9 million, using the skills and talents of approximately 30 full-time scientists, engineers, technicians, community relations specialists, and clerical and administrative staff members.

Progress toward formal implementation of cleanup of the site has been slow because of the lengthy time required to complete the document preparation requirements before cleanup can begin. However, since becoming a CERCLA site, in two cases we implemented Removal Actions to begin remediation of VOCs in ground water even before acceptance of the ROD for the OU.

CERCLA Restructuring

Early in FY95, LLNL and the DOE Oakland Operations Office recognized that our decreasing DOE budget for environmental restoration was not and would not be sufficient to complete the work obligations required by the FFA. In addition, we were experiencing severe frustration at the length of time required to complete what we considered to be unnecessary regulatory documents. Consequently, at our Remedial Project Managers (RPM) Meeting in April 1995, we proposed an alternative approach to the regulatory agencies. We recommended that the Site 300 team (regulatory agencies, DOE, and LLNL) should stand back and take a serious look at how we had organized the project of environmental restoration of Site 300. We proposed to identify possible ways to meet the goals and needs of all parties to the FFA while, at the same time, save the taxpayers and DOE millions of dollars by moving more rapidly to the clean up phase. We proposed restructuring CERCLA as it applies to our restoration activities.

During the course of three RPM Meetings held in April, May, and June 1995, the Site 300 team considered the following:

For the Pit 6 OU, the FS had already been completed, and the regulatory agencies had approved a natural remediation remedy for TCE in ground water. Additional capping of the landfill still needed to be done.

For the Building 850/Pits 3&5 OU, the regulatory agencies had already agreed that no remedial action was required with respect to ground water other than continued ground water monitoring because 1) no economically feasible method is available for removing tritium from ground water, and 2) the tritium poses no risk to the public. Potential future release sites, however, need to be addressed.

For the HE Process Area OU, the regulatory agencies were amenable to discussing options other than writing FS/PP/ROD documents to address low concentrations of TCE and HE compounds in ground water.

Because the regulatory agencies had already agreed that active ground water remediation was not needed or feasible at the Pit 6 OU and B850/Pits 3&5 OU, we concluded that there was no value in preparing three regulatory documents (FS/PP/ROD) for each OU just to identify and select a No Action remedy (other than continued ground water monitoring). Furthermore, the team concluded that any ground water remediation that may be required at the HE Process Area OU could be accomplished much faster and more economically through judicious use of Removal Actions rather than following the conventional and costly CERCLA FS/PP/ROD process.

Consequently, LLNL and the DOE Oakland Operations Office, in close consultation with the State and Federal regulatory agencies, have created a model strategy for CERCLA restructuring as it applies to three Site 300 OUs: Pit 6, Building 850/Pits 3&5, and HE Process Area. The restructuring provides for:

1) Continued protection of human health and the environment.

2) Conduct of Removal Actions where warranted in the OUs to remove the threat of future contaminant releases to the environment. These Removal Actions will consist of landfill capping at the Pit 6 OU; excavating buried drums in the HE Process Area OU; and landfill capping, constructing a surface water diversion ditch, or digging an interceptor drain at the Building 850/Pits 3&5 OU.

3) Continued ground water monitoring at the OUs to monitor the extent and movement of contaminant plumes. This monitoring will enable us to detect any unexpected changes in location, movement, or concentration of the plumes so that remedial action can be taken if warranted.

4) Preparation of contingency plans to establish criteria and procedures for remedial action in the event the ground water contaminants migrate and pose an unacceptable risk to the public or the environment.

5) Moving the ground water monitoring tasks for each OU to a Site 300 Monitoring OU that encompasses all parts of Site 300 where monitoring only is required.

6) Dissolving the three OUs as discrete entities, thereby removing the need for preparation and submittal of conventional CERCLA documents for these OUs.

7) Preparing a Site 300 Monitoring ROD that will, in part, incorporate the results of Removal Actions performed in item 2. The Removal Actions will demonstrate that remedial actions have been taken at the sites of potential releases in the three former OUs.

This restructuring substitutes relatively short Removal Action Engineering Evaluation/Cost Analysis documents, monitoring plans, and contingency plans for the much longer studies and plans (FS/PP/RODs) currently required by the CERCLA FFA. Two other OUs are well along the conventional CERCLA process, and will continue to follow that path to final remediation. One of these OUs has off-site ground water contamination; the other has high concentrations of TCE in ground water. We must complete our remedial investigation work at the sixth OU before determining its ultimate CERCLA pathway. Figure 3 illustrates our expectation of the final resolution as it applies to CERCLA restructuring of Site 300.

Fig. 3

CONCLUSION

With this approach to CERCLA restructuring of the LLNL Site 300 environmental restoration project, human health and the environment will

remain protected, potential sources of future releases to the environment will be remediated, low risk areas of ground water contamination will continue to be monitored, and regulatory requirements for completion of RODs will be met. Furthermore, this new process will hasten the start of remediation in some cases by 1-1/2 years and reduce costs by millions of dollars. We believe this approach of working with the regulatory agencies to identify mechanisms to streamline the CERCLA process is well worth the effort and should be attempted at all sites where feasible.

Session 50 -- DECONTAMINATION & DECOMMISSIONING
Co-chairs: Gilles Chevier, NUMATEC;
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50-1
WHAT MAKES DECOMMISSIONING SO COMPLICATED?
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ABSTRACT
In Germany decommissioning has become a long lasting process connected with considerable costs. The reasons for this development are analyzed. Two possibilities are pointed out to save time in the decommissioning procedure; the package solution and the black box solution. Further on the psychological situation of decommissioning is discussed with its impact on the decommissioning process.
INTRODUCTION
While there are no plans in Germany to set up new nuclear power plants, an important nuclear activity is dealing with decommissioning. The decommissioning process itself is governed by the atomic energy laws and is therefore subject to authorization. These legal parameters demand a special licensing procedure.
Our organization, the TUEV-SW, an Association for Technical Supervisory, advises the authorities in nuclear affairs safety. Concerning decommissioning we are producing surveys, to give the authorities a decisive support for the approval.
THE LEGAL SITUATION OF DECOMMISSIONING
In the Federal Republic of Germany all nuclear matters are governed by the atomic energy laws, AtG (1) and by the subordinated radiation protection ordinance, StrlSchV (2). The decommissioning of a nuclear plant, which has been licensed by 7 AtG (1), is subject to authorization too, 7(3) AtG (1). This legal parameter needs a special licensing procedure:
  Decommissioning has to be planned, documents and a safety report have to be made.
  The authorities have to be asked for a decommissioning license putting on the documents and the safety report.
  The documents have to be examined, if they fulfil the requirement of 7 (2) AtG (1), with its main aim to protect life and environment against the danger of radioactivity, whereby the steps to protection must fulfil the state of arts; that means especially, that decommissioning work can be done without the risk of significant radioactivity releases into the environment.
  Decommissioning is licensed by the authorities.

As the decommissioning documents have beside the technical meaning also a legal one, mistakes or inconsistencies have to be corrected. The revision of documents can take a lot of time. If there is any deviation in the decommissioning process, the deviation is also subject to authorization, what may require an additional revision of documents. These facts show, that decommissioning needs a special strategy to avoid time wasting mistakes.

THE TECHNICAL SITUATION OF DECOMMISSIONING

In most cases nuclear plants are very complex systems, therefore decommissioning can not be done in only one step. In consequence of that decommissioning has to be done step by step either progressing from one cell or room to another or progressing from system to system. A nuclear plant, which was shut down, is contaminated in the most areas with a more or less amount of radioactivity. The remaining radioactivity is enclosed in the plant within the same barriers as in time of running. Decommissioning and dismantling means, to open these barriers and it must be the objective that within this opening process there are no significant releases of radioactivity. Appropriate safety measures have to be made for every point of intersection. Therefore methods and procedure of decommissioning and dismantling must be laid down in the documents and in the safety report.

Results from the Legal and Technical Situation for the Decommissioning Process

Consideration of the legal parameters and the technical situation of a nuclear plant, which is to be decommissioned, shows that a move for decommissioning needs, because of its nature, a lot of separate technical measures, which are to be described in the documents and which are to be examined in detail.

The analysis of the licensing procedure leads to the following time schedule:

1) Producing of documents and safety report
 Propose a move
 Putting on documents and the safety report
2) Examination of documents
 Producing of the survey
3) License
 Performance

The three time consuming steps depend from one another:

t (license) = f (Q examination, survey)
t (examination) = f (Q documents)
t (examination) = f (A documents)
(t = time; Q = Quality; A = Amount)

The better the quality of documents, the sooner the survey can be produced and the license be given. If the documents do not fulfil the quality requirements and the amount of documents is very high, the examination needs more time and the license gets more complicated, as it contains more regulations.

Looking for time saving measures in the licensing procedure, the first is, to cut the range of the move for decommissioning from one over all step into several steps. This method has the benefit, that all time consuming steps in the licensing procedure can be finished in realistic time intervals, revisions of documents included.

If two or more independent facts are connected in the move for decommissioning, which do not fulfil the requirements of 7 (2) AtG (1) at the same time, the license will only be given when the requirements of

all systems connected in the move are fulfilled. These considerations just lead to the package solution.

Fig. 1

The package solution: The whole range of decommissioning is divided in several packages. Within a package it should be avoided to connect independent facts or systems, which do not fulfil the requirements of 7 (2) AtG (1) at the same time.

Within the package solution there is an optimizing potential. A system or a part of a system (Sys) foreseen to be dismantled has various points of intersections (poi) to the next one (for instance pipes, medium, power, measuring and control systems etc). These points of intersections must be made safe, especially against uncontrolled release of activity. The system may be enclosed from a volume element (cell, room) from which the number of the points of intersections to the next one are less. There is a relation between the size of a volume element of a system (Vol E (Sys)) and the number of points of intersections. The number of points of intersection are direct proportional to the time, which is needed for the different steps in the decommissioning licensing procedure. On the other hand the size of a volume element of a system, which is to be dismantled, is important for the progress of the decommissioning process itself. There is only a little progress in the decommissioning process, if the volume element has been chosen too small. These considerations lead just to the black box solution. To get much progress in the decommissioning process the size of Vol E (Sys) should be at a maximum, while the points of intersections (poi) should be at a minimum.

Fig. 2

The black box solution: Points of intersections are minimized, while the size of a volume element of a system to be destructed is at a maximum.

The benefit of the black box solution is, that the actual time expenditure of producing documents is minimized, as there are less points of intersections and the safety requirements can be proved by worst case analysis. The examination of the documents can be done quickly by the same reasons. In addition the black box solution is high flexible in the performance of the decommissioning license. With both solutions, the package solution and the black box solution, it is possible to save time in decommissioning process.

THE PSYCHOLOGICAL SITUATION OF DECOMMISSIONING

People like to be creative, that means to built up things, while decommissioning and dismantling is basically non-creative. Therefore decommissioning is done reluctantly from the very first beginning. Beside this negative motivation the decommissioning process is connected with the lost of jobs. So everyone of the personnel of a plant, which is to be decommissioned, tries to save his job. This is additionally contraproductive for the decommissioning process and can lead to a further waste of time. But when a nuclear plant is to be decommissioned never the less the question has to be answered, who should do the decommissioning, the plant personnel, who had run the plant or an extern firm, which is doing decommissioning professionally? The plant personnel has a great specific knowledge about the plant and the incidents which have occurred. This knowledge is very valuable for the decommissioning process. But the above mentioned facts, especially the lost of jobs are standing in opposite to it. A great disadvantage of an extern firm, which is doing decommissioning professionally is, that it has no specific knowledge about the plant. On the other hand the extern firm is only

primarily interested in getting the order. If the order is booked, the decommissioning firm may not be unhappy when decommissioning is connected with delay, as time brings money.
Therefore the question, who should do the decommissioning, can not clearly be answered. In both groups, the plant personnel or an extern firm, there is a tendency to slow down the speed of the decommissioning process. Perhaps a solution with a mixed decommissioning personnel might be useful.

CONCLUSION

The decommissioning process is governed by the atomic energy laws and is therefore subject to authorization. These legal parameters demand a special licensing procedure. To fulfil the legal requirements and depending on the nature of decommissioning process a great number of separate technical measures has to be considered and laid down in documents. These legal as well as technical conditions must be taken in to account to avoid time wasting mistakes. A time saving strategy in the decommissioning procedure could be the package solution, whereby the whole range of decommissioning is divided into several packages. Within the package solution there is an optimizing potential, the black box solution, in which the number of points of intersection is minimized, while the volume of the system or room, which is to be destructed, is at an maximum. If a decommissioner takes this strategy - the package and the black box solution - into account before he starts, he can avoid time wasting mistakes and can reduce the costs.
The decommissioning process is basically non-creative and connected with the lost of jobs. Both facts yield a tendency to slow down the speed of the decommissioning process. This retard-effect plays a part in the composition of the decommissioning team. The plant personnel has the advantage of a great specific knowledge in contrary to an extern firm. This valuable plant knowledge should not be given up. Perhaps this retard-effect can be stopped by a solution with a mixed decommissioning team, consisting of plant personnel and personnel of an extern firm; this could be combined with tailor-made contracts containing stops, where the decommissioning success must be proved.

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50-2

REACTOR HALL DECONTAMINATION AT THE A-1 NUCLEAR POWER PLANT, BOHUNICE, SLOVAKIA

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ABSTRACT
The reactor hall at A-1 Nuclear Power Plant at Bohunice became contaminated as a result of incidents both before and after reactor closure in 1979. In particular, a recent spillage of corrosion inhibiting solution led to wide spread contamination in the hall. Liquid also seeped into storage holes, a trench, and into rooms below the hall. Radiation levels of up to 1 Sv/hr are present in some areas.
A decontamination plan and safety case were prepared by AEA Technology (AEAT) on behalf of Slovenske Elektrarne (SE), the utility responsible for A-1. AEAT were also awarded the contract to manage the project including the supply of specialists and key decontamination and robotic equipment. SE are responsible for providing labor, materials and decontamination reagents.
The work is divided into three tasks:
1) decontamination of the reactor hall where contamination levels are relatively low.
2) decontamination of a specific area in the reactor hall which is highly contaminated.
3) decontamination of the rooms below the hall where contamination levels are very high.
The aim of the project is to reduce contamination and radiation levels to permit decommissioning operations in the reactor hall to resume.
Decontamination of the hall is now complete, together with three cranes and various fuel handling machines. Most work up to this point has been carried out manually, either by industrial mountaineers working from ropes, or reactor staff working from a cradle suspended from an overhead crane. Lower levels were decontaminated by hand from platforms. Reagents were applied by hand and a number of reagents were developed by SE and used with great success.
High radiation and contamination levels in one area have necessitated local containment and remote decontamination methods. An area is now enclosed by a MODUCON temporary containment and decontamination is being carried out using an ARTISAN manipulator working through the containment wall. MODUCON and ARTISAN units were supplied and installed by AEAT.
Decontaminating such a large and complicated facility is a very demanding task, and the high radiation areas are particularly challenging. However, close co-operation between AEAT and SE has enabled the problems to be overcome and the project is now 1 year into the program.
INTRODUCTION
This paper describes the decontamination of a redundant nuclear reactor hall and associated equipment, which has become heavily contaminated over the years. The work is a collaborative venture between AEA Technology (AEAT), UK and Slovenske Electrarne (SE), Slovakia. On successful completion of the work, decommissioning operations will resume.
The Site
The A-1 reactor at Bohunice is a 150 MW heavy water moderated, CO2 cooled demonstration plant of Russian design constructed in the 1960s. It was fuelled by natural uranium metal fuel, clad in a magnesium-beryllium alloy. The reactor hall is large; 70 m long, 11 m wide and 30 m high and has a total internal surface area, excluding equipment, of 6000 m2. It houses the reactor charge face, fuel handling machines, and numerous

other large items of equipment including Equipment for Fuel Treatment (EFT), and is served by three cranes.
Fig. 1
History
The reactor was shut down in the 1976 following a number of accidents and is due for decommissioning. The reactor hall first became contaminated in the 1970s as a result of a reactor accident which ejected a fuel element into the hall, together with 40 tons of active CO2 coolant.
In 1991, a corrosion inhibiting reagent ,'Chrompik', a solution of dilute Sodium Chromate, was ejected from the EFT which resulted in severe contamination of the rig itself. Later 3 m3 of water was spilled onto the reactor hall floor which re-dissolved 'Chrompik' residues and distributed activity over 70 m2 of the reactor hall floor and penetrated two rooms below. Despite some remedial action local radiation and contamination levels remain high.
The Task
In 1993 AEAT found low level contamination was widespread and radiation levels varied from 10 mSv/hr in the upper regions of the hall to many mSv/hr in some areas despite local shielding. Around the source of the accidents, the EFT, maximum levels were around 200 kBq/cm2 loose contamination and over 700 mSv/hr surface radiation. The reactor hall, and its large and complex items of machinery thus represented a considerable decontamination liability.
After site visits and discussions, AEAT was awarded the contract to plan and manage the decontamination tasks at Bohunice. The tasks and objectives are as follows :

To decontaminate the reactor hall and its equipment; to permit normal working with respect to both airborne contamination and external radiation levels.

To decontaminate the EFT rig together with the adjacent floor area, including the mortuary hole covers and service trenches; to levels which permit hands-on working.

To decontaminate the rooms below the reactor hall.

The work was divided contractually into two distinct phases : the Production Phase for issue of the decontamination plan and safety case and the Realization Phase for carrying out decontamination tasks.
PRODUCTION PHASE
The first task was to agree a strategy program, and in particular to prepare a decontamination plan and safety case addressing the issues of access, reducing the risks of recontamination and the principles of radiological protection.
Planning required the collation of data relating to Slovak regulations, the plant, its history and current condition, collaboration with station engineers and site visits including a video survey.
Strategy for Decontamination
The low contamination areas of the reactor hall required simple methods; working from the highest areas down to the floor and finally to the rooms below. Walls and intricate systems of cables and wires, equipment and miscellaneous items would be cleaned manually using application reagents such as foams, pastes and gels.
For heavily contaminated areas around the EFT and the rooms below, the removal of surfaces by remote methods within ventilated zones would be necessary. Size reduction and scabbling or chiselling tools would be needed for cutting service pipes and concrete removal respectively.

Wastes generated from these processes would be put into shielded containers supplied by the station staff and handed over to them for storage and ultimate disposal. Secondary arisings would be estimated and minimized wherever possible.

Risk of Recontamination

Areas of high risk would be enclosed in a temporary ventilated containment. Two were proposed based on the MODUCONTM containment system. Radiological Protection measures would be strengthened by full use of barriered 'zones'. In addition the existing ventilation system would be upgraded and fitted with HEPA filters.

Access

The tops of the walls are 24 m above the reactor hall floor and much of the wall surface is painted and featureless; other areas are festooned with cables and services. The rails for the main cranes are mounted on the tops of the longitudinal walls; other lower ledges support smaller cranes and two refuelling machines. (See Fig. 1). A suitable method for providing access to the wall surfaces, the cranes, refuelling machines etc. was needed. Industrial mountaineers had been used for repainting work elsewhere at Bohunice, and so it was proposed and agreed to use them to decontaminate the walls and equipment high above the reactor hall floor.

Work below 13 m would be carried out from a cradle suspended from the cranes. Work on the lowest areas, including a substantial ledge and rail system, would be carried out from platforms and ladders from the floor.

Work in High Radiation Areas

Areas which had been directly contaminated with 'Chrompik' on and around the EFT and in the rooms below, posed problems of operating in high levels of radiation. Surface dose rates were recorded as high as 700 mSv/hr in places and necessitated the use of remote methods. A telerobotic manipulator, ARTISANTM was selected as the most suitable tool available, capable of carrying out all the processes envisaged in the congested surroundings of the EFT and within a MODUCON containment booth.

The Decontamination Plan

The Plan specified the scope of work, decontamination methods, waste arisings, engineering and material support, timescale and cost of the project. This was supported by a probabilistic safety case which justified the proposed actions by quantifying the hazards, and a quality program describing the management arrangements proposed to control the work.

COLLABORATION

Documentation

The proposal and safety case were duly accepted after constructive comment and revision by the plant Management team and Slovak authorities, agreeing the 2 year program.

Preparatory Work

A number of tasks were completed prior to the commencement of the work, including:

- An inventory of the reactor hall identifying over one hundred items for removal.
- Fitting a filter to the main ventilation system.
- Delivery of a mobile ventilation system for use on a MODUCON booth.
- General cleaning of the main crane and rail.
- The decontamination and painting of the internal hall roof section.
- Modification of an active waste drain from the hall into storage tanks.

Obtaining materials and machines from the UK and local sources including vacuum cleaners and a proprietary decontamination system for routine floor cleaning.

REALIZATION PHASE

Having established the overall strategy and produced a detailed decontamination plan, the next stage required the deployment of AEAT staff and equipment on site to prepare and set up for operations. The staff included a full time project manager and technical specialists in robotics and decontamination as required.

The equipment supplied by AEAT included:

An ARTISANTM remote handling system.

Decontamination equipment (including a vacuum collection unit and foam generator).

MODUCON containment booths.

Preparation

The ARTISANTM system was thoroughly tested both in the UK and in a non-active 'mock-up' at Bohunice, including its capability with regard to access and deployment of tooling. An important task for the ARTISANTM was to remove shielding and scabble concrete trenches and this was successfully demonstrated together with the vacuum recovery of debris. Decontamination tools were demonstrated and training given to SE staff.

Two MODUCON containments were erected; one small booth for handling liquid waste, and another larger facility to contain the EFT. Staff from AEAT set up the containments and ventilation equipment supported by SE operators.

During this phase of the work trials with equipment and reagents were undertaken in the UK to assess their suitability for the envisaged application. This included comparing UK and Slovak reagents permitting selection of local products.

Decontamination

The plan submitted to SE reflected strategic decisions and the work was actually carried out as shown in Table I. Many of the tasks overlapped or ran in parallel to utilize available resources.

Table I

The decontamination techniques and progress are discussed in the following sections.

The Reactor Hall and its Equipment

Significant quantities of dust were visible in the hall on all the horizontal surfaces, including machinery and the large number of miscellaneous items of equipment. Airborne activity levels fluctuated depending on work being carried out, (e.g. crane movement). The use of wet methods on a small controllable scale using foam, gels and pastes were therefore preferred, thus minimizing resuspension.

Although most surfaces (steel and concrete) were painted in an epoxy coating which eased cleaning considerably, cleaning was compounded by the complexity of the equipment and years of grime, particularly on all horizontal surfaces.

Each area was thoroughly considered and detailed working instructions were written by AEAT in collaboration with SE staff who had particular knowledge of the plant.

Detailed working practices and specific reagents were thus agreed for each area, and indeed for the many components within, including a number of chemical formulations developed by the A-1 Decontamination Manager for particular applications.

The areas and type of surface within the hall may be grouped as follows:

- Roof 'containment' constructed from corrugated materials.
- Painted walls including rails/ledges of cranes and equipment.
- Three Cranes comprising countless different components.
- Two re-fuelling machines and two manipulators.
- Floor. Painted steel or covered concrete.
- Many miscellaneous items.

An inventory was prepared of every item of equipment in the hall and a grid reference system was set up which enabled the location of each item, or section of surface area, to be referenced. Each item or area was considered in relation to decontamination technique and reagent, the work time and the radwaste arisings. A dose assessment was made for each job compliant with the dose restraint objective of 15 mSv per year per operator.

Foam reagents were tested on large areas of the walls. There was considerably more grime and grease on these surfaces than expected and the use of degreasing emulsions and gels were found to be more effective and easy to use by the mountaineers.

All horizontal areas were thick with dust and were first suction cleaned to remove dust and loose contamination. Debris interceptors and HEPA filter units were used.

The cranes, re-fuelling machines and manipulators were also coated in years of greasy dust. These were decontaminated by hand swabbing using degreasing foams, gels and pastes to minimize recontamination across the surfaces and the risk of damage.

The reactor hall floor was initially decontaminated with a proprietary steam cleaning unit to minimize the spread of contamination by foot traffic. Mops and buckets are now used for routine housekeeping as this was found more amenable to the complex floor at A-1.

Operations over 13 m were carried out by a team of 'Industrial Mountaineers' who were available locally and who were experienced with working in active areas. They gained access to the high areas of the walls and ledges via defined access routes and worked suspended on conventional mountaineering equipment secured from above. Each man carried with him swabs, sponges and buckets of reagent to complete 1-2 m2, returning to replenish materials from time to time. After treatment, all areas were rinsed by wiping with a dilute detergent.

Fig. 2

Operations between 3-13 m were carried out by teams of SE operators working from a cradle suspended from a convenient crane, using hand swabbing methods and reagents.

The lowest 3 m areas, together with a substantial service ledge comprising pipes cables and a rail, were cleaned from floor level, accessed by ladders where necessary. Wall surfaces were festooned with a complex array of services and pipes and methods using emulsions and abrasive pastes applied by sponge were successful

The other equipment in the reactor hall, over 160 items in all, were also decontaminated by 'hands-on' methods using a variety of reagents.

The Equipment for Fuel Treatment (EFT)

This equipment consists of two heavily shielded cylinders in which canned fuel was drained, cropped and repackaged for transport. The immediate floor area surrounding the EFT contains numerous complicated features such as floor storage holes, and a service trench. The main problem is that of gaining access to the many contaminated surfaces to carry out

decontamination. For example, the dose rate on the bottom of the concrete service trench is 750 mSv/hr, but the trench was filled with 4.5 tonnes of steel shot and covered with large lead bricks. A MODUCONTM containment now encloses the EFT preventing the escape of active dust re-suspended during decontamination operations.

Fig. 3

The ARTISANTM hydraulic manipulator is being deployed through the walls to provide access and to carry out the decontamination operations. All the temporary shielding (lead bricks and steel shot), together with all service pipes have now been removed using ARTISANTM. Lead bricks were picked up individually by ARTISANTM and placed in a basket which was removed by crane through a hole in the MODUCONTM containment for decontamination later. Steel shot was removed by suction using a powerful vacuum unit supplied by AEAT which automatically collects debris (dust/spoil/shot) in 200l drums ready for disposal. This proved a highly efficient process.

Services pipes were cut using hydraulic shear tools deployed by ARTISANTM and later removed and packaged into shielded drums through the Waste Posting Port fitted to the MODUCONTM containment.

Operations continue. Concrete removal by a small scabbling tool with vacuum recovery proved too slow and produced excessive quantities of very fine dust by-passing the debris cyclone and collecting directly on the filters of the spoil collector; causing local radiation levels to rise. Up to 20 mm of concrete is now being removed from the floor using chisels deployed by ARTISANTM. Debris is being shovelled up and deposited in tins for posting out in containers: shielded to minimize dose to operators during handling and storage. When ARTISANTM operations are complete, operators will enter the MODUCONTM to unbolt the EFT for removal to another area for dismantling and decontamination at a later date.

The Rooms below the Reactor Hall

These two rooms are situated one above the other, below the reactor hall floor. The ARTISANTM manipulator will be installed through a shielded access plug in the reactor hall floor to carry out the decontamination of the upper room.

Fig. 4

The lowest room will be accessed through a MODUCON containment booth constructed around a door at floor level. The highly contaminated floors and stairs will be decontaminated by scabbling and chiselling, debris being collected in shielded drums using the vacuum unit. The steel duct will be treated by swabbing with appropriate reagents. Remote decontamination and size reduction will be carried out by ARTISANTM to reduce dose, allowing manual decontamination to follow, using suction cleaning, scabbling and treatment with foams.

SUMMARY AND CONCLUSIONS

As a result of a small number of operational accidents, the plant at Bohunice represented a considerable decontamination liability. Having been awarded a contract to plan and implement the decontamination, AEAT faced a number of technical challenges and responded with appropriately engineered solutions to meet them in collaboration with the local Slovak management.

Selection of methods for a specific task depends on a number of local factors including the resources and the type of waste treatment facilities available. Tasks such as the reactor hall have the potential to produce a very large volume of secondary wastes and it is important

that the most appropriate technology is selected to minimize arisings. Furthermore such large facilities pose special difficulties with regard to decontamination on account of their size, access and, in some cases, high radiation levels.

The resulting decontamination plan and supporting documents have gained the approval of both the customer and the Slovak authorities. The work has commenced and is now one year into the program. The joint AEAT/SE management team, together with the station staff and contractors, have maintained a flexible response to a number of technical and contractual constraints. This teamwork has played a significant part in the success of the work to date.

ACKNOWLEDGEMENTS

The development of the ARTISANTM remotely operated manipulator was funded through the UK Department of Trade and Industry.

Dr D Keltos, (VYZ.o.z, A-1, NPP Bohunice) Slovakia, for specific reagent formulations used in the decontamination of the Reactor Hall and its equipment.

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THE JAPAN POWER DEMONSTRATION REACTOR DECOMMISSIONING PROGRAM - LESSONS LEARNED ON OPERATION AND MAINTENANCE OF AUXILIARY EQUIPMENT DURING DISMANTLING ACTIVITIES

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ABSTRACT

Dismantling of the Japan Power Demonstration Reactor Decommissioning Program (JPDR) was initiated in December 1986, and planned to be finished by March 1996. During this period of dismantling, several useful lessons have been learned such as use of existing facilities, dismantling tools, and project management. Among these, facility maintenance and operation are two key elements that must be considered for effectively conducting dismantling activities.

INTRODUCTION

The Japan Atomic Energy Research Institute (JAERI) is conducting the decommissioning demonstration project of the Japan Power Demonstration Reactor (JPDR) (1,2). The JPDR decommissioning project has been successfully conducted so far; since March 1994, almost all components have been removed from the project site, and the project is currently in its final stages and is planned to be finished by March 1996. Decommissioning efforts have been focused on demolition of buildings.

The following provides a list of objectives for dismantling the JPDR:

- to demonstrate the techniques developed in Phase 1,
- to obtain experience on dismantling activities, and
- to establish decommissioning database for future decommissioning projects of commercial nuclear power plants.

Various cutting tools with remote operation developed in the early state of the project have been used successfully in dismantling highly-activated components. These techniques were proven to be useful in minimizing radiation exposure to workers.

During dismantling activities, various data on project management have been collected continually and are stored in the decommissioning database (3). In addition, several useful lessons have been learned from the dismantling activities such as use of existing facilities, dismantling tools and project management. Among these lessons, facility maintenance and operation are two key elements that must be considered for effectively conducting dismantling activities.

The lessons learned regarding the maintenance of existing facilities are discussed in this paper.

OUTLINE OF THE JAPAN POWER DEMONSTRATION REACTOR DECOMMISSIONING PROGRAM

Facility History

The JPDR was a prototype nuclear reactor built to gain experience on construction and operation of a light water power reactor and to train engineers in nuclear technology. The plant provided nuclear power generation for the first time in Japan in October, 1963. Since then, training of operators and irradiation tests have been conducted.

In March 1976, the plant was shut down as a result of problems in the cooling system. After a few years, it was decided the JPDR should be used for development of reactor decommissioning technology in consideration of future decommissioning of commercial nuclear plants. Table I provides a list of the major specifications of the JPDR.

Table I

Japan Power Demonstration Reactor Decommissioning Program

The JPDR decommissioning program was initiated in 1981 under contract with the Science and Technology Agency (STA). The purpose was to develop and confirm the appropriateness of these technologies by applying them to actual dismantling activities. The research and development of decommissioning technology began in 1981 and nearly finished by 1986. The mock-up test facility was built in a space neighboring on the south of the JPDR, and several mock up tests of dismantling technology were initiated there.

Dismantling Activities

Actual dismantling began in December 1986 with a planned completion date of March 1996. Figure 1 shows the JPDR facilities before the dismantling started in the ground plan. Among the buildings shown in the figure, the office building, the dump condenser building, and the warehouse will remain after completion of the JPDR decommissioning program. Figure 2 illustrates the schedule of the JPDR decommissioning program.

Fig. 1

Fig. 2

During the first stage of the JPDR dismantling activities, machinery around the reactor in the reactor building was removed to provide necessary space for installation and operation of the large remote-operated cutting tools needed to dismantle the JPDR reactor. The reactor core and the biological shield were then dismantled in the order from inner to outer parts. The major dismantling activities are described in the following paragraphs.

The reactor internals were removed by the underwater plasma arc cutting system. The plasma torch was operated, in most cases by a mast type manipulator having four degrees of freedom. Otherwise, the master-slave robotic manipulator was used for handling the plasma torch to demonstrate and verify its newly-developed robot technology.

The Reactor Pressure Vessel (RPV) was dismantled using an underwater arc saw cutting system, after removing the pipes connected to it. Before

installing the underwater arc saw cutting system, a temporary cylindrical water tank was installed in the space between the RPV and biological shield. The tank was filled with water for cutting the RPV underwater. Following the dismantling of the RPV, the neutron-activated portion of the biological shield was dismantled by using both the diamond sawing/coring and abrasive water jet cutting techniques. First, the diamond sawing/coring technique was performed, followed the abrasive water jet cutting technique which was used to cut the upper and lower half of the projected portion of the biological shield, respectively. The controlled blast was applied to the remaining portion of the biological shield, which was only slightly activated. At first, the layer of 40cm width from the inner surface was dismantled by explosives installed into horizontal holes drilled by workers. The waste was placed into containers for storage. The outer layer of the biological shield was dismantled by both controlled blasting and conventional tools. The concrete from the outer layer is being disposed in the shallow land burial place in a site at the JAERI as the first test case.

OPERATION AND MAINTENANCE OF AUXILIARY EQUIPMENT

Refurbishment of the Entrance Room to Radiation Controlled Area

Approximately 200 workers a day were expected to access the radiation controlled area when the full-scale dismantling began. Therefore, the existing entrance room to the radiation controlled area located in the control building was enlarged. Refurbishment of the entrance room was performed from October 1986 to March 1987, and a temporary control gate was established in the equipment entrance located in the northern part of the turbine building while the existing control gate was under construction.

After construction, the enlarged entrance room was equipped with a reception counter, lockers, changing area, decontamination equipment, whole body monitor and personal dosimeters (alarm pocket dosimeter and film badge).

Existing equipment, such as the water supply, drain system, power supply, and ventilation system, were utilized for the refurbished entrance room. The room was located a distance of about 30m from the waste treatment facility and was interconnected by piping. In 1991, a small-scale liquid waste disposal system was established in the dump condenser building because the existing waste treatment facility was going to be dismantled. Then, liquid waste piping from the entrance room was connected to the new liquid waste system and the piping was extended to 50m through an underground tunnel.

In 1994, a new facility for access to the radiation controlled area was installed beside the dump condenser building. It was designed as an exclusive entrance for the dump condenser building.

Dismantling of the existing entrance room began in October 1994 and was finished in March 1995. Since then, the dump condenser building entrance room was temporarily arranged and used as the entrance for whole radiation controlled areas in the JPDR. However, the temporary entrance room was too small to accommodate several workers, and adequate time was required to access the work sites.

Demolition of major buildings such as the turbine building began in April 1995, and several access gates to the controlled area were established near these buildings, then the route to another part of the facility was closed and the entrance room of the dump condenser building was rearranged as the entrance room.

Utilization of Existing Liquid Waste Treatment System

The original liquid waste system was composed of six waste tanks, two systems of waste filters, a waste demineralizer, and several small tanks. When the JPDR decommissioning program began, three large tanks and one system of waste filter was not in use. In addition to this equipment, two seawater pumps (1,500 ton/hour flow) for condenser cooling system were used for waste dilution when liquid waste was discharged to the ocean.

In the JPDR decommissioning program, acceptable criteria of radioactivity and pH of liquid waste was established, radioactivity was 3.7Bq/cc and pH was 5.8~8.6. Therefore an exclusive filter system was installed when the dismantling work such as plasma cutting of the reactor internals, and removal of the RPV using an arc saw started, in which large quantities of radioactive liquid waste were produced.

The existing liquid waste treatment system was utilized while dismantling of the relatively highly contaminated or irradiated components was performed. Before removal of this system, a small-scale liquid waste system was constructed in the basement floor of the dump condenser building in 1991. The liquid waste system was equipped anew and consisted of three waste tanks and dilution water piping from the water service system. The place where waste tanks were constructed was too narrow for a large tank. Therefore, each tank had a capacity of approximately 1.5 m3, which is only one-tenth the size of former waste tanks.

The new waste tank was completed, but frequency of waste treatment increased as a result of the small tank capacity. Because the new waste treatment system had no filtering device to reduce radioactivity in the liquid waste, it was impossible to discharge even slightly radioactive waste to the ocean. Then, the liquid waste in waste tanks had to be transported to a waste treatment facility (research establishment) in Tokai. Reducing the amount of liquid waste throughout the progress of dismantling, reduced frequency of the waste treatment to twice a month. Figure 3 illustrates the new waste tank.

Fig. 3

Utilization of Existing Ventilation System

Components of existing ventilation system were in full service when the actual dismantling activities were initiated. After removal of components around the reactor and the spent fuel storage pool in 1991, delivery blowers were shut down and exhaust blowers were utilized for the ventilation of the reactor building. In 1993, an exclusive exhaust pipe for the ventilation system of the dump condenser building was remade. After that, other existing ventilation equipment was shut down and temporary blowers were installed by contractors who were dismantling object parts.

Dust and fumes generated by dismantling activities are dissimilar from fumes generated by the maintenance of plant operations. The local ventilation systems and contamination control envelopes were installed temporarily at each working area in consideration of dust and fume characteristics. During the JPDR dismantling project, the local ventilation ducts were connected to the existing ventilation system. The local ventilation system has its own filter so that the influence to the existing ventilation system was very small. However, leakage was observed that resulted in the choking of filters in the existing system, while the chamber-type local ventilation system was in use. The likely cause of the leakage was insufficient air tightness of envelope and local ventilation systems. Because it may be difficult to avoid leakage from a temporary

contamination control envelope, it was determined that the self containment type local ventilation system was more effective than the chamber type.

Utilization of Existing Power Supply

In the turbine building and the dump condenser building when the dismantling project began, there were respective systems for power receiving equipment of 3,300V from the central power station in the Tokai research establishment.

In the mock-up test facility, the 6,600V power supply equipment was installed; it provided three phase 3,300V, 440V, 210V and single-phase 210V/105V for dismantling activities and several tests performed in this facility. When the reactor pressure vessel was dismantled with the underwater arc saw cutting system, 3,300V of power was supplied directly.

In 1987, the exclusive supply system for dismantling work was installed near the reactor enclosure. The power of 3,300V was supplied from the mock-up test facility and provided 220V to several distribution panels in the dismantling sites.

Removal of components and the power supply in the control room and the turbine building, respectively were conducted in 1992. The power supply for the turbine building and others was modified so that the power was provided from the mock-up facility instead of the existing power supply. Figure 4 illustrates the mock up test facility power supply.

Fig. 4

However, the power supply of the dump condenser building was detached from the turbine building power supply and modified to provide the power to remaining buildings, such as the office building, the warehouse, and the dump condenser building.

When the power was provided to the building where the dismantling was performed on the mock-up test facility, some problems occurred. Heavy loads were added to the system; for example, start-up of big ventilation fans, and severe voltage drops occurred, resulting in malfunctioning of monitors and other instruments. The power was sometimes interrupted locally by function of the protection circuit. Because the mock-up test facility power supply was designed for mock-up tests and several cutting machines, the power capacity was not enough for both original use and temporary use. Some rearrangement of ventilation fans start-up processes and changes of protection circuit settings were solutions to these problems.

SUMMARY OF LESSONS LEARNED

Since the entrance to radiation controlled areas basically consisted of existing equipment, refurbishment construction was required several times during dismantling activities. Relatively large amounts of workpower were necessary for refurbishment work; this was an inefficient system. Instead, it may be more efficient to built temporary control gates instead of utilization of existing entrance room.

The existing liquid waste system was used while the components were removed and replaced with small-scale equipment. However, it was too small to treat liquid waste from dismantling activities. Capacity and arrangement of substitute liquid waste systems should be carefully studied.

The contamination control envelope and local ventilation system were used in dismantling work. The duct of the local ventilation system was connected to existing ducts and at times, leakage occurred from the temporary equipment, resulting in choking of filters in existing

ventilation systems. Selection of blowers for the local ventilation system and arrangement of contamination control envelopes are important to avoid problems in existing ventilation systems.

When electric power was provided temporarily from the mock-up test facility, some problems occurred as a result of the power supply shortage. Temporary power supplies must have enough capacity for not only dismantling machines, but also maintenance of facilities after removing existing power supplies.

In the JPDR decommissioning program, auxiliary equipment was dependent on existing equipment. Therefore it was required to separate some components from the equipment in use for facility maintenance. During initial removal of components, a detailed study on the dismantling process was important to avoid any expected problems. Information exchange between dismantling and maintenance groups was indispensable.

CONCLUSION

The JPDR decommissioning program has been successful and is now in the final stages. Many useful experiences and lessons learned on the decommissioning and dismantling activities can be gained. In addition, various data on project arrangement collected in the dismantling activities are stored in the decommissioning database. These resulted from the JPDR decommissioning program and will contribute to future decommissioning of commercial nuclear power plants.

In such activities of dismantling, facility maintenance and operation were inconspicuous but important for effective dismantling activities. Effective utilization of existing equipment was important for dismantling activities. Through the dismantling program, auxiliary equipment was carefully studied in each stage of the decommissioning program. Knowledge from these studies may be useful for future decommissioning projects of commercial nuclear power plants.

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ROSIE: A MOBILE WORKSYSTEM FOR DECONTAMINATION AND DISMANTLEMENT OPERATIONS

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ABSTRACT

RedZone Robotics, Inc. and Carnegie Mellon University's Field Robotics Center have undertaken development of an advanced remote worksystem capable specifically designed to meet the challenges of performing a wide range of decontamination and dismantlement operations in nuclear environments. This project is funded by the US Department of Energy's (DOE's) Environmental Management Office of Technology Development through the Morgantown Energy Technology Center. Currently, we are working on the third phase of this effort; having completed the design and fabrication

of the worksystemRosie, we are concluding endurance testing and characterization of Rosie's work capabilities, and commencing enhancement of the system design to ensure suitability for use in a DOE facility decommissioning project.

The Rosie worksystem includes a locomotor, heavy manipulator, operator console, and control system for remote operations. The locomotor is a highly mobile platform with tether management and hydraulic power onboard. The heavy manipulator is a high-payload, long-reach boom used to deploy a wide variety of tools and/or sensors into the work area. rosie's advanced control system, broad work capabilities and hardening/reliability for hazardous duty make it a new and unique capability that facilitates completion of significant cleanup projects throughout the DOE and private sector.

The paper presents a system overview, summarized results from endurance testing, and future plans.

INTRODUCTION

The development of the Rosie worksystem has occurred in three separate work phases. The first phase consisted of gaining a knowledge of the DOE's D&D needs, upgrading a pre-existing worksystem with state-of-the-art technologies and enhanced controls for ease of operation, then undertaking a program to test the system and determine its capabilities and weaknesses. The second phase involved developing a second-generation worksystem (Rosie) to perform D&D operations based on knowledge gained in Phase I. During the third and current phase Rosie is being tested extensively to determine general capabilities to deploy tools and perform decontamination and dismantlement (D&D) tasks, as well as assess reliability and maintenance issues.

The tasks and constraints which characterize D&D applications are essential background in the development of this technology. The environment in which such worksystems must perform D&D operations range from areas in which no worker protection is needed to areas in which human entry is precluded. Dangers can include exposure to alpha, beta, and gamma radiation; uranium, plutonium, and tritium; volatile organics, acids and caustics; mercury; TRU waste, asbestos; and mixed waste. Facilities in which D&D is likely to occur include uranium enrichment facilities (including gaseous diffusion plants, centrifuge plants, and other separation plants); research and production reactors; hot cells, canyons, and vaults; stacks and cooling towers; silos and waste storage tanks; analytical research labs; and weapons production and assembly facilities.

Given the wide range of tasks that must be executed, the hazards present, and the difficulty in predicting conditions or comprehensively understanding task needs, remote technology must be highly versatile and reliable. These worksystems require capabilities to handle a variety of tools, they must combine brute force for heavy work with dexterity for fine manipulation, they must be reliable for extended use in areas where human intervention is difficult or impossible, and they must be adaptable to a range of work conditions and settings.

In the second phase of this project we undertook the design and fabrication of a worksystem specifically designed to meet D&D needs. We discovered in our Phase I study, the requirements for a worksystem are very diverse, ranging from human-scale manipulation tasks to large, industrial-scale equipment removal. While it is impossible to build one system capable of meeting all of D&D needs, we selected a concept capable

of addressing a major segment of tasks for which current technology is inadequate.

WORKSYSTEM DESCRIPTION

Rosie is a mobile robot worksystem developed for nuclear facility decommissioning and dismantlement. Its primary function is to perform a variety of dismantlement tasks remotely by deploying tools, sensors, and/or other robotic equipment in hazardous areas. Rosie's capabilities and system design address the need for durability and reliability in these environments, and enable performance of tasks such as piping and process equipment removal, structural demolition, vessel segmentation, waste handling and transport, and wall/floor decontamination.

The system includes a tethered robot, a power distribution unit (PDu), and a control console for robot operation. The robot consists of two major subassemblies, the locomotor and the heavy manipulator. The locomotor is a hydraulically powered, omni-directional platform with onboard tether management. It provides mobility to transport the heavy manipulator, tools, or other payloads within the work area. The heavy manipulator is a four degree of freedom, high-payload, long-reach mechanism capable of carrying a variety of tools, one or more dexterous manipulators, or any other payload of up to 900 kg (2,000 lb) throughout a generous work envelope. Rosie is a teleoperated system with low-level automation features that facilitate more efficient remote operations and allow a single operator to maneuver and work effectively.

Fig. 1

Locomotor

The locomotor is a mobile platform with specifications as shown in Table I. Its frame is an aluminum weldment which supports wheel modules at each corner. Each wheel module has independent drive and steering motions providing an omni-directional capability.

The front two wheels are mounted on extensions which can change the front wheel tread width from 193 cm (76 in)to 345 cm (136 in). The two rear wheels are mounted on a pivoting beam which allows each wheel 5 cm (2 in) of vertical travel for obstacle negotiation.

Located within the locomotor is the hydraulic power supply, which is a 45 kW (60 Hp) supply, providing 114 l/min (30 gpm) of hydraulic fluid at 20.7 MPa (3,000 psi) for all robot motions. The hydraulic fluid reservoir is located at the front center of the locomotor. Directly behind it is the hydraulic pump and its electric drive motor. All of the control valving for the system is located above the pump and motor, inside the locomotor frame. Filters, an accumulator, and the hydraulic fluid cooling equipment are all located in one of two side enclosures suspended from the frame. The other side enclosure contains all onboard control electronics for the system. At the rear of the machine is the tether reel which can carry up to 53 m (175 feet) of tether (up to 38 m (125 ft) of unreeled tether can be included to extend the vehicle's range).

Table I

Heavy Manipulator

The heavy manipulator is mounted on the deck of the locomotor. It is a four degree-of-freedom mechanism providing a long-reach, high-payload capability for tool deployment. It can carry up to 900 kg (2,000 lb) with a 6,800 Nm (60,000 in-lb) moment load, at a distance of 6 m (20 feet) from the shoulder joint. The heavy manipulator consists of four joints; a waist rotation motion on the locomotor deck, a shoulder pitch, a forearm extension, and a wrist pitch at the tip of the forearm. Each of

the four joints has integral position feedback and is servo-controlled based on operator commands. The configuration of the heavy manipulator is shown in Fig. 2 and its specifications are shown in Table II.

Fig. 2

Table II

Figure 3 shows the tip over load limits of the locomotor for loads at the wrist of the heavy manipulator, with the front wheels fully extended, a 900 kb (2,000 lb) counterweight mounted on the manipulator turret, and the rear pivoting axle in its locked position. These load limits include a 455 kg (1,000 lb) safety margin to accommodate dynamic loads.]

Fig. 3

Feedback

Rosie's operator is provided with a complete set of feedback information to support remote operations. Audio and video feedback are provided from onboard microphones and up to ten onboard cameras. Various other onboard sensors provide full system status and health monitoring. Additional user-specified sensors can be installed to provide remote monitoring of key environmental parameters.

Audio/Video System

The audio/video system takes multiple camera views and microphone inputs from the robot and displays them at the console. Rosie can support up to 10 cameras including the following:

- Four cameras with remote focus, zoom, lights, and pan and tilt motions
- Four cameras with remote lights and tilt motions (fixed focus)
- Two cameras with remote lights (fixed focus)

All cameras are modular to allow easy replacement or relocation in order to accommodate different tooling or task requirements.

System Status/Health

The status and health of the system are constantly monitored by various onboard sensors. These include hydraulic fluid temperature, pressure, and reservoir level. In addition, the voltage levels of onboard electronic components as well as their temperatures are also monitored so that a fault may be detected before it can cause a complete system failure. Control and sensing signals are monitored automatically and error checking is performed to ensure reliable communications.

Position Sensing

All of the remotely controlled motions of the system incorporate position sensing. Locomotor wheel steering and drive motions are equipped with resolver feedback which is utilized by the computer control system to coordinate these motions in several different driving modes. This also provides the operator with a quick means to determine wheel positions. The four heavy manipulator motions (waist rotation, shoulder pitch, forearm extension, and wrist pitch) also have resolver feedback. Again, this allows computer controlled coordination modes and a clear understanding of heavy manipulator position/orientation for the operator. The front wheel extensions incorporate limit switches so that the operator can easily discern whether they are extended; this information is also used by the control system in performing automated initialization sequences. The tether reel has limit switches so that the tether cannot be completely unwound from the reel, or wound on beyond the reel's capacity.

User-specified Feedback

The system has the capacity to support user-specified sensors installed on the robot and transmit their data back to the console. Such sensors

can be used to provide remote monitoring of key environmental parameters, such as radiation levels, ambient temperature, the presence of toxic gases, etc.

Control System

Rosie's control system is comprised of an operator control console shown in Fig. 4 and onboard control system components linked by a telemetry system. Control system functions are distributed across two primary computers (CPU's) one in the console and one onboard the robot. The control console CPU displays status and sensor data coming from the robot, interprets signals from joysticks and other switches, and sends appropriate commands to the onboard CPU. The onboard CPU executes commands from the console by closing motion control loops, acquiring sensor data, coordinating axes, and activating video and other onboard equipment. Both CPUs perform continuous error checking and monitoring of communications between the robot and console.

Using this control system, a single operator stationed at the console can control the Rosie worksystem. Primary system functionslocomotor, heavy manipulator, system power, tether, and camerasare controlled using switches and joysticks on the desk top. Less frequently used functions and status information are accessed through the touch screen. Three video monitors, with quad-splitting capabilities, display the onboard camera views. The operator can select any camera view for any of the monitors using the touch screen controls. In this way, each operator can configure the control console monitors to suit his or her particular preferences. In addition, the views can be changed during operation of the system, as needs arise.

Fig. 4

Control Modes

All axes are servo-controlled enabling precise, variable speed motion control for dexterous positioning either by teleoperation or by computer control. This servo-control allows the computer to coordinate the motions of the locomotor wheels in any of three different steering modes. In addition, the heavy manipulator can be operated in two different control modes. These modes are as follows:

Steering Modes

The locator wheels are controlled in any of three driving modes:

4-wheel steering: Front and rear wheels steer in opposition, allowing a turn of any radius, including a pivot about the vehicle's center.

Crab Steering: All wheels steer in the same direction. This mode is especially useful for tight maneuvering and allows an operator to translate side-ways and work continuously along a wall surface, eliminating the need to frequently back away and reposition the vehicle.

Rotate-about-a-point Steering: Wheels automatically steer to turn the locomotor about a predetermined point. Assigning the tool location at this point allows the vehicle to be repositioned without moving the tool.

Boom Modes

The heavy manipulator can be controlled in either of two modes:

Joint Control allows the operator to individually control each joint on the heavy manipulator at a continuously variable speed.

Coordinated Control allows the operator to steer the endpoint of the heavy manipulator and all four joints are automatically coordinated to achieve Cartesian motion. This control mode is an efficient and intuitive way to control the heavy manipulator and allows an operator to perform

difficult tasks like tracking a wall or floor surface with a single joystick motion.

Power and Telemetry

The power and telemetry subsystem allows power and signals to be transmitted from the console to the locomotor and routed onboard to the various sensors and actuators. A Power Distribution Unit (PDU) located between the console and robot provides a location to input site electrical power needed for onboard functions. A tether is used to transmit all power, control, and video signals to and from the robot. All signals from the console pass through the PDU and are combined with the power and routed into the tether. When operating in a contaminated location, the PDU can be located outside of containment, minimizing the number of conductor penetrations required through containment.

The heart of the electrical system onboard the locomotor is enclosed in a sealed box mounted on the left side of the frame. This enclosure houses transformers, control computing, power supplies, video modulation equipment, and heat exchanger units.

Tooling and Auxiliary Services

A wide variety of tools or dexterous manipulators can be deployed from the heavy manipulator or locomotor deck. Highly accurate variable-speed motion control allows an operator to position tools quickly and perform work tasks effectively. Rosie's work envelope allows floor to ceiling reach with most tools.

Both hydraulic and electric power are available at the boom tip to power tools. As much as 57 1/min of hydraulic fluid at 207 bar (15 gpm at 3,000 psi) and 20 amps of 120 VAC power are available. Any user specified tooling can be deployed subject to powering and payload (up to 900 kg/2,000 lb) constraints, including:

Component Removal	Decontamination
hydraulic pipe sheer	pressurized water
reciprocating saw	CO2
abrasive disk	mechanical scabbler
impact wrench	sealant spray
plasma torch	
Demolition	General Purpose
jackhammer/breaker	dexterous manipulator
pulverizer	dual-arm work system
concrete hole saw	
abrasive water jet	
Material Handling	
wet/dry vacuum	
excavation bucket	
dozer blade	
drum grapple	
cable winch	

SYSTEM FEATURES

The Rosie worksystem incorporates many onboard and offboard features which provide significant benefits in remote dismantling. With construction-grade durability, high maneuverability, and power to spar, Rosie can operate effectively to deploy tools, transport materials, and meet the unexpected challenges of D&D.

Work Capability

Rosie is capable of deploying a wide variety of tools and other payloads throughout a generous work envelope. The heavy manipulator extends to reach 26 ft above the floor and at least 12 ft on all sides of the locomotor. All wheels are independently driven and steered, making Rosie highly maneuverable in tight or cluttered spaces. Front wheels extend for added stability. Rosie can be driven with wheels extended or retracted. The pivot-mounted rear axle provides compliance when working on uneven floors and crossing obstacles. Rosie is hydraulically powered, providing high power density suited to dismantling work; as a hydraulic system, Rosie is intrinsically sealed against contamination.

Reliability

Reliability is essential in environments where manual recovery of failed equipment is difficult, costly, or precluded by hazards. The rugged construction of this system is suited to the abusive conditions of dismantlement operations and it is designed to withstand inadvertent collisions or falling objects. The electrical system is designed with sufficient noise immunity and error monitoring to ensure the reliable communication of control signals between the operator's control console and the robot. Sensors are used to monitor the status of critical components and automatically alert the operator of potential problems. Critical actuation's, such as driving and steering, are functionally redundant each wheel module is individually driven and steered, and sufficiently powered to compensate for limited failures. The wheel drive motions freewheel when unpowered to enable emergency recovery towing. Rosie is a tethered system for guaranteed communications and power for extended work durations. Onboard tether management ensures that the tether is not endangered by being dragged.

Decontamination

In nuclear environments, the ability to decontaminate equipment is critical to allow maintenance, storage, and transportation of equipment without incurring personnel exposure. All onboard components on this system are sealed for pressurized washdown. The system's structures are designed to minimize exposed surfaces and areas where contamination can collect and be trapped. Areas that can't be sealed are left as open as possible in order to facilitate cleaning and washdown.

Radiation Hardening

This system is designed to operate in areas where radiation exposure is present. Materials and components have been selected to reduce the potential for radiation degradation. The robot portion of the system is designed to withstand a cumulative radiation dose of 105 R. Higher levels of radiation hardening are achievable if necessary by shielding of critical electronics and using more radiation tolerant components.

Ease of Operations

Rosie can be used to perform D&D tasks without reducing an operator's efficiency or requiring specialized skills. Automation of low-level functions and other control system features allow a single operator to maneuver both the locomotor and heavy manipulator and to work very efficiently.

All motions incorporate position sensing and servo-control, enabling precise motion control for dexterous positioning either by teleoperation or computer control. High resolution and continuously variable speeds allow an operator to move slowly for fine positioning, or quickly for efficient large motions.

Up to 10 onboard video cameraswith lights and pan/tiltsprovide an operator with effective views for navigation and tool deployment.

Modularity and Maintainability

The system is designed to be as modular as possible to expedite maintenance and deployment of alternate tools. Modularity allows the quick replacement of components or subsystems in order to keep the system in service, and allowing failed components to be repaired offboard and off-line. Critical components in the electrical and hydraulics systems are readily accessible and can be modularly replaced.

SUMMARY OF ROSIE TESTING AND LESSONS LEARNED

Over four months of testing have been performed to date at Oak Ridge National Laboratory using the Rosie worksystem. The goal of this testing was to determine areas requiring refinement or modification and to quantify the system's overall capabilities. Over 100 hours of operations have been logged deploying tooling in concrete demolition and metal cutting exercises. Concrete demolition was performed on more than ten reinforced concrete highway barriers and other pre-cast shapes using a hydraulic breaker tool mounted to the boom tip. Metal cutting tasks have been performed using an abrasive wheel grinder deployed from a master-slave dexterous manipulator mounted to the boom tip. Most testing has been performed outdoors; rain occurred on several occasions (operators were unaffected), and the system was stored outdoors when not in use for several weeks.

Overall, the system performed without failures; some minor adjustments were necessary and several design enhancements have been identified to improve durability and operability. Target areas include:

- improve function and reliability of tether management fairlead.
- add reinforcements to several areas on the heavy manipulator to reduce deflection and weld stress.
- secure hardware and components against long duration vibration loads.
- optimize control console for more efficient, less strained operations (switch and touchscreen layout, joysticks, etc.).
- reduce onboard control system complexity to increase reliability.
- increase hydraulic system cooling.
- upgrade hydraulic valving to improve stability and reliability.

FUTURE WORK

Through endurance testing we have evaluated the system and its ability to perform D&D tasks. Overall, the system performs well and meets the general requirements of D&D applications. Phase III testing will continue to characterize Rosie's capabilities and test enhancements that are currently being added. Testing and enhancement activities are also being focused on fully addressing issues such as decontamination and ease of maintenance. We are also focusing toward specific DOE facility decommissioning projects, including Argonne National Laboratory's CP-5 reactor, and commercialization of this system as a broadly applicable worksystem for hazardous applications.

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DECOMMISSIONING OF REPROCESSING PLANTS AT BNFL SELLAFIELD - A PRACTICAL SUCCESS

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ABSTRACT
The BNFL Sellafield site has developed extensively since the start of the first construction in the late 1940's and even without further development there will have been some 120 radioactive plants in operation. The decommissioning of these plants requires a co-ordinated and integrated program. The paper describes BNFL's approach to the challenge, its overall strategies, the constraints and the progress achieved to date using real experience from the 18 decommissioning projects currently underway.

INTRODUCTION

When the current reprocessing programs are complete the BNFL's Sellafield site will have contained over 120 radioactive plants all of which will require decommissioning at a currently estimated cost of almost $9B. A formal decommissioning program was initiated in the early 1980s and has expanded to reach the current level of eighteen plants undergoing decommissioning and a program of work stretching over several decades. The management philosophy needs to ensure the optimum approach both for safety and cost, satisfactory interaction with the regulatory bodies and the effective management of the practical work.

THE SELLAFIELD SITE

Initial work at Sellafield in the late 1940s involved the construction of the two Windscale Pile reactors and the associated plants to cool, decan and reprocess the fuel together with the plutonium and uranium finishing lines and the associated waste and effluent treatment facilities. In the mid 1950s the first of the Magnox reactors came on line and it was apparent that the proposed expansion of the civil Magnox program could not be supported by the existing reprocessing plants. This led to the construction of the Magnox reprocessing plant which came on stream in the mid 1960s together with additional pond storage and decanning facilities, many of the existing waste and effluent plants however continued to be utilized with some capacity enhancements. This philosophy of maximizing the utilization of capital investment has continued and the recently commissioned THORP plant for the reprocessing of oxide fuels utilizes supporting plants from both the original military and Magnox programs. Reprocessing of Magnox fuel is anticipated to continue until the later part of the first decade of the next century with oxide fuel reprocessing continuing some years longer. Additionally there is an accumulation of highly active liquors and medium active wastes (mainly fuel cladding) which needs to be recovered and conditioned ready for ultimate disposal. The site interactions are therefore extremely complex and the decommissioning program both for the plants already shut down and operating plants must be integrated with the operational needs of the site.

OVERALL STRATEGIES

BNFL brings together all its waste and decommissioning liability planning in the Waste Management Review (WMPR) and the Decommissioning Review (DPR). These reviews summarize the strategies and programs together with the associated costs thus allowing optimization on a macro scale and the calculation of the necessary financial provisions. WMPR addresses the waste routing and treatment issues together with interim storage. Its aim is to minimize the overall cost by maximizing the utilization of existing facilities and minimizing the number of new plants needed. DPR recognizes the longer term nature of the decommissioning program, anticipated to extend beyond 2100, and seeks to priorities and program the plant

decommissioning to achieve the minimum discounted overall cost recognizing such factors as plant risk, surveillance and maintenance costs, waste route availability, interaction with other plants etc.. Not surprisingly the more fragile plutonium facilities attract a higher priority for early dismantling than for example the extremely robust reactors. Decommissioning cost modelling techniques (1) have been developed to facilitate this long term programming.

PROJECT STRATEGIES

Once identified within the overall decommissioning program the actual project strategy is developed on a case by case basis recognizing the specific features of the plant any links with operating plant and waste routing. Extensive optioneering involving techniques such as Kipner Traego and value engineering are utilized calling on the extensive knowledge of the plants available on the site and experience on similar projects to ensure the optimum cost and safety effective solution is identified. It would be normal to carry out a HAZOP one study on the preferred option to ensure there are no obvious safety show stoppers before investing engineering effort in further design and planning. Any necessary development work in support of the project would normally be identified at this stage. Although every effort is made to obtain all possible plant data it is frequently necessary to recognize a number of uncertainties and projects often proceed in a number of phases where a scope of work can be reasonably well defined, both for contractual and safety reasons, and would often include the acquisition of further data to allow the next phase of the project to be planned in detail. A project manager is nominated who is responsible for the overall control of the project and normally for the control and safety of the plant, occasionally this responsibility remains with the existing plant operators if decommissioning is being carried out in a section of an operating plant.

INTERACTION WITH REGULATORS

Nuclear licensing arrangements in the United Kingdom include the requirement for the licensee to prepare decommissioning plans and programs. For Sellafield this has been developed into an agreed 15 year Post Operational, Waste Retrieval and Decommissioning program updated annually. For each decommissioning project however there is a more detailed interaction with the regulators which varies depending on the assessed level of risk within the project. The more difficult projects, which include the majority of the plutonium decommissioning, will require an overall safety case, detailed safety justifications for each phase of decommissioning and safety documentation for the construction and commissioning of any major facilities in support of the decommissioning. The overall safety case, which is not particularly detailed and is at a safety strategy level, and the first detailed phase submission would normally be required before work could commence. Further detailed phase submissions would be made at the appropriate time. For more simple projects it may be possible to cover all aspects in one document. Additional justification is required for waste generation and discharge authorizations. Particular attention is paid to projected dose uptake during the safety justification stage and formalized ALARP studies are implemented for most projects. A number of dose rate modelling techniques have been developed to assist the process and they allow rapid assessment of benefits achieved by additional shielding or removal of high sources. It is common for regulatory approval to be given in a number of stages

with agreed 'hold points' where performance to date can be reviewed. Frequent contact with the regulator 'Site Inspector' is normal.

PROJECT EXPERIENCE

Decommissioning has been undertaken at Sellafield since the mid 1950s, primarily to create space for new facilities, but since the start of the formal decommissioning program in the early 1980's six plants have been completely decommissioned, a further sixteen major project phases have been completed and eighteen are currently undergoing decommissioning. The following are brief descriptions from a few of the current projects:

First Storage Pond

Built in the early 1950s to receive, cool and decan the fuel from the original Windscale Piles, and later adapted to handle Magnox fuel from the Calder and Chapelcross reactors, the plant ceased operation in 1963. The plant consists of two open cooling ponds (Fig. 1) with an adjacent building housing twelve decanning bays and six withdrawal bays. The pond still contains some 190 skips containing fuel, isotopes, fuel hulls and a mixture of other wastes. Additionally there is a general accumulation of sludge and debris. Within the decanning bays there are over spill wastes from normal decanning operations including fuel rods, graphite and cladding together with residual material from experimental work for chemical decanning and provision of uranium 'pennies' for pilot reprocessing. Due to the period before decommissioning the building no longer achieved modern standards and following substantial stripping out new ventilation, environmental monitoring and cranage has been installed. Current operations center on the stripping out of all equipment from two decanning bays to allow the installation of the skip washing and sorting equipment. Most of the installation involves operations in up to 18' of contaminated water with extremely poor visibility. The sorting of the pond contents to allow appropriate downstream treatment and disposal is due to commence in 1996.

Fig. 1

First Separation and Head End Plant

Built for the dissolution and chemical separation of the Windscale Pile fuel, including separation of the Plutonium, Uranium, medium active and highly active waste streams, the plant is extremely large being over ten floors (60 meters) high and consisting of four highly active and two medium active cells. The plant continued in operation for the reprocessing of Magnox fuel until 1965 when it was replaced by the Magnox reprocessing plant. The north half was then washed out and permanently shut down but the south side underwent extensive alterations including the removal of metal fuel dissolvers, installation of shearing, dissolver, accountancy and maintenance cells to allow head end operations on oxide fuel utilizing part of the existing solvent extraction plant. Decommissioning of the plant poses a particular challenge due to the height of the cells, absence of in cell cranage, no designed access routes for equipment, varying radiological conditions, limited radiological data and absence of accurate as built drawings. A progressive approach is being taken to the decommissioning with the MAN cell, the least radiologically challenging, being dismantled first followed by the other cells, thus allowing the proving of techniques prior to deployment in the other cells. Typically for older plants the cell ventilation was inadequate by modern standards and a new fully filtered system has been installed to support all decommissioning operations. A waste handling facility (Fig. 2) has been constructed

adjacent to the MAN cell and incorporates automatic remote robotic size reduction, mainly using plasma arc, linked to an integrated control system and 3D modelling. This facility will handles the waste from all the cells. Access for the cell dismantling machine (Fig. 3) has been provided at high level and incorporates a manipulator system, deployed at the various levels in the cell, and a hoist to lower the cut components to the ground floor export link to the waste facility; wherever practicable standard components are used with the main development effort directed to special tooling and an integrated control system which links from a three dimensional model of the cell to the manipulator and size reduction robots and is aimed at minimizing operator fatigue and maximizing productivity.

Fig. 2

Fig. 3

Solvent Regeneration Plant

Built for regeneration of the Butex solvent utilized in the original reprocessing plant the plant is about 30 meters high and consists of six cells two of which were fitted out for the process, two were held in reserve, one was for general shielded R&D and one was for post irradiation examination of fuel. The solvent process performed better than expected and the two spare cells were released for other experimental work. Decommissioning of the first of the solvent regeneration cells is underway. There was no installed cell cranage but there were removable cell top concrete panels provided for initial construction. Optioneering concluded that top entry was the best approach and a size reduction facility incorporating cranage, tooling and access arrangements has been constructed and commissioned on top of cells 1 and 2. Following the provision of a filtered extract system it was possible to complete the removal of the residual inventory, particularly the Butex solvent which poses a significant fire hazard. Dismantling of cell 1 utilizing manual techniques has commenced though the hazard posed by the possible presence of solvent necessitates special precautions to be taken. It is anticipated that Cell 2 will be dismantled utilizing similar techniques but the later cells will require progressively more remote methods. Radiological data is limited for some areas.

Plutonium Purification Plant

Built to purify the plutonium stream from the reprocessing plant the plant is large, being effectively two mirror image cells, four stories high with a brick wall as the secondary containment, the vessels and pipework being the primary containment. The anticipated level of plant containment was not achieved and several leaks (all contained within the cells) and remedial works have led to a very high level of internal contamination. As the cell extract was not filtered the plant was a major contributor to the site aerial discharges and prior to allowing routine man entry for decommissioning it has been necessary to dismantle several feed system gloveboxes and install a new ventilation system, commissioned in late 1993. Following commissioning of the new ventilation system cell entries were possible and these confirmed the predicted extremely high levels of contamination and also discovered localized radiation sources in excess of 15 mSv. Intrusive surveys of the cell vessels is now in progress to assess the amount of residual liquor. Dismantling will be undertaken manually, moving components to a waste handling facility, currently being constructed on the South side of the plant, for size reduction and packaging. All external control, sampling and fuel cabinets

are being progressively removed. As with many plutonium facilities there is the potential risk of criticality and extensive in situ inventory monitoring will be undertaken prior to the movement of vessels to the size reduction facility.

Finishing Line 3

This was the main plutonium finishing line for metal and oxide production from 1963 until 1987. It is constructed in a series of over twenty gloveboxes linked by two conveyor systems and operated through a shielded face. The plant operating life and throughput greatly exceeded design expectations but has resulted in a significant hold up of material, particularly in the concentrate stock tanks, the conveyor systems and the furnaces. Contamination is extensive and radiation levels exceed 10 mSv in several areas. Because of the radiation levels it was intended to utilize remote methods but it quickly became apparent that the extremely difficult access (Fig. 4) and the criticality hazard associated with the high residual inventory would make this approach difficult and costly. Extensive optioneering, modelling and value engineering studies were then applied to optimize the dismantling strategy and in particular the choice of manual in preference to remote dismantling approaches. The manual size reduction is practicable provided boxes are removed from the line and the location chosen for the size reduction facility was that left vacant following the earlier dismantling of the adjacent Co-precipitation Plant (see 3.7). Installation of the size reduction facility and associated tooling is underway. The same method of size reduction will be employed as on the current First Floor dismantling with plasma arc as the preferred tool. Gloveboxes will be isolated and removed from the line to the size reduction facility where plasma cutting and other techniques will be utilized prior to inventory assessment and packing in 200 liter drums. As for B203 there is the need to avoid risk of criticality and again inventory assessment will be needed. In this case however the high background from adjacent plant items precludes high levels of accuracy and gloveboxes will undergo secondary assay once removed from the line. Because of the anticipated inventory additional monitoring of items ready for placement in waste drums will be undertaken to ensure the drum fissile material limit is not exceeded. These techniques are already in use on a number of other projects.

Fig. 4

Fast Reactor Fuel Facilities

The Co-precipitation and Dry Recovery plants associated with the conversion of recovered plutonium and uranium for the fast reactor program have already been totally decommissioned. These plants formed the basis for the development of many of the techniques and equipment needed for plutonium plant decommissioning including in situ inventory assay, containment and decontamination, size reduction, and recirculating suit showers with water treatment. The current operations center on the PFR Fuel Fabrication Facility and the associated Dry Granule Production Plant (DGPP) which provided the mixed oxide fuel granules. The PFR plant converted the granules into pellets which were then loaded into fuel pins and finally assembled into fuel assemblies for shipment to Dounreay. Decommissioning of the final assembly area and pin filling line was completed utilizing manual techniques. The next stage of the project involving removal of all other plant which can reasonably be achieved manually is currently underway. The final phase of the project, the fuel line where the pellets were prepared, is the most heavily contaminated

and will require remote dismantling. Following the earlier development work on Co-precipitation and Dry Recovery plants the PFR project can be termed production decommissioning with a large number of plant items to remove against tight financial and timescale targets. The DGPP plant was initially a pilot facility upgraded to full scale production. The result has been a significant decommissioning challenge in terms of residual inventory and dose rates combined with very restricted access. In addition to the techniques used on other projects it has been necessary to develop a manipulator system to allow remote dismantling of the main part of the plant. Following extensive off site development and training, utilizing full scale mock ups, the machine is now deployed on the plant, this will be moved to the PFR plant on completion of DGPP. Many of the systems are common with the machines being developed for other projects. In most cases the same control, viewing systems, end effect manipulators and tooling are used, the main differences being the deployment platforms.

Other Projects

The above projects represent a cross-section of the decommissioning activities being carried out by BNFL across the spectrum of facilities at Sellafield. Others include the well advanced decommissioning of the Windscale Piles Chimneys, the Magnox Reprocessing Plant Plutonium Corridors and Old Evaporator Cell, several other plutonium plants, R&D facilities and the waste recovery operation on Silos. The refurbishing of the Highly Active Dissolver Cell in the main Magnox reprocessing plant, while not a decommissioning project, involved the decontamination, removal and replacement of the fuel dissolver and associated equipment. The successful application of new and existing technology in this and other very challenging environments at Sellafield over many years has provided a very firm base of experience. This real experience has been a major factor in the success of the overall decommissioning program to date.

TECHNOLOGY AND DEVELOPMENT

The projects described depend on the use of a wide range of technologies and techniques to enable them to achieve their objectives. A substantial Research and Development program (2) was established in support of the Sellafield decommissioning projects in 1989 and has successfully provided important equipment and techniques. Examples of these include the in situ and fixed plutonium inventory assay equipment with significantly increased accuracy and the ability to cope with a variety of isotopic compositions, remote equipment backed up by a range of integrated control, modelling and viewing systems which minimize operator fatigue and improve effectiveness, data acquisition systems including three dimensional imaging, radiation modelling codes allowing the prediction of source data from a limited number of dose rate measurements and advanced decontamination methods which are extremely effective but with near zero discharges. The emphasis of the program in support of current projects is to continually increase effectiveness and reliability recognizing that decommissioning is a production scale operation. Whilst it has been demonstrated that the current decommissioning program can be achieved with technologies available today the development program also supports longer term developments, sometimes involving radical and emerging technologies, but all with the overall objective of reducing the cost of decommissioning in an environment when the constraints in the form of

regulations, dose targets and discharges are expected to become increasingly more restrictive.

WASTE MANAGEMENT

Successful waste management is integral with an ongoing decommissioning program. Waste must be produced in a form either ready for ultimate disposal or readily recoverable for further treatment for disposal. Decontamination, segregation, avoidance of cross and secondary contamination and adequate inventories and records are all important factors. Interim storage for Plutonium wastes from decommissioning has been provided cost effectively within part of a decommissioned plant and provisional agreement for a number of encapsulated waste forms has been reached with UK NIREX, the agency responsible for the provision of the UK disposal facility. Techniques for decontamination to be very low or free release level using near zero discharge techniques together with metal smelting are being developed to further reduce waste volumes.

DECOMMISSIONING COSTS AND ACHIEVEMENT

The technical achievement of projects, whilst in itself of satisfaction to the engineers, must be matched by the ability to predict and achieve financial targets. BNFL has since 1988 assessed long term decommissioning costs for all its facilities, shut down, operating, under construction or planned and this is updated on a regular basis. The ability to forecast such long term costs is required to allow for the financial provision for future liabilities both on a global and plant specific basis. BNFL has developed a detailed plant decommissioning costing model (1) which utilizes plant construction information, known or estimated radiological data to generate decommissioning costs, manpower and material requirements, decontamination effluent arisings, waste volumes and disposal container requirements. The reference data used in the calculations is based on currently available technology and techniques and where available real experience on decommissioning projects or other experimental or forecast data. The model has been used on twenty two major plants at Sellafield, including THORP where the provision for decommissioning is included in the cost of reprocessing. This long term forecast for all BNFL's liabilities has shown a downward trend, matched by a reduction in the parallel long term forecast of waste management costs. The reductions in the latter are very much due to the integrated site approach to the problem. For the current decommissioning projects up to April 1995 BNFL had completed some twenty two projects or major project phases and this has been achieved at 85% of the originally estimated cost. The current program encompasses some 18 decommissioning projects underway and the latest predicted out turn (December 1995) is $179M against an estimate of $206M.

CONCLUSIONS

The decommissioning of the Sellafield site presents an ongoing challenge requiring an integrated and coordinated program. The successful completion of a number of projects and the large number of projects currently undergoing practical decommissioning demonstrate that reprocessing plant decommissioning can be successfully and cost effectively accomplished.

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SHOREHAM: HIGHLIGHTS OF A SUCCESSFUL DECOMMISSIONING PROJECT

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ABSTRACT

In May 1995, the Shoreham Nuclear Power Station became the first large, NRC-licensed, commercial nuclear power plant to complete decommissioning and have its license terminated. The pioneering nature of the project made the Shoreham experience unique, interesting and challenging.

Planning for the decommissioning of the Shoreham plant began in 1989. Radiological characterization of the Shoreham facility and site was initiated in early 1990. Radiological surveys were performed on plant structures, systems, the reactor pressure vessel (RPV), RPV internals, the biological shield wall and the facility grounds.

The decommissioning alternative selected for Shoreham was DECON, i.e., prompt removal of radioactive materials to attain conditions for release of the facility for unrestricted use. Other alternatives, such as SAFSTOR and ENTOMB were evaluated but eliminated because of the desire to make the facility available for re-powering or other industrial uses as soon as possible.

Much of the Shoreham decommissioning work was accomplished through dismantlement and removal of equipment. Field tested and proven processes were used to dismantle plant structures and systems. High pressure water lancing (hydrolyzing) and a variety of mechanical decontamination techniques were used during the course of the project.

An extensive final radiological survey (termination survey) was performed to demonstrate that the residual radioactivity levels had been reduced in accordance with Shoreham release criteria specified in the NRC-approved Decommissioning Plan and the Termination Survey Plan.

The cost and schedule aspects of various decommissioning activities were closely monitored and controlled. As a result, the project was successfully completed on schedule and under budget (about $186 million).

In April 1995, the NRC issued a License Termination Order terminating the Shoreham Nuclear Power Station's license and authorizing the unrestricted release of the Shoreham facility. This brought to a successful conclusion the seven-year long Shoreham decommissioning effort.

BACKGROUND

The Shoreham Nuclear Power Station was located on the north shore of Long Island (Town of Brookhaven, Suffolk County, New York) approximately 50 miles east of New York City.

In April 1965, Long Island Lighting Company (LILCO) made public its plans to build the Shoreham plant. In May 1968, LILCO filed an application with the United States Nuclear Regulatory Commission (NRC)--then the United

States Atomic Energy Commission -- for a permit to build the Shoreham plant. LILCO received a Construction Permit in April 1973.

LILCO applied for an Operating License in September 1975. In December 1984, the NRC issued a license authorizing fuel loading and cold criticality testing. This was followed by a low power license, issued by the NRC in July 1985, authorizing testing at power levels below five percent of rated power. The Shoreham plant operated intermittently at low power levels from July 1985 to June 1987, generating total gross thermal energy of only about 120,000 MWH. At the time of the plant's final shutdown in June 1987, the burnup of the fuel was limited to two effective full power days or 48 Megawatt Days/Metric Ton Uranium (MWD/MTU) compared to approximately 38,000 MWD/MTU for a typical BWR core which operated for its full cycle.

In February 1989, the State of New York and LILCO entered into an agreement under which LILCO agreed not to operate the Shoreham plant and to transfer the plant and certain areas and buildings on the Shoreham site to a newly created State agency, the Long Island Power Authority (LIPA) for decommissioning. The agreement was subsequently approved by the Board of Directors of LILCO, the Board of Trustees of LIPA, and the New York State Public Service Commission. The agreement became effective in June 1989 when LILCO's shareholders voted to approve it.

Although the NRC granted a full power license in April 1989, LILCO never operated the Shoreham plant after the previous five percent power testing period. In fact, LILCO began defueling the Shoreham plant shortly after LILCO's shareholders voted to approve the agreement.

Fuel removal from the reactor to the spent fuel storage pool was completed in August 1989, and by NRC Confirmatory Order issued in March 1990, the Shoreham license was modified such that fuel could not be reloaded in the reactor without prior NRC approval. The license was formally amended to a Possession-Only License (POL) status in July 1991. The transfer of this POL from LILCO to LIPA subsequently became effective in February 1992.

In preparation for decommissioning the Shoreham plant, LIPA entered into a Management Services Agreement with the New York Power Authority (NYPA) under which NYPA was contracted to provide technical and management services for the decommissioning of the Shoreham plant. NYPA in turn hired a number of consultants and contractors to carry out the planning, licensing and implementation of the decommissioning project, and also established a composite project organization utilizing LIPA, NYPA, LILCO and contractor personnel.

DECOMMISSIONING ALTERNATIVES

The current NRC regulations define decommissioning as the removal of a nuclear facility safely from service and the reduction of residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license (1,2).

Several alternative methods of decommissioning are available. These alternatives are: DECON, SAFSTOR and ENTOMB. The terms DECON, SAFSTORE, and ENTOMB are defined as follows:

DECON is the alternative in which the equipment, structures, and portions of the facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of operations.

SAFSTOR is the alternative in which the nuclear facility is placed and maintained in a condition that allows the nuclear facility to be safely

stored and subsequently decontaminated (deferred decontamination) to levels that permit release for unrestricted use.

ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete; the entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting release of the property for unrestricted use.

The NRC regulations allow all three alternative approaches to decommissioning, although the ENTOMB alternative is endorsed only with some reservations.

SELECTION OF DECON

The contamination and activation levels were low at Shoreham because of its limited operation. Based on these limited contamination and activation levels, it was advantageous to proceed with the DECON (immediate decontamination and dismantlement) decommissioning alternate. There were several other reasons for the selection of the DECON decommissioning alternative; maximum flexibility in selection of future near-term use of the site; availability of personnel who were knowledgeable about the facility and its operating history; the ability to decommission the facility without significant occupational radiation exposure; the elimination of the need for long-term monitoring, surveillance, and maintenance; predictability of low level radioactive waste disposal options and costs; and the fact that DECON alternative would cause no significant environmental impact.

SHOREHAM DECOMMISSIONING PLAN

In December 1990, LIPA submitted the Shoreham Decommissioning Plan (3) to the NRC. The Decommissioning Plan was prepared in accordance with the requirements of Title 10, Code of Federal Regulations (10 CFR) 50.82. Draft Regulatory Guide DG 1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors" was used for guidance in preparing the Decommissioning Plan.

The Shoreham Decommissioning Plan was accompanied by a "Supplement to Environmental Report (Decommissioning)" which was prepared in accordance with 10 CFR 51.53(b) (4). The Supplement reflected earlier environmental analyses prepared by LIPA to comply with the requirements of the New York State Environmental Quality Review Act (5).

In June 1992, after two rounds of questions and responses, numerous discussions and complex legal maneuvers, the NRC issued an order approving the Shoreham Decommissioning Plan and authorizing the decommissioning of the facility (6).

SITE CHARACTERIZATION

Radiological characterization for decommissioning of the Shoreham site was performed in stages during the project. The primary goal of the initial effort was to identify those areas and plant systems which were above the proposed release criteria. This information would define the scope of decontamination and dismantlement activities. The results of the initial site characterization effort were reported by LILCO in May 1990 (7) to support preparation of the Shoreham Decommissioning Plan. It consisted of three main efforts or tasks: 1) determine the extent of areas and plant systems with surface contamination levels above 5000 dpm/100cm2 total or above 1000 dpm/100cm2 removable, 2) identify components and structures in the vicinity of the reactor core with neutron induced activity that exceeded either the above surface contamination limits or the proposed gamma exposure rate limit of 5R/hr

at one meter and 3) to identify the principal radionuclides present in surface contamination deposits and in activated in-situ materials.

The site characterization was performed under the administrative control of a specially prepared Site Characterization Program. The first task was performed using the existing Shoreham survey instruments and radiological survey procedures, supplemented by a written work instruction for field surveys. Survey design and survey methods largely followed the NRC guidance for final (termination) surveys available at that time. The second task consisted of neutron activation analysis of the reactor pressure vessel (RPV), RPV intervals, and the biological shield (bioshield) wall. This task was preformed using the ORIGEN2 computer code supplemented by analytical calculations to estimate isotopic compositions. Dose rates from activated components were determined in the initial characterization effort using the Microshield code. The third task was to identify the radioactive composition in areas of the facility where neutron activated materials were the most likely source of residual contamination. This was performed using the neutron activation analysis results supplemented by radiochemistry analysis of core samples.

Several factors prevented complete characterization at the time of the initial site characterization. These were 1) the presence of fuel in the pool prevented physical access to the Fuel Pool and associated plant systems; 2) the NRC operating license and associated Technical Specifications were in effect, limiting destructive removal of materials to gain information; 3) radiological "interferences" were present in the vicinity of the RPV which prevented accurate surveys of gamma exposure rate from the bioshield wall; and 4) the exact requirements of the final site release criteria were not known.

The initial site characterization effort identified 12 areas (nine plant systems and three structural areas) containing levels above Regulatory Guide 1.86 (8) limits. These were the following:

- Reactor Water Recirculation System
- Reactor Water Cleanup System
- Control Rod Drive System
- Fuel Pool Cleanup System
- Residual Heat Removal System
- Condensate Demineralizer System
- Core Spray System
- Sampling System
- Liquid Radwaste System (partial)
- Dryer Separator Pit (structure)
- Reactor Cavity (structure)
- Radwaste Laydown Area (structure)

Initial estimates of decontamination and dismantlement work scope focused on the nine systems identified above. Ultimately, an additional 15 systems and 13 structures were identified which contained levels above the release criteria. The resulting major additions to the decommissioning work scope included portions of the nuclear boiler system, major portions of the liquid radioactive waste system and the bioshield wall.

DECONTAMINATION AND DISMANTLEMENT

Radioactive materials were removed from the Shoreham facility by two general methods: dismantlement and removal; and decontamination in place. Dismantlement and decontamination activities were confined primarily to the Reactor and Radwaste Buildings, with minor activities in the Turbine

Building. Activated portions of the bioshield wall were removed. Systems and equipment removed included the RPV (except for the bottom bowl) and major portions of the plant systems that were characterized as being radioactively contaminated. Control rods, fuel channels and fuel storage racks were removed intact and shipped directly to Barnwell, South Carolina, for burial. About 25 percent of the piping and equipment that was on site before decommissioning began was removed. The majority of contaminated piping and equipment removed from the facility (more than four million pounds) was sent to an off-site vendor for volume reduction. This resulted in a significant reduction in waste burial volume, from a pre-decommissioning estimate of over 80,000 ft3 (no volume reduction assumed) to 8,350 ft3. The removal of radioactive materials was accomplished through 386 well managed, flawless shipments.

The large majority of Shoreham decommissioning work was accomplished through dismantlement and removal of equipment. Industry accepted and field proven processes were used to dismantle plant structures and systems. Techniques ranged from simple, manually operated power band saws used to cut small bore piping to more sophisticated techniques such as pipe mounted cutting machines which were used to mechanically cut large bore piping. The selected options reflected consideration of the radiological conditions associated with their application.

Many areas, equipment and piping were successfully decontaminated and left in place. This included 15,000 ft. of embedded radioactive waste system drain piping, determined to be slightly contaminated over much of its length. This piping was decontaminated in place using high pressure water. Major structures that were successfully decontaminated included the suppression pool, reactor cavity, spent fuel pool and dryer-separator pit. Decontamination techniques were consistent with those routinely used in the nuclear industry. In-situ high pressure water lancing, and a variety of mechanical decontamination techniques were used.

Segmentation of the Reactor Pressure Vessel and Internals

The RPV and internals were decontaminated to the extent practical while in place by water flushing, and were segmented, packaged and shipped for use by others, or for off-site disposal. The bottom bowl of the RPV was decontaminated by grinding the internal surfaces to remove surface contamination. Enclosed cutting stations with filtered exhausts were for the cutting of contaminated components in order to prevent the spread of contamination.

Segmentation of the neutron activated reactor internals was performed using underwater, semi-automatic plasma arc cutting equipment. The RPV was segmented into shell sections using a platform mounted rotary mechanical cutting machine from inside of the RPV. The shell sections were then cut using oxy-fuel into pieces appropriately sized to permit safe and efficient handling, packaging and shipping. Highly radioactive materials for disposal were loaded into liners for transportation in shielded shipping casks.

Guiding Principles

Certain guiding principles were very effective in controlling overall project cost and schedule. For example, it was recognized early in the project that it was difficult (and costly) to perform radiological surveys of piping and equipment to demonstrate compliance with release criteria. This led to the general practice of removing contaminated systems for volume reduction and burial rather than attempting to

decontaminate and survey them in place. This philosophy was succinctly phrased as: "when in doubt - cut it out."

The second guiding principle was to perform waste processing work off-site whenever possible. Volume reduction (compaction, incineration and metal melting) were very effective in reducing the volume of waste shipped for burial to the low level radioactive waste disposal facility. Radioactive material processing at Shoreham was limited to segmentation of large components for ease of handling and shipping. Volume reduction was effective in reducing the burial volume to about one-tenth of the original estimate. Additionally, off-site waste processing minimized the potential for schedule delays, competition for available work spaces and resources and the likelihood of cross-contamination of clean areas.

RADIATION PROTECTION

The total radiation exposure for the entire project was 3.2 Person-Rem (compared to the initial projection of 187 Person Rem), a significant achievement for a project which employed about 1000 decommissioning workers at its peak.

The radiological source terms at the Shoreham were relatively low. The total Curie content in the facility, excluding irradiated fuel, was about 600 Curies with maximum dose rates of about 200 R/hr found on RPV internals. Such low levels, however, did not preclude a necessary attentiveness to minimizing personnel radiation exposures.

The reduction in radiation exposure was the result of an aggressive "dry run" policy, a strong management commitment and an effective ALARA suggestion program which provided recognition and monetary incentives. Additional techniques and features used to implement ALARA during decommissioning activities included the following:

Flooding of the RPV prior to cutting and removing vessel internals with water level controlled to provide optimum shielding benefits.

Extensive underwater cutting of components. Submerged lighting and television cameras provided continuous surveillance of underwater cutting operations.

Installation of an elevator platform in the RPV for raising and lowering cutting tools and removal of cut components from the pressure vessel.

Utilization of temporary indices on walls, cables and beams to assist crane operators achieve proper azimuthal and elevation alignment for disposition of components in transit.

Installation of a floating fume collection hood with HEPA filtered exhaust over the wet cutting station to preclude the escape of potential airborne contamination during plasma arc cutting.

Rigorously enforced policy of clearing personnel from areas when highly radioactive sources ("hot picks") were moved or transported.

An example of the success of the Shoreham ALARA program is illustrated by the task of packaging and shipping 137 activated Control Rod Blades (CRB). This task involved 35 "hot picks" of liners holding four CRBs each. The liners had dose rates of about 25-50 R/hr on contact. Each liner was packaged underwater and removed from the spent fuel pool and placed in specially made shielded transport boxes. In spite of the high dose rates involved and the extent of physical handling the total exposure was limited to less than 0.6 Person-Rem.

FUEL DISPOSAL

The NRC regulations do not treat spent fuel disposal as a decommissioning activity, nor is the cost of fuel removal and disposal included in the

decommissioning cost estimate. Fuel deposal is considered to be an operational activity. It is, however, necessary to remove the fuel from the spent fuel storage pool before the pool and the associated systems can be decontaminated and the facility released for unrestricted use. Therefore, the disposal of the slightly irradiated Shoreham fuel became critical to the success of the decommissioning project.

After developing several other disposal options, including reprocessing in Europe and on-site dry storage, LIPA reached an agreement with Philadelphia Electric Company (PECo) to transfer the slightly used fuel to PECo's Limerick plant for reuse. The fuel shipments got underway in September 1993 after a series of unsuccessful challenges, including legal actions by the State of New Jersey reaching all the way to the United States Supreme Court. New Jersey attempted to block the passage of the barges carrying the Shoreham fuel through its coastal waters. The fuel shipments were completed three months ahead of schedule in June 1994.

FINAL TERMINATION SURVEY

To prepare for the termination survey, a detailed final survey plan (9) was submitted to and approved by the NRC. The Plan was based on the guidance provided in NUREG/CR-5849 (10). For the conduct of the survey, the facility was divided into 385 separate entities called "survey units." These included areas within the facility structures, outside areas and plant systems. Within each survey unit an average of 265 locations were selected for measurements. A total of approximately 230,000 measurements of direct and removable surface activity and gamma exposure rate were taken in the survey. Approximately 105 samples of soil and bulk materials were also collected and analyzed. The survey covered all the buildings, plant systems and site grounds within the secured area fence, about 20 acres.

The termination survey was conducted in parallel with decommissioning activities, and was carried out in four major phases. A Termination Survey Group was established within the decommissioning project organization, which had at its peak 74 people including 40 survey technicians. Survey measurements began in January 1993. A site back out plan was developed to allow the survey to proceed in parallel with fuel shipments and decommissioning activities. Innovative survey techniques and tools (e.g., pipe crawlers fitted with instruments for embedded piping surveys) were developed and employed. The cost of the survey was approximately $12.3 million.

The Final Termination Survey Report (11) submitted to the NRC demonstrated that all 385 survey units (and hence the facility) satisfied the release criteria. The confirmatory surveys performed by the NRC verification contractor, the Oak Ridge Institute for Science and Education (ORISE), concluded that Shoreham was ready for unrestricted release.

CONCLUSION

As the Shoreham decommissioning project was nearing completion, two public meetings were organized (December 1994 and March 1995) to inform the public about the Shoreham decommissioning activities and the results of the final radiological surveys. Although public participation at these meetings was somewhat limited, the meetings received wide coverage in the media.

The order terminating the Shoreham Operating License (12) was issued in April 1995. After allowing 20 days for the public to voice any opposition

and file a request for a hearing (none was received), the License Termination Order became effective in May 1995.

The lessons learned and insights gained from the Shoreham experience have led to proposals for better decommissioning regulations and guidelines from the NRC, particularly as they apply to prematurely shut down plants (13,14).

The successful completion of the Shoreham project was an important step in demonstrating that decommissioning of large nuclear power plants can be achieved at predictable cost and schedule. It holds valuable lessons in a number of areas including decommissioning planning, project management, decontamination and dismantling methods, control of radiation exposure, termination survey techniques and spent fuel disposal.

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50-7

DECOMMISSIONING PLANNING ACTIVITIES FOR TWO RUSSIAN NAVY LAND BASED NUCLEAR SUBMARINES AT PALDISKI, ESTONIA

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ABSTRACT

Paldiski, Estonia had been the site of an important nuclear submarine training facility for the USSR Navy. The facility included two scaled submarine mock-ups, one delta and one echo class, each containing an operational nuclear reactor. Construction of the facility began in the early 1960's with the first training submarine (Unit #1) going critical in 1968 with a 70 MWt reactor. A second training submarine (Unit #2) was added in the early 1980's and went critical in 1983 with a 90 MWt reactor. Unit #1 was refuelled and retrofitted with upgraded steam generators in 1980. The two reactors were operated into 1989. Both training submarines were housed within a single building in a common high bay area. Auxiliary site facilities included a liquid waste processing facility; storage buildings for solid and liquid radioactive waste; a central facility ventilation center; cooling towers; a cooling water pump facility; a central heating plant; a radioactive laundry facility; and a radiochemical laboratory. In 1994, as part of the negotiated Russian troop withdraw from Estonia, Russia agreed to defuel and safe store the reactors prior to 30 September 1995, when control of the site was transferred to Estonia. The spent fuel from the reactors was transferred under Russian control to Russia in October 1994. Russian preparation for safe storage of the two reactors included dismantling non-active components, systems related to reactor operation, and some of the associated auxiliary facilities. Russia also constructed two concrete sarcophagi around the remaining hull sections containing the reactor vessels. This work was completed prior to mid-September 1995.

An international expert group was established in early 1994 at the request of the Estonian government. This group is known as the Paldiski International Expert Reference Group (PIERG). The purpose of PIERG is to promote the safe and timely decommissioning of the Paldiski nuclear facility by advising Estonia on technical, legal, organizational, financial, waste management, and radiation safety matters related to the decommissioning activities. Utilizing the resources of PIERG, a

Conceptual Decommissioning Plan has been developed for the site. This plan has been reviewed by the IAEA and Estonia and accepted for implementing as funding permits. The Conceptual Decommissioning Plan is now the basis for ongoing work and planning of future site activities.

BACKGROUND

Paldiski is located approximately 50 km south-west of Tallinn, capital of Estonia. Paldiski had been an important military base and submarine harbor for the former USSR since 1939. In the early 1960's construction began on a land-based training center for nuclear submarine crews of the Soviet Navy. In 1968 the facility with a 70 MWt training nuclear reactor (Unit #1) was commissioned, Unit #2, rated at 90 MWt, went critical in 1983. Unit #1 was refuelled once in 1980. The reactors with all shipboard auxiliary systems were situated in scaled submarine hulls located in the high bay area of the Main Technological Building. Both reactors were shut down in 1989.

After Estonian re-proclamation of independence in 1991, the withdrawal of Russian troops from Estonia territory, closure of the Paldiski training center, and decommissioning of the reactors became a subject of intense negotiations between Estonia and Russia. Ultimately, an agreement on the decommissioning of the Paldiski Training Center was signed in July 1994. According to this agreement, Russian experts removed non-radioactive and classified components of the facility and prepared the site for turnover. The Russian Federation officially transferred ownership and control of the site to the Republic of Estonia on 26 September 1995. The agreement (1) on transition of the Paldiski facility has the nature of a political treaty and technical aspects of the agreement are non-existent or leave considerable margins of interpretation to concerned parties. This had caused difficulties with the planning of decommissioning work as control of the site remained with Russian authorities who's decommissioning activities were performed with limited involvement or supervision by international or Estonian observers.

Normally, the operator is responsible for all aspects of the safe decommissioning of a reactor including preparation of the decommissioning plan and its submission to the regulatory body for approval. In the case of Paldiski, the situation is unique as the reactors were owned and operated by an organization in one country while located in another. The consequence of this has been that the turnover of the site and related documentation has been hampered by multi-national politics.

SITE DESCRIPTION

The site occupies approximately 22 hectares (~9 acres) and consists of the Main Technological Building (MTB) and 8 relevant auxiliary facilities which are: a Liquid Waste Treatment Facility (LWTF), a Liquid Radioactive Waste Storage (LWS), a Solid Radioactive Waste Storage (SWS), a Ventilation Facility, a Laundry Facility, a Radiochemistry Laboratory, Cooling Towers, and a Boiler Facility. The site plan is presented in Fig. 1.

Fig. 1

SITE STATUS, PHYSICAL AND RADIOLOGICAL DESCRIPTION OF THE SITE

Main Technological Building (MTB)

The MTB consists of the high bay area containing Units #1 and #2, two spent fuel storage pools (only one used), processing rooms for primary and secondary coolant water, and an annex of offices. The operational data of the reactors is presented in Table I.

Table I

The reactor fuel was removed and shipped to Russia in October 1994. As part of preparing the site for transition, Russia removed non-contaminated and secret equipment and dismantled the submarine hulls except the sections associated with the reactor vessels. Cast concrete sarcophagi were built by Russia around these remaining hull sections. A preliminary radiological characterization by a US Department of Energy team in 1995 (2) determined that the radiological situation of the MTB is reasonably good with only limited areas of contamination e.g. a spent fuel storage pool and some associated rooms.

Liquid Waste Treatment Facility (LWTF) and the Liquid Waste Storage (LWS)

Liquid radioactive waste is stored in the tanks of the LWTF and LWS facilities. The LWTF contains the equipment used for treatment of the contaminated waste water generated on site. Equipment includes evaporators, flocculation tanks, ion exchange columns, six receiving tanks (total capacity 1020 m3) and two tanks for holding water prior to discharge.

The LWS consists of six tanks (total capacity 2400 m3). These tanks were originally intended as a final depository for processed/concentrated liquid radioactive waste. Four of the six tank at the LWS are now empty and have been decontaminated. The remaining known liquid radioactive waste on site is presented in Table II.

Table II

Radioactive contaminated areas and systems were observed during a preliminary survey of the LWTF. The highest radiation readings recorded were in the vicinity of the ion exchange columns. Detection of slightly contaminated water in pipe chase areas in the basements of the LWTF and LWS indicate leakage from the tanks or plumbing. This will only increase the volume of waste to be decontaminated because in addition to the liquid waste there is now increased areas of contaminated concrete. Recently, 14 additional liquid radioactive waste holding tanks were discovered on site. At present, no information is available on their contents.

The Solid Waste Store (SWS)

The SWS consists of a concrete structure divided into 10 compartments. The top of the storage consists of concrete slabs, or poured concrete with removable plugs all of which have been covered with a layer of crushed gravel and asphalt. This facility was originally intended as a final repository for the radioactive operational waste from the site. The Russian Navy has stated that only three compartments were used and described the contents of the three compartments as:

No 1 - 8 steam generators and a circulation pump from Unit # 1 replaced during the refuelling operation in 1980, contaminated equipment, and miscellaneous contaminated material.

No 4a - Protective clothing and sealed beta sources

No 5 - 20 irradiated control rods, high level radiation sources, contaminated equipment, and miscellaneous contaminated material.

All the waste in the SWS has been disposed without any conditioning and packaging. The Russian Navy has stated that the total activity of the waste in the SWS is about 3.7 TBq.

Laboratory Building

The laboratory building is a three floor building containing offices, health physics training rooms, environmental and radiochemical laboratories, and instrument calibration facilities. Preliminary

radiological characterization by a US Department of Energy team in 1995 indicate that contamination is limited and very low level.

Laundry Facility

The laundry facility is a two floor building that processed all site laundry. The upper floor was reserved for non-contaminated clothing and the ground level for contaminated protective clothing. Prior to site turnover, Russian personnel performed a release survey of the facility subsequently releasing all equipment and piping as scrap metal. Results of surveys performed in the emptied building in 1995 by a US DOE team appear to support the Russian survey data.

Ventilation Facility

The ventilation facility provided off-gas capability to all reactor operation facilities on site. Connected were the Main Technological Building, the Laundry Facility, the Liquid Waste Treatment Facility, and the Laboratory Building. All building off-gas passed through a single stage HEPA filter before discharge up a 100 meter stack. US and Russian survey data of the HEPA filter enclosures do not indicate radiation levels above background. Russian personnel reported that the filters had never been changed during the entire operating life of the facility and no records exist indicating filter integrity testing. It is believed that the filters are either completely intact or have long since disintegrated. Investigations are planned in the future. Work areas within the facility are not contaminated.

Vehicle Decontamination Garage

This facility is only mentioned here. It was designed for vehicle decontamination in the event of a nuclear accident. It was never used for any decontamination purposes serving instead as a vehicle maintenance garage and for supply storage.

INTERNATIONAL CO-OPERATION, THE PALDISKI INTERNATIONAL EXPERT REFERENCE GROUP - PIERG

Preparations for transfer the Paldiski site from Russian to Estonian control revealed that Estonia lacked the necessary technical and financial resources to adequately manage and safely decommission the Paldiski facility. To properly manage the facility, in the near term, international assistance would be necessary.

Even at the moment of regaining independence, Estonia recognized that it would ultimately inherit the Paldiski facility. Realizing that in-country nuclear expertise was non-existent, the Estonian government initiated an international campaign aimed at garnering assistance and support for the management and decommissioning of the Paldiski site. The IAEA and numerous Nordic region countries and organizations were approached for assistance. At the initiative of the Swedish Minister of Foreign Affairs, and after consultations with other concerned countries, a meeting was held in Stockholm in January 1994 which was the beginning of an active international engagement on the issue of decommissioning the Paldiski facility. At a later meeting the Paldiski International Expert Reference Group (PIERG), was established with participation of Estonia, Finland, Germany, Russian Federation, Sweden, USA, IAEA and CEC. Later Denmark and France joined the group.

The objective of PIERG is to promote the safe and timely decommissioning of the former Soviet Union Nuclear Training Center at Paldiski by advising and assisting the parties participating in the decommissioning work on technical, legal, organizational, financial, waste management and radiation protection matters. Part of PIERG's charter is to appoint task

groups to deal with a specific project or technical issue. These task groups currently consist of:

Group Task

- C1 Future Uses for the Paldiski Site
- C2 Conceptual Decommissioning Plan
- C3 Site Radiological Characterization Plan
- C4 Conditioning of Liquid Radioactive Waste
- C5 Radiological Worker Training
- C6 Control Rods and Solid Waste Packaging Options
- C7 Workshop on Radioactive Metal Recycling
- C8 Site Plan and 3D-CAD Modelling
- C9 Comprehensive Site Health Physics Program
- C10 Site Management Plan
- C11 Conditioning of Remaining Liquid and Semi-solid Waste in LWTF and LWS
- C12 Interim Waste Storage Facility Development
- C13 Conceptual Development of an Estonian Final Repository

The C1 Task Group produced a report which was presented and accepted by PIERG. The C2 Task Group produced a Conceptual Decommissioning Plan (3) which was peer reviewed by the IAEA (4) and accepted by Estonia and PIERG for implementation. The Conceptual Decommissioning Plan is a dynamic document and is continuously being updated as new information and resources become available.

The Finnish government using the services of IVO International Ltd., Finland, took the lead for the C4 Task Group and began treatment of liquid radioactive waste in January 1995 (5) using their transferable liquid waste treatment system NURES (Nuclide Removal System). By August 23, 1995 when this project was completed over 760 m3 representing 80% of all radioactive liquid waste on site had been processed. In preparation for the next phase, IVO has carried out characterization of the tank sludges and made preliminary cost estimates for the sludge conditioning.

The Swedish government, coordinated through the Swedish Radiation Protection Institute and using the services of SKB and Studsvik, two Swedish nuclear companies, has taken the lead in the C1, C2, C7, C8 and C10 - C13 Task Groups. These activities include funding and leadership of PIERG; engineering plans for development of an interim waste storage facility utilizing an existing site building; a hands on workshop on materials recycling; development of an accurate site plan and 3D model; and most recently, an offer to collaborate with the US on characterizing the waste in the SWS and to package the waste later this summer.

The President of the United States committed the U.S. Department of Energy to participate in a technical co-operation program with the Republic of Estonia. The scope of the co-operative program included a technical site assessment and establishment of a technical agreement with the Estonians on decommissioning facilities. The Department of Energy's action plan was developed around the President's commitment and with the goal of empowering the Estonian government to safely and effectively manage and decommission the Paldiski site with a minimum of foreign assistance. A primary goal of DoE's involvement has been to demonstrate and transfer US technology. The US has the lead role in the C3, C5, and C9 Task Groups and is participating in the C2, C6, and C11 Task Groups. Several US technical teams visited Estonia during 1995 and performed a site characterization to determine the location and extent of radiological and other hazardous material contamination. The teams were

composed of personnel from the Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), Remote Sensing Laboratory (RLS) and Grand Junction Project Office (GJPO) have had extensive involvement. The teams have obtained direct radiation measurements in addition to performing a limited sampling program; conducted a limited structural analysis on the buildings; and evaluated the presence of non-nuclear hazardous materials on site. An airborne multi-spectral and radiological overflight was performed to map the status of the site and the peninsula on which it is located. Airborne data was acquired that will permit an environmental impact analysis of the site and surrounding peninsula from hazardous material insults. These data will be processed in early 1996. A specialized radiation and hazardous material training course was provided for the Estonians that would be working at the Paldiski site. Participating in the training course through a co-operative agreement with the IAEA were members of Latvian and Lithuanian radiation safety organizations. At the request of the Estonian government, DOE provided a resident technical consultant from B&W Nuclear Environmental Services, Inc. In addition to providing technical assistance on radiation safety, radioactive waste management, and facility management, the resident technical consultant functions as a local coordinator for DoE's initiatives ensuring that logistical requirements are handled with a minimum of delay and that cultural differences are respected.

In early 1996 a DOE team will perform a specialized radiological characterization of the site solid waste storage facility. This facility contains 10 concrete bunkers of which 3 are known to have been used. Un-inventoried waste includes unpackaged irradiated control rods, steam generators, industrial radiography and high level calibration sources, and other mixed radioactive wastes. Utilizing specialized gamma imaging equipment the team will generate visual and radiological images of the waste as well as spectrally analyze the waste. It is also planned to radiologically image the reactor compartment hull sections. This work will be co-operatively performed with Sweden. Discussion are underway with Finland to co-operatively evaluate the remaining radioactive liquids and tank bottoms.

ESTONIAN DECOMMISSIONING ACTIVITIES

As stated earlier, Estonia had no in-country expertise to utilize for the management and decommissioning necessary at the Paldiski site. In 1995, in conjunction with international assistance, the Estonian government established a wholly owned company, AS ALARA, Ltd, for the purpose of managing the Paldiski site and associated decommissioning activities as well as being responsible for all radioactive waste within the country. In addition to Paldiski, Estonian organizations generate radioactive waste from industrial, medical, and research activities. ALARA's staffing is currently at 15. There is a Manager of Technical Operations and four health physics specialists as well as staff maintenance and security personnel. The President of ALARA has over 40 years management experience in the North American energy industry at nuclear and non-nuclear facilities. ALARA management has decided that staffing will be maintained at a custodial level with contract personnel being utilized as needed. The technical consultant provided by the US DOE is working with ALARA personnel to develop a comprehensive site health physics program and provide training in survey and decontamination techniques. As ALARA's

staff gains expertise it is envisioned that ALARA will offer its technical expertise throughout the Baltic region.

THE CONCEPTUAL PALDISKI DECOMMISSIONING PLAN

To assist the Estonian government with the planning of decommissioning activities at the site a PIERG task group (C2) prepared a conceptual decommissioning plan. The Conceptual Decommission Plan took into account technical and non-technical conditions/constraints which included:

- site control by Russia until 30 September 1995;
- ongoing international co-operation (PIERG);
- absence of a clear Estonian policy on decommissioning and waste management and lack of relevant legislation;
- shortage of technical and financial resources and significant uncertainties on when these resources will become available (e.g. final repository for long-lived decommissioning waste).

Due to these factors, decommissioning based on immediate decommissioning and unrestricted site release was not realistic. As a result of the submarine reactor compartments being sealed within sarcophagi other decommissioning alternatives for the reactor vessels were not elaborated on by the task group. The Task Group C2 considered that from a technical and radiological point of view decommissioning activities should be initiated without delay as resources permit.

An aspect of designing and construction of the reactor enclosures should be that eventual dismantlement and decommissioning should be facilitated. At Paldiski, the reactor sarcophagi were designed and constructed by the Russians without co-ordination with Estonia. The approach followed by the Russians appears to maximize long term integrity rather than facilitate eventual decommissioning.

An element of developing a safe enclosure strategy is that physical/radiological surveillance and maintenance requirements are minimized. This requires that the extent of active areas be minimized and a passive configuration reached. This was addressed in the Paldiski Conceptual Decommissioning Plan. At the completion of decommissioning activities for safe storage all residual radioactivity will be confined in a stable form at the Paldiski site. A safe storage strategy requires a robust, durable structure to safely contain radioactive materials for long periods. The MTB appears to offer suitable characteristics for long term storage of the reactors and radioactive wastes. Investigations are underway to establish the suitability of converting the Main Technological Building into a long term monitored waste storage facility. This will solve the problem of where to store the radioactive waste generated as the contaminated buildings on site are decommissioned. In addition,this will provide ALARA a facility for waste storage that arises from other in-country generators.

The Conceptual Decommissioning Plan proposes a decommissioning strategy for the site to be implemented by Estonia. Work to improve and further refine the decommissioning plan continues as resources and relevant information become available. The Conceptual Decommissioning Plan has therefore served to provide a good starting point for development of a Site Management Plan and a Detailed Decommissioning Plan.

The highest priority of the decommissioning plan is the prevention of the spread of radioactive, non-radioactive, and hazardous material to the environment as well as the minimization of waste volume. Treatment of non-radioactive wastes will be coordinated with other environmental restoration projects underway on the Pakri peninsula where Paldiski is

located. Site specific procedures are being established for the treatment, packaging, and storage of the radioactive materials. A radioactive waste management plan for wastes arising from decommissioning activities at the site has been developed and is shown in Fig. 2. The final waste management plan will be approved in accordance with regulatory requirements imposed by the Estonian regulatory body. At present, Estonia has no national radiation regulations. Draft regulations are currently under considered by the Government for passing into law.

Fig. 2

The Conceptual Decommissioning Plan proposes that decommissioning work be split into 12 primary work packages:

- Infrastructure and documentation
- Radiological characterization and survey of contaminated systems, equipment, structures, buildings and the site grounds.
- Characterization and solidification of all tank liquids and sludges in tanks in the LWS and the LWTF
- Converting the MTB into an intermediate waste store
- Laboratory
- Laundry
- Ventilation Center and the Stack
- Decommissioning of the LWS
- Decommissioning of the LWTF
- Active pipes and ventilation ducts in the ground
- SWS
- Restoration of the site

A Site Management Plan is being developed which divides these activities into short, medium and long term actions. The short term action program during 1996 - 97 covers decommissioning and restoration work that are urgently required from radiological and environmental point of view. More extensive and costly work packages are envisaged during the medium term period from 1998 and continue 3 -5 years. The long term action program will carry out daily operations and supervision of the facility. The configuration of the site after the medium term period depends on the selected intermediate waste storage option, if buildings on site are utilized for other purpose, or if they are decommissioned. Figure 3 illustrates an alternative to the interim waste storage facility being located in the Cooling Water Pump Building for waste storage and decommissioning the Main Technological Building. The sarcophagi are shown as free standing buildings. The size of the fenced part of the site has been reduced.

Fig. 3

CONCLUSION

Lack of Estonian in-country nuclear expertise is requiring the involvement of foreign experts to a greater degree than a decommissioning project would normally necessitate. It is desirable to involve the Russian experts who hold knowledge in the design and construction of the reactors and auxiliary buildings as well as the operation of the reactors and the auxiliary systems at Paldiski. This is especially important during the development of the detailed decommissioning plan. Western and European technical experts are providing valuable assistance with the decontamination and decommissioning planning of the facility and with the development of the necessary site operations infrastructure.

Establishment and funding of AS ALARA, Ltd. by the Estonian government

demonstrates that they are aware of the difficulties to be encountered at Paldiski and are ready, willing, and prepared to meet them.
With international assistance, ALARA has organized and begun staffing. Development of a decommissioning plan has provided a goal to work toward and a vision of the future. With international assistance, ALARA is developing a long term site management plan and a comprehensive health physics program. ALARA's staff, with these tools and interaction with their international colleagues, is quickly developing the skills required to manage the site today and begin decommissioning activities as they move into the twenty first century.

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Session 51 -- RISK, SAFETY AND PERFORMANCE ASSESSMENTS
Co-chairs: Paul Hunt, WSRC;
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51-1

PREDICTED HUMAN AND ENVIRONMENTAL IMPACTS RESULTING FROM CONTAMINANT MIGRATION INTO A TEMPERATE WETLAND

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ABSTRACT

Atomic Energy of Canada Limited is proposing to build the Intrusion Resistant Underground Structure (IRUS) facility for the disposal of low-level radioactive waste at Chalk River Laboratories, CRL, Ontario, Canada. IRUS is a near-surface repository that is to be constructed above the water table on a sandy ridge. IRUS will consist of a concrete containment facility for the waste, including a reinforced concrete roof and multilayer earthen cover to both warn and impede intruders, and to prevent the infiltration of rain water and snowmelt. Sorbing backfill will surround the waste packages in the repository. The roof covering the facility will be fabricated of very high quality cement, but the roof will gradually lose its integrity. Once roof failure begins, an increasing portion of the vault will become wet and start to degrade, and the contents of the waste packages will begin to leach downward into the underlying aquifer and eventually return to the surface environment. The

transport of radionuclides away from the facility by advective flow has been modelled mathematically. Some radionuclides may never migrate more than a few meters from the facility before they are sorbed to soil or decay radiologically. However, some of the more mobile radionuclides, such as 3H, 36Cl and 99Tc may be transported downslope through the subsurface aquifer and surface in a wetland, known as Duke swamp. This presentation reports estimates of the total annual dose rate to a resident human through several particular pathways. Several of the pathways investigated as part of this wetland scenario are not realistic or are highly unlikely, such as drinking water from the wetland. These predictions were included as part of the scoping or bounding calculations required for a complete assessment. The highest to lowest dose rates by radionuclide are ranked in the order 3H > 14C = 237Np > 239Pu > 94Nb > anaerobic99Tc > 36Cl > 234U > aerobic99Tc. The most important internal pathway is through ingestion of vegetation; 14C and 237Np are the radionuclides giving the greatest dose rate. The most important external pathway is through groundshine from 94Nb.

INTRODUCTION

In temperate environments, particularly in glaciated areas demonstrating considerable topographic relief, performance assessments of waste disposal facilities may require wetland pathways. This presentation describes an assessment of the human dose consequence from dissolved radionuclides that have been transported in groundwater from a radioactive waste disposal facility and discharged into a wetland. We present here how we interpreted the physical environment and groundwater flowpaths, calculated radionuclide residence time in the wetland, developed the human and animal exposure pathways to be modelled, constructed a simple model, and treated special nuclides and pathways to carry out the assessment. The calculated endpoints are the total annual dose rate to a resident human. AECL is proposing to construct the IRUS facility as part of the strategy to manage its low-level radioactive wastes (LLRW). This facility for the near-surface disposal of a portion of the LLRW currently managed at CRL is now in the licensing phase. A safety case bringing together the key elements of facility design, site characteristics, waste form and inventory and performance assessment is in advanced stages of preparation. The analysis presented here resulted from a Features Events Processes Scenario (FEPS) analysis applied to the IRUS project (1).

The IRUS facility is a robust engineered containment structure for packaged wastes. The packages are surrounded by sorbing backfill inside the structure consisting of a 1-m thick concrete roof and concrete walls with a permeable bottom of sorbing buffer. Several cover layers above the roof both warn and impede intruders, protect the concrete roof from freeze-thaw cycles, and prevent the infiltration of rain water and snowmelt (Fig. 1). Although the roof covering the facility will be fabricated of tailored high quality concrete, the safety analysis assumes infiltration through the roof can begin at 500 a. Once infiltration begins in sections of the vault, the contents below will become wet and start to degrade, and the contents of the packages will begin leaching through the backfill and buffer and out the permeable bottom into the groundwater below.

Fig. 1

The proposed site for IRUS is above the water table on a sandy ridge above a coarse sand, large-volume aquifer that discharges into a small

wetland called Duke swamp. The transport of these radionuclides away from IRUS by advective flow has been estimated using mathematical models. Many of these radionuclides will be retarded by sorbing to the solid particles of the buffer, backfill and aquifer. Some may never migrate more than a few meters from the facility before they decay. However, some of the more mobile nuclides, such as 3H , a special cation, and the anions, such as 36Cl, 99Tc and 129I, may reach the wetland.

The Physical Environment, Groundwater Flow Path, and Residence Time in the Wetland

Using water table elevations, physical characteristics and hydrology data of the drainage basin, we sketched in an area of the wetland that would receive the downslope subsurface groundwater flow from the IRUS facility. A rock ledge lies along the west side of the wetland, and there is a drainage basin divide across the middle, near the "thumb" position on the east side (Fig. 2). These features will confine the discharge to a narrow band of wetland along its eastern edge. The sandy aquifer at this discharge location is 3 m deep. The wetland volume receiving the discharge of radionuclides is defined to be 60,000 m3 (400 x 50 x 3 m). This assumption regarding the size of the receiving compartment is not important, because the volume of water flushing it controls mixing in the compartment, and the consequence.

Fig. 2

The flux of radionuclide entering the swamp is simply the total radionuclide (Bq/a) leaving the upslope aquifer, as estimated with the NSURE3 performance assessment code (2), and a correction for radiological decay. We assumed that this activity mixes instantaneously in the wetland compartment. The total concentration in the receiving volume, Ct in Bq/m3.a, is then

Eq. 1

The activity that enters the wetland will be associated with both the water and the peat solids. The ratio between these two constituents in the receiving volume is expressed using the volumetric moisture content of the peat, q in m3water/m3peat, and the peat dry bulk density, rb in kg/m3 dry peat. So the total volume is

Eq. 2

We assumed that the swamp is always saturated, and that the moisture content, q, is 0.9 m3/m3 and the dry bulk density, rb , is 300 kg dry peat/m3peat. The moisture in the peat can be converted from a peat volume, W, to a peat weight basis, W' , through the bulk density, where

Eq. 3

A fraction of each radionuclide reaching the wetland will be associated with the peat solids (having been sorbed through processes such as chemical exchange and complexation) and the remainder will be dissolved in the water. The partitioning of the radionuclide between the peat and the pore water was estimated through the solid/liquid partition coefficient, Kd (Bq/kg solid)/(Bq/m3water) (3). The radionuclide concentration in the peat, Cp in Bq/kg dry peat, was determined through the relationship:

Eq. 4

The radionuclides, dissolved in the water, that flow through the wetland volume will eventually leave and flow down Lower Bass Creek (Fig. 2). An annual flow of water is defined using the downstream weir data, which yield an average of 2.04 x 105 m3/a over the last 7 years. Assuming the porosity of the modelled wetland compartment is 90%, 54,000 m3 of this

volume (0.9 x 60,000 m3) is water. The loss of dissolved radionuclides from the wetland through advected water, Cl in Bq/a, is expressed as
Eq. 5
Turnover rate in the wetland volume is 26% (54,000/204,000 = 0.26), or 26% of the water in the volume is replaced with clean water annually. However, since pipe flow along conduits of old trees and other differential decomposition patterns is quite common, it is more likely that a smaller percentage of the creek flow passes through the contaminated portion of the wetland each year. We define a fraction, f, that passes through the wetland in order to estimate dilution. We redefine Cl as Clf,
Eq. 6

The fraction 0.2 is the site hydrologists estimate of the correct dilution and the 0.1 value is a conservative lower bound. Both the annual flow and the fraction of water flowing through the wetland are held constant over time. The total concentration remaining under this assumption of quasi steady-state conditions is then
Eq. 7
The varying annual nuclide flux entering the wetland compartment is used to recalculate the concentration of radionuclide in the water (Cw) and the peat (Cp), using equation (4). The concentrations Cp, Cw and Ct are then used in the dose rate calculations in the subsequent sections to assess the human and environmental impacts.

EXPOSURE PATHWAYS

A set of exposure pathways was developed based on the primary dose rate calculations for the IRUS safety case, that had been made with the NSURE assessment code (2), and the need to provide scoping or "bounding" calculations for the disposal assessment. We also drew on extensive experience from analyses for high-level waste disposal (4). The pathways considered for humans resident near the wetland were 1) human ingestion of unfiltered drinking water, used as an upper limit for wetland water use, 2) human ingestion of vegetation grown on contaminated peat of the wetland or on peat brought to a garden to ameliorate the sandy soil, 3) human ingestion of game that drinks water from the wetland, and 4) human ingestion of game that eats vegetation from the wetland. Various combinations of dose rates via these pathways can be produced by simple addition of the individual pathway values, for example, humans eating vegetation grown on contaminated soil and game that drinks swamp water. All direct contamination activities associated with the wetland involve terrestrial activities; no aquatic pathways such as fishing or swimming have been included because there are no fish, and the wetland is not amenable to swimming.

If the humans import peat from the wetland to condition garden soil, the root zone uptake is influenced primarily by the concentration in the imported peat; we conservatively assumed the root zone (top 30 cm) is completely composed of peat. This amelioration technology is well known for use in heavy clay soils to enhance root penetration, and in sandy soils to enhance water holding capacity (5). This calculation is not only conservative, but can be used as an upper bound. We assumed that the residents also hunt game and meet some (44%) of their meat requirements from game. The game animal most likely to drink and browse from Duke swamp is a moose, so we used the water and vegetation ingestion requirements of a moose. Since the home range of a moose (5 km2, (6)) is

much larger than Duke swamp, we used a fraction describing the area of wetland to the area of the moose's home range to stipulate the proportion of browse taken from the wetland. However, we conservatively assumed the moose comes to the wetland for all of its drinking water.

COMPARTMENT MODEL CONSTRUCTED FOR SCOPING CALCULATIONS

The radiological dose rate calculations were carried out for each of the four pathways using the equations of the CALDOS model (7). The equations are of the form:

Dose Rate = Concentration x Rate of Intake x Dose Conversion Factor.

Input Inventory

The time-dependent source flux of radionuclides out of the sandy aquifer downslope of the IRUS facility over 50,000 a, were generated from the SYVAC3-NSURE assessment code (2,8). Microsoft EXCELTM was used to fit a function with fixed time steps to the NSURE output which had variable timesteps. Time steps of 10 years were used to provide enough detail to describe the aquifer fluxes for 50,000 a after the onset of roof failure began at 500 a. In the case of 3H, the time steps were changed to 1 year due to its rapid release. The Duke Swamp assessment was performed for radionuclides that had either a large initial inventory, characteristics that render them mobile in the environment, a long half-life, or high radiological toxicity. For this assessment, the selected radionuclides are 3H, 14C, 36Cl, 94Nb, 99Tc, 234U, 237Np and 239Pu. Technetium and U are subject to more retentive soil behavior under low chemical redox potential, typical of geochemical conditions in low lying wetlands and bogs. Dose rates from 99Tc were calculated twice, once for aerobic conditions and again for anaerobic conditions. All radionuclide independent and radionuclide dependent parameters were tabulated in the full report on this study (9), along with the values used, their units and their source.

Major Pathways

A simple compartment model was constructed using STELLA software to calculate the human consequences from the four defined pathways. Radiological decay was included; however, ingrowth of daughters was not. The pathway equations solved are presented in more detail in Sheppard et al. (9).

Other Pathways and Special Radionuclides

Soil ingestion is primarily important for those radionuclides with high Kd values (3,10). Dose to humans from inadvertent ingestion of soil from hands and soil adhering to plants, Ds, was calculated for 94Nb, a strong gamma emitter with a high Kd. The dose rate (Sv/a) for the soil ingestion pathway is calculated as:

Eq. 8

External pathways, such as standing on contaminated ground or using water for bathing, could increase the dose consequence to a resident. Human external dose rates from bathing were calculated for 94Nb, the radionuclide most likely to produce such a dose. The dose rate (Sv/a) from standing on contaminated ground or from groundshine (Dg) was calculated as:

Eq. 9

The dose rate (Sv/a) from immersion (bathing) in swamp water (Di) was also calculated for 94Nb using:

Eq. 10

Calculation of the dose rate to a resident human from 3H used a separate model based on the isotopic dilution of 3H with stable H, an approach

often referred to as a specific activity model. This calculation was also performed for a resident moose and given in more detail in the full report (9). The simple specific activity model gives an upper limit on the dose rate. It is assumed that all of the water in the body is tritiated and multiplies the ratio of the water in the body to the body mass by Cw to give Bq/kg body mass. This concentration in the body is then multiplied by the internal dose conversion factor for tritium (2.9E-8(Sv/a)/(Bq/kg soft tissue) (7), to give an annual dose rate (Sv/a).

RESULTS AND DISCUSSION

Concentrations in Peat and Swamp Water for Comparison with Background Soil and Drinking Water Levels

Total concentrations in peat (on a dry weight basis) and in the swamp water were calculated for comparison to federal and provincial background soil and drinking water standards (Table I), and for use in non-human biota dose rate calculations to be reported elsewhere.

Table I

Dose Rate from Individual Internal Pathways

The individual pathway maximum dose contributions are shown with the time of the maxima for 20% interception of the aquifer flow to Lower Bass Creek (Table II). The internal pathways responsible for the largest human dose for 94Nb, 234U, 239Pu and 99Tc aerobic is drinking unfiltered water directly from the wetland. The next most important internal pathway for these radionuclides is from eating vegetation grown in the wetland or in a garden with peat imported from the wetland. This vegetation pathway delivers the highest dose rates from 14C, 237Np,36Cl and 99Tc anaerobic. The maximum dose rates result from large dose conversion factors and large inventories of these nuclides. The meat ingestion pathway plays a minor role in the dose from all of these nuclides.

Table II

Total Dose Rate

Total dose rate from the four pathways are summed to provide an estimated time-dependent total dose rate for all radionuclides (Fig. 3). Differences in Kd, radiological half-life and radionuclide flux discharged from the aquifer influence which radionuclide is most important at a given time. The total dose rate for 94Nb includes the extra external pathways of groundshine, soil ingestion and immersion in water. Although external exposure pathways are generally unimportant for most radionuclides, groundshine represents a significant portion (67%) of the total dose rate for 94Nb. Groundshine for the other retentive radionuclides is less important, because their dose conversion factors are five orders of magnitude below that of 94Nb. The dose rate from ingesting soil contaminated with 94Nb is 1.4% of the total, and immersion in water (bathing) accounts for only 0.02% of the total dose rate from 94Nb. The aquifer flux inventory ranks the radionuclides from largest to smallest peak inventory (at different times), in Bq, as:

3H > 14C > 99Tc > 36Cl > 94Nb > 237Np > 239Pu > 234U

The total dose rate prediction ranks the doses from largest to smallest as:

3H > 14C = 237Np > 94Nb > 239Pu > an99Tc > 36Cl > 234U> ae99Tc

Fig. 3

The earliest and largest dose rate (8.1E-6 Sv/a) comes from 3H peaking at 22 a after roof failure in the 20% case. The next largest dose rate for this flow case is at 1600 a from 14C (Fig. 3). The peak dose rate for

237Np in this case comes through at 2300 a and the next largest dose rate is from 94Nb at 20,290 a, primarily through the groundshine pathway.

CONCLUSIONS

The most important pathway for dose consequences to a resident of Duke swamp, based on these calculations, is through the drinking water pathway. However, it is highly unlikely that someone would take all their drinking water from a wetland and not process or filter it in some way. The results from human ingestion of untreated swamp water and garden vegetables grown on imported peat from Duke swamp are more than conservative estimates, they are upper bounds. Both 14C and 237Np are important radionuclides for human ingestion through plants/vegetables. The total dose rate for the 20% case, from contaminated vegetation is 2.3E-6 Sv/a at 1600 a for 14C and 1.4E-6 Sv/a at 2300 a for 237Np. The peak dose rate from 3H occurs well before the end of the 100-year period of institutional control of the site. Humans will not have access to Duke swamp during this period. The only radionuclide to give a significant external dose in this case is 94Nb (1.2E-6 Sv/a) at 20,000 a, and this is primarily through the groundshine pathway.

ACKNOWLEDGEMENTS

We thank Bill Selander for helping us define the scenario, Doug Killey for the hydrological information on Duke swamp, Steve Wilkinson for biosphere parameter values and Fenella Lane for the NSURE aquifer flux input data. We appreciate the helpful comments of C. Macdonald and G. Bird (Atomic Energy of Canada Limited) and J. Cook (Savannah River Technology Center) on a draft of this paper.

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51-3
A GENERAL VIEW OF AN ORIGINAL PERFORMANCE ALLOCATION COMPUTER TOOL: OASIS

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ABSTRACT

Following the law of Dec. 30th, 1991 (2), ANDRA has established a work schedule until 2006 in order to propose a feasibility assessment for HLW deep disposal, based on R&D studies in two underground laboratories. In the first phase, performance for each disposal subsystem has to be defined. For this early process, a specific general assessment tool is needed: OASIS.

OASIS is a deterministic mass transport simulation software which relies on simplified modelling of the pathways. The system (from source to biosphere) is represented by a multi-1D network where hydraulic properties of each branch come from input data. The system is built by associating modules containing elementary transfer functions. The overall physical validity of OASIS needs to be supported by external studies which can rely on more sophisticated and more specific 2D or 3D codes.

OASIS also contains a calculation assistance called "study generator", whose role is to prepare sets of input data (fixed, variable or probabilistic) and to extract relevant values from the resulting blocks. All these operations are defined by the user in a object-oriented approach. The program also provides a user-friendly interface.

An unrealistic but illustrative situation has been chosen as an example to show OASIS capabilities. Simple, mono and bi-parametric studies can be performed as well. The purpose of this type of analysis is to give inputs to performance allocation, using a radiological impact criterion. OASIS is able to perform a great number of these calculations with very low CPU times.

OASIS is now in an ongoing development process where models are versatile and continuously adapted to specific needs. Future versions of the program will contain higher-level functions in order to keep the treatment or the results at a certain degree of simplicity.

BACKGROUND

The field of work for ANDRA has been defined by the law of dec.30th, 1991. It is specified that ANDRA has to give the results of a 15-year research and development work about the feasibility of a deep geological disposal of HLW by the year 2006.

Several milestones have been identified during this period ; the first one, 1997, will correspond to a first selection of main disposal concepts. Efforts are developed within this phase to define the awaited performance of the repository, in terms of functional requisites. This process is called "Performance Allocation", and must be an iterative process. It is related to the results of site investigations and should provide input to the definition of underground laboratories programs and R&D studies prioritization. Technical and economical aspects are

considered for this global process, but the basic criterion to define performances is safety.
The Basic Safety Rule RFS III-2-f specifies the overall safety objectives, principles and functions to be considered. During the Performance Allocation process, the general performance requirement is translated into lower-level specific requirements. This is done for any of the considered scenarios.

DESIGNING THE FACILITIES

The ANDRA approach for long-term disposal design is based on a simple logic: ask "What" you really need in terms of safety before asking "How" to carry it out technically. This principle leads to identify the main function the system has to perform. For the disposal, this function is the long-term annual dose limitation under a regulatory level.
This function is divided into lower-level functions which can be associated to concrete technical solutions. Except for the basic design principles given by the RFS III-2-f, all the disposal systems (the "How") follow the low-level functional requirements (the "What"). Basically, this means that the main task can be achieved by performing several little tasks which form the general specifications for the disposal design.
The requirements from each low-level function must be quantitative. This corresponds to the Performance Allocation, i.e. calculation of a performance level and confidence bounds for each function. It is a systematic and necessarily iterative process.
The only way to define those individual performances is to use simulation. The system is modelled by including all the design data available at that time. Then, through the simulations, the individual performance can be defined in order to achieve the main goal of the system, i.e. not exceeding a dose limit.
Simple and generic performance allocation can be carried out sometimes by hand-made calculations dealing with orders of magnitude. But when iterative, numerous and precise results are necessary, computation is the only answer. To do this, we have designed a new software: OASIS.

WHAT IS OASIS?

OASIS is a multi-purpose scientific software that has been developed by ANDRA for the specific need of long-term disposal safety studies. Rather than a classic scientific code, OASIS must be considered as a toolkit that helps engineers to perform advanced studies. The main features of this software are the following:

OASIS is a software that simulates transport of any quantity (activity, mass, heat) inside any physical system. These systems are based on network modelling, to simplify system representation:
Fig. 1

It is based on a general modular approach: every subsystem such as faults, waste canisters, seals, or biosphere in the case of a disposal, forms an individual module that can be assembled with any other to reproduce the network. This feature allows us building any physical system where a quantity is transported. There is a specific physical system for each release scenario, each underground facility, each waste type and each site.

Each module uses simplified modelling to obtain low CPU times and to keep some coherence with uncertainty levels. In addition, simplified models bring the user to an easier understanding of the system behavior.

Actual models for deep HLW disposal rely on analytical zero or one-dimensional schemes.

OASIS is associated with a general validation process which is geared to build confidence in the models. OASIS needs to be supported by a phenomenological approach mainly represented by 2D or 3D numerical codes. These programs are an intermediate step between OASIS and experimental results. They focus on a limited number of physical aspects but they provide a very detailed modelling of these specific processes. The physical analysis has two main goals: providing input data to OASIS (replacing a non-modelled process by equivalent data or tables) and helping to assess and justify modelling hypothesis inside OASIS.

OASIS provides a study generator, to establish various calculation modes easily. More than for classical calculations, this feature is designed for sensitivity and uncertainty analysis. Deterministic and probabilistic analysis are available (probabilistic here means random sampling of input data, Latin Hypercube Sampling Monte-Carlo for example). According to the user's demands, the generator builds data sets, runs all the needed simulations and then extracts the relevant results for decision-aid processes.

A user-friendly interface is also included so that access to information (data, models and results) is possible without special computing knowledge. From the user's point of view, OASIS appears like an object-oriented software: a physical module, a transported element, a calculation mode with its parameter ranges or a graphical result are specific objects that can be defined independently. That means that several objects from the same type can be defined simultaneously during just one calculating session (more than one calculation type on a system, for example). The user can do any association between objects, as shown further.

OASIS relies on well-known computing solutions such as UNIX, FORTRAN or C. No "exotic" languages or operating systems in order to guarantee long-life for the software. A complete Quality-Assurance process is associated to the development cycle.

HOW IS IT BUILT?

The main technical features of OASIS form the basis of a particular internal structure divided into 5 levels. This structure and some other characteristics have been patented in 1994:

Level 1 contains all the basic modules. It is a model database filled with individual simplified models. In our case, each of the following components are placed inside a specific analytical module: source term, buffer material, disposal architecture, geological media, special geological features or biosphere. Each module contains its own numerical default database (values and uncertainties) related to each of the model parameters.

Level 2 contains the network modules. They are used at each node of the transport network as flux interface between level 1 modules. In an underground hydraulic system, these modules can divide the flux going into parallel pathways or sum it where different pathways converge. These modules are necessary to build the multi-1D approach.

Level 3 contains the scenario generator. This step provides time dependency for the multi-1D network, in order to represent a complete scenario with its event logic. Behavior of the studied system is totally described at this stage. The three first levels are called the

"calculation kernel" and provide results similar to those from classical scientific codes.

Level 4 contains the study generator and has been described. Its instructions come either from the level 5, or from hand-made data sets provided independently. These data sets must be formatted in a specific language especially developed for that purpose, the LSE (from french "Langage Spcifique aux Etudes"). This choice allows running the code independently from level 5, starting from any remote text terminal.

Level 5 contains the user-friendly interface. It is a graphic facility where every instruction coming from the user is translated in LSE. Level 5 can be run independently from the rest of OASIS (to prepare data sets or to analyze old results) or simultaneously (classical use).

PHYSICAL MODELLING CONSIDERATIONS

This chapter provides more detail on levels 1 and 2 and on the validation process for the deep disposal application.

The source term module is a flux or concentration emitter which considers the following options:

- Direct contact between waste and the following media.
- Presence of a hydraulic buffer between wastes and the media.
- Immediate activity release from waste.
- Time-dependent activity release.

Waste matrixes, overpacks and solubility limits are considered in the module. The flux or concentration signal is calculated depending on the model option.

The engineered barrier and the features of the geologic media are associated to transport modules. There are modules for different phenomena:

- Pure advection.
- Advection-dispersion.
- Pure diffusion.
- Sorption and chain decay are also considered.

These modules exist for various 1D geometries: cartesian and axisymmetric. Boundary conditions can be chosen between flux, concentration or flux/concentration relation at the input or at the output.

The biosphere corresponds to a module with a conversion factor per radionuclide.

These informations form a model database. The connection between these modules allows building a large number of pathways for scenarios.

In consequence, each pathway is a sum of physical hypothesis which need to be evaluated by validation exercises. An example for validation is presented on Fig. 2:

Fig. 2

A specific combination of modules with OASIS (source with advective-dispersive geosphere) has been compared to a 2D numerical code, DIMITRIO (finite-element code developed by the French Atomic Energy Commission). An initial 2D simulation has been performed to define equivalent 1D hydraulic characteristics, by identifying the main flow tube (length and water travel time) and by performing a sensitivity study on the main parameters. The resulting hydraulic values have been entered for the OASIS 1D transport model. The figure shows that differences between 1D and 2D are acceptable, even if retardation is changed.

This result shows a part of the validity field for the 1D hypothesis, but it remains very specific. This example has to be completed by many other numerical comparisons, to explore the limits of the models.

HOW TO USE OASIS?

OASIS is implemented on an UNIX workstation and has been developed with standard and portable techniques. A PC version may be developed in the near future. In the case of the HLW waste disposal, the main classical steps to perform calculations with OASIS are the following:

- Define the study name.
- Choose the physical modules in the database. Choose the right boundary conditions, geometry, physical processes for your scenario.
- Link the modules in order to build the multi-1D network.
- Choose radionuclides to be transported. These can be fission products or actinides. Chain decay can be defined afterwards considering any nuclide.
- Define one or more calculation objects. Each calculation object contains the study type (reference, sensitivity, probabilistic), calculation techniques (deterministic, mono or bi-parametric, probabilistic), reference values for fixed parameters, intervals for continuously variable parameters and several general-purpose parameters (min and max simulation time, result filters,...).
- Define graphical results (results can be activity or dose rate, concentration at any point,..). Each object contains all the operations to be done on the results before plotting: curve maximum , means or standard deviations, linear combination of curves, curve intersections with specific levels, etc... All the graphic features must be defined at this stage.

All this is supposed to be done with the help of the user-friendly interface: the mouse is widely used. It is also possible to work without a graphic device: then the user has to write or modify a LSE data file where all the above steps can be found in a special format.

A study managed with OASIS induces few constraints. There is no need to do the operations in a monolithic one-way process: excepting some common-sense operations (like defining a module before modifying its data), many operations can be done independently or done by iterations. Calculations may be performed before defining the treatment of the result.

AN APPLICATION WITH OASIS

The following application deals with the HLW disposal. It is an illustrative example of the type of calculations performed within the Performance Allocation phase. The aim was to determine performance of the safety functions in a disposal so that dose rate does not exceed a fixed limit. A strongly degraded, unrealistic and simplistic situation has been chosen. This has been done to identify the effects of various functions potentially involved in radionuclide containment. It also permits to have simplified models and easily readable results.

Let us consider no engineered barrier and a geosphere hydraulically short-circuited. Waste packages are supposed to be enclosed in an overpack with a finite lifetime, which performs a retardation function. After this period, all the overpacks are supposed to fail simultaneously. When this failure occurs, the waste matrix starts releasing radionuclides at a certain rate. Then, the nuclides reach the biosphere almost immediately with an additional limitation for the low-solubility ones: they are supposed to precipitate near the source, after their release. Three functions can be identified easily:

Flux limitation by the waste matrix release rate.
Flux limitation by retarding the release with the overpack lifetime.
Flux limitation for low-soluble nuclides with the low hydraulic flow.

The first question is "What performance do we need for each function (associated to a specific subsystem) to ensure that dose will not exceed the limit?". During each calculation, the function is supposed to operate alone: the remaining functions are absent or operating at very pessimistic levels.

Figure 3 provides the answer for the second function: lifetime of the overpack. This function relies on radioactive decay. The first and the third functions are taken at a pessimistic level (high). This graph is obtained by performing mono-parametric calculations with many different overpack lifetimes. Then, the maximum relative dose per radionuclide is extracted from each result and plotted in the [dose rate vs overpack lifetime] performance graph.

Fig. 3

The intersection of each performance curve with a horizontal dose limit gives the performance level required from the function for each radionuclide (cumulative dose can also lead to a specific performance). Care must be taken for chain actinides, whose contribution can increase and then decrease with performance: the relevant intersection is the last one in that case (on the right hand of the graph). By evidence, overpack lifetime (or retardation) seems to be much more relevant for short-lived elements than for long-lived or end-of-chain actinides. The absolute result is the performance value for each radionuclide, the relative result is that it might be more interesting to count on retardation for some radionuclides rather than for others (a matter of feasibility). This work can be done for the two other functions.

The second question is "What should the performance of two functions be if their contribution to safety was simultaneous?". This is done by performing bi-parametric calculations on the following parameters: overpack lifetime (retardation) and waste matrix release rate (limitation).

The following figure (Fig. 4) shows a network of iso-dose curves. These curves are obtained by plotting the previous performance graphs with many different release rate values. For each of these graphs, the intersection of the curves with a dose level provides the retardation value. Then, this retardation value associated to the release rate value of the graph form the [x,y] coordinates for the iso-dose graph. Each curve corresponds to four different levels of relative dose: R, R/2, R/4 and R/20.

Fig. 4

The area below an iso-dose curve is a zone where every combination of the two parameters leads to a dose exceeding the limit. The upper area represents all the parametric combinations leading to a dose under the limit: this zone can be interpreted as "acceptable". By evidence, the lower the dose limit, the higher the performance of both parameters must be.

This graph also shows that reducing the dose level might be done in an optimized way. At places in the graph where curves have a horizontal tendency, it seems to be more convenient to increase the waste matrix performance rather than the retardation performance. In that case, lower dose levels seem to be reached easier. The opposite observation can be made in places of the graph where curves have a vertical tendency. In

complement, the other zones show that joint performance increasing of both parameters seem to be the fastest way to reduce biosphere impact. These conclusions are given without taking care of any cost or feasibility aspects, but the main purpose of these graphs is to provide designers with all the relevant performance limits in terms of safety. It is possible to agree with any optimization of the disposal, as long as the system behavior remains inside these safety boundaries.

OASIS allows performing these calculations and many others very easily and with low CPU. The last graph (the bi-parametric one) needed 2,000 simulations of the base case to have all the necessary values. The order of magnitude of the whole calculation time was a few dozens of seconds.

THE FUTURE OF OASIS

The development of OASIS is an ongoing process which will provide the software with features designed to take into account the growing complexity of the calculations. These features can be the following:

- Consideration of time dependence for models: treatment of complex scenarios.
- Inverse techniques for problem resolution.
- Consideration of scenario probabilities - risk approach.
- More complex and complete modules, calculation modes and result treatment.
- Alternative resolution methods (semi-analytical, Laplace's transform, numerical).
- Extension to non-radioactive products migration (e.g. chemical).
- Links with an independent database managing system, like ORACLE.
- Implementation of OASIS on a network, possibly on PCs.
- 2D or 3D models if necessary (much later).
- Visual simulation of mass transportation (visible evolution).

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51-4

THE INFLUENCE OF INVENTORY DISTRIBUTION ON THE SAFETY ASSESSMENT OF THE ABADIA DE GOIAS REPOSITORY

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ABSTRACT
Due to an accidental breakage of a Cs-137 radiotherapy source, September 1987, 40.1 Tbq of waste was generated in Goiania, Brazil. All of this waste will be disposed of in a Low Level Waste Repository. The proposed repository is above grade with a soil cover. It is 60 m in length, 20 m wide, and 5 m deep.
An analysis of the influence of inventory distribution is conducted using the BLT computer code. BLT (Breach, Leach, and Transport), is a two dimensional finite elements code, that calculates container degradation (Breach), radionuclide release from the wasteform (Leach), and migration of radionuclides to the facility boundaries (transport).
Two cases are considered: a) the concentrations of Cs-137 in the near field, and b) the concentrations of Cs-137 at a well 100 m down gradient from the repository.
INTRODUCTION
In September 1987, an accidental breakage of a Cs-137 radiotherapy source generated 40.1 Tbq of waste near Goiania, Brazil. During the decontamination work, the waste collected in several different types of packages and removed to an interim storage facility near Goiania city (1). The waste occupies 2650 m3 and is stored as follows (1):

- one package containing the remaining source,
- 90 concrete containers,
- 16 cylindrical carbon steel containers with a volume of 5.7 m3,
- 987 rectangular carbon steel boxes with a volume of 1.7 m3,

Both types of carbon steel container have a wall thickness of 6.35 mm. The waste is 90% soil, rubbish and scrap, 8% paper, plastic and clothes, and 2% organic material. The distribution of waste within the containers is such that 62% of the activity is contained in 1.7% (45 m3) of the volume. Also, 92% of the inventory is contained in 20% of total volume (503 m3).
The proposed repository is above grade with a soil cover. It is 60 m in length, 20 m wide, and 5 m deep (2), see Fig. 1a. The bottom of the repository is 4 m above the aquifer. For most subsurface facilities, the ground water pathway is largest contributor to dose (3). This is particularly true when the distance of the aquifer is small, as is the case at Goiania. Therefore, the basis for comparison for this report will examine release from the repository Other pathways, such as run-off, may be important but are out of scope of this paper.
In most performance assessment, it is assumed that the inventory is uniformly distributed within the facility. This is not the case at Goiania. The non-uniform distribution of waste at Goiania requires an analysis if the potential releases from the repository based on the actual distribution of wastes. For comparison, potential releases will also be estimated for a uniform distribution of wastes. The analysis will be conducted using the BLT code (4). The impact of non-uniform inventory distribution within the repository will be assessed and if significant differences in potential dose occur, recommendations on waste emplacement will be made.
Fig. 1
METHODOLOGY

For this analysis the BLT (Sullivan, 1989) computer code is used to estimate ground water concentrations. The BLT code, is a two dimensional finite elements code, that calculates Breach (container degradation), radionuclide release from the wasteform (Leach), and migration of radionuclides to the facility boundaries (Transport).

BLT reads an input file with information pertaining to the system geometry, water flow, container degradation, waste form leaching, and contaminant transport. This information is used to predict the time-dependent movement of radionuclides through the trench for a single radionuclide species. Wasteform release may be modeled based on three processes: diffusion, dissolution, and rinse.

As a preliminary calculation, complete failure in the engineered barrier as well as in the containers is assumed to occur instantly. Therefore, the infiltration velocity is equal to the containers rate, i. e., rainfall - evapotranspiration (Vz= - 1.9 e-6 cm/s, i. e., 60 cm/yr.)

The finite element mesh used to simulate the disposal facility is presented in Fig. 2.

Fig. 2

The cross section analyzed is the transversal one A-A, see Fig. 1-b, because it provides a shorter radionuclide migration path to the border of the repository.

The initial condition for all the 252 nodes is zero concentration, and the boundary conditions are zero mass flux entering the upper edge of the region and zero concentration at the bottom, leading to a maximum release.

Due to the large number of waste containers it will not be possible to model each of these containers individually. Some containers will be lumped together to form a representative wasteform/container system. Different arrangements of these representative groups will be analyzed. The waste containers are divided into three types that are represented in the waste containing region.

Type 1 is the metal box, type 2 represents the concrete container and type 3 represents the source container.

In total, there are 15 containers containing elements.:

Elements 187,188, 199, 200, 201, 212 represent type 1 containers,
Elements 121, 122, 123, 160, 161, 162, 186 represent type 2 containers,
Elements 213, 214 represent type 3 container.

In the initial stage of the assessment, it is assumed that all containers fail instantly at closure of the facility, t=0. Refinements to the analysis will consider different failure times for each container type. The release from the waste forms is assumed to be controlled by the rinse process which gives instantaneous release of all the inventory. This is the most conservative assumption that can be made about release from the waste form.

For the simulation which considers non-uniform distribution of wastes within the repository, the type 1 containers have 69 Ci per container, type 2 have 69 Ci per package and type 3 have 93 Ci per package. For the uniform distribution case, all containers have 72.3 Ci per package.

In the BLT code, the repository was modeled with 4 different material types:

1) undisturbed soil
2) concrete-engineered barrier
3) backfill-compacted soil
4) wasteform-mixture of cement and bentonite

Site specific data was available for the distribution coefficient in the undisturbed soil. Estimates of other material properties are based on typical literature values.
The properties used are as follows:
Table I
Preliminary Results
For preliminary calculations, it was considered as a first step that the entire inventory is distributed over the packages, and after that, a non-uniform distribution of the waste. In the following table are presented the concentration peak with the corresponding times.
Table II
From Table II, there is not a great difference neither between peak concentrations nor between the times of peak concentration for the two cases (uniform distribution of wastes versus non-uniform). The very short time for peak concentration was due to the high infiltration velocity corresponding to a complete failure of barriers, and the short distance to the aquifer (4 meters).
Well Scenario
For the well scenario, it is considered a well 100 m down gradient from the repository. The finite element grid for this case is shown in Fig. 3.
Fig. 3
It has been considered only two material types: the undisturbed soil and waste/concrete. Also, different soil Kd values have been analyzed (5). According to reference (5), the soil Kd values are greatly influenced by the Cs molar concentration, and therefore the well concentration will be function of the Cs molar concentration. The expected average value for Abadia de Goias soil Kd is 430 cm 3/g, for a molar concentration of 1e-08. The distribution coefficient of the site soil values have been determined by the batch method. Table III shows some values for well concentration versus soil distribution coefficient Kd. These concentration have not been influenced by the wastes distribution.
Table III
From Table III, to exceed a ground water dose of 4 mrem/yr. from the ingestion of 2l/day of contaminated water, it is needed to insure that the ground water concentration does not exceed 1.1e2 pCi/l.
CONCLUSIONS
A preliminary investigation into the influence of inventory distribution within the Abadia de Goias Repository has been conducted. As a comparison, releases to the near field have been determined for two cases, i. e., uniform and non-uniform wastes distribution within the repository. The distribution of wastes was found not to influence peak concentrations greatly.
For the well scenario, the analysis has shown that for the conservative assumptions (instant barrier failure and total release of the inventory), the distribution coefficient of the aquifer would have to be less than 50 cm3/g . The measured distribution coefficient for the site is approximately 430. Therefore, the expected doses are far bellow the standard.
For the well scenario, the analysis has shown that for the conservative assumptions (instant barriers failure and total release of the inventory), the distribution coefficient of the aquifer would have to be less than 50 cm3/g in order to have a ground water concentration above the limit of 4 mrem/yr, for a water consumption of 2 l/day.

However, if all the waste was instantly released, the concentration would be less than 1e-9 molar in the disposal facility and therefore, the expected doses at the well are far below the standard limit.

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THE ROLE OF A DETAILED AQUEOUS PHASE SOURCE RELEASE MODEL IN THE LANL AREA G PERFORMANCE ASSESSMENT

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ABSTRACT

The Performance Assessment for the LANL Low-Level Radioactive Waste (LLRW) Disposal Facility, Area G, is on-going. A detailed review of the inventory data base records and the existing models for source release led to the development of a new modeling capability to describe the liquid phase aqueous transport from the waste package volumes. Inventory is sorted into four release form categories and screened in a comparison of leachate concentrations to the drinking water limit. Percolation through the disposal unit is prescribed in an independent surface water balance model incorporating site rainfall statistics. Waste package types and the disposal unit matrix have independently specified solubility limits and solid-liquid phase partition coefficients, or Kd values. Analytic solutions for inventory limited release of each nuclide in each of the four different waste package release forms are computed. Isotopic contributions are summed over elements to limit the waste package liquid phase concentrations to the elemental solubility limits. Time dependent

releases from the waste packages for each nuclide which may be inventory or solubility limited are specified as model output which is provided as the source term to the unsaturated transport model. The waste package efflux is distributed over the 2-D unsaturated zone model grid points corresponding to the cross-sections for 5 representative disposal units within the mesa top. Results show the Area G release is dominated by the inventory in the 'rapid release waste form' (Kd = 0), which percolates from the waste packages over 5-100 years and from the disposal unit over 50-1000 years. Nuclides in waste package categories with larger Kd values are released proportionately slower. Uranium and thorium are the main nuclides of concern released as solubility limited nuclides from the 'historical inventory' at Area G. The analytic models provide an efficient means to explore the sensitivity of the results to variations and uncertainty in the model parameter values.

INTRODUCTION

A preliminary draft of the site Performance Assessment (PA) for the Los Alamos National Laboratory (LANL) low level radioactive waste (LLRW) disposal facility at Area G has been completed 1) as required per DOE orders. A detailed review of the inventory data base, records and existing models for source release led to the development of a new modeling capability 2) to describe the liquid phase aqueous transport from the waste package volumes. The code output, a time dependent afflux to the disposal unit volume (as detailed in App. 3B to Ref. (1), is provided as input to a sophisticated 2-D unsaturated flow model 3) for subsequent transport within the disposal unit and throughout the vadoze zone.

The Area G disposal facility is located on the top of a narrow finger-like mesa composed of volcanic tuff (Bandelier Tuff), deposited in stratigraphic layers of ash and solidified volcanic flows. Waste disposed at Area G is placed into pits excavated in the volcanic tuff, crushed in place, and backfilled with the native crushed tuff to about 30% waste package and 70% tuff by volume. These disposal operations are evolving to minimize future disposal volume, and to assure stability of emplaced waste.

Disposal operations at Area G began in 1959 and disposal inventory is established in a detailed electronic data base dating back to 1972. (Disposal for operations prior to that date is recorded in log books currently being transcribed to a new data base.) Nuclide quantities are associated with 50 waste codes, which characterize the physical and chemical form of the waste.

ANALYSES

These waste codes were sorted into four waste package release categories, based on preliminary modeling efforts which indicated the release characteristics of concern. The four release categories include rapid release (with compartment release time of one year, and with solid-liquid phase partition coefficient, Kd, set equal to zero, taking no credit for solid phase reabsorption), soil (Kd for Bandelier tuff at neutral pH), concrete/sludge (Kd for concrete at pH=12, and corrosion associated with metal waste forms (with compartment release time of three hundred years, and with Kd=0).

Of the 60 inventory nuclides listed with half-life greater than 5 years, 30 were screened out in a 'leachate screening', and 30 were followed in detail by the source release model. The leachate screening compared the total inventory concentration in the disposal unit moisture to the

drinking water concentrations that result in 4 mrem/yr at the standard ingestion rate. Of the 30 nuclides followed in detail, 15 were decay chain parents, with 48 non-secular equilibrium decay chain products evolved self-consistently.

Geochemistry for the waste packages was evaluated in terms of the equilibration coefficients, Kds, and elemental solubility limits, Csl, interpolated from the literature by Longmire (4). Yucca Mountain Kds (5,6) were used as best estimates for Kds in tuff and as conservative estimates in soil, justified by a favorable comparison of water chemistry between Bandelier and Yucca Mountain tuff samples (4). Solubility limits for tuff (7) were applied to all waste packages except waste in concrete/sludge, where values taken from the literature (8) were applied. A program to evaluate site-specific Kds under field conditions is underway at Los Alamos. This will include five key nuclides in several Bandelier Tuff stratigraphies and over a range of water chemistry.

Percolation calculations were run with a modified version of the SPUR code (9) which does a detailed water balance at the surface. Results for the base case closure cover, 90 cm of crushed tuff, 10 cm soil and a thin gravel-mulch seeded surface, showed an average of 4 mm/yr percolation from the disposal unit bottom (10). Recent results with revised crushed tuff hydrologic properties show an infiltration rate of 5.5 mm/yr. The statistical analysis shows the average percolation rate is for a highly skewed distribution composed of 80% values equal to zero. The implications on the source release model of the actual distribution rather than the average value are under investigation. A range of infiltration model parameter variations were evaluated to account for evapotranspiration including rooting depth and leaf area index, and for run-off including surface slope and other factors important in the surface water balance (10).

The waste release model is based on a compartment representation of the package afflux, and depends upon package size, percolation rate or Darcy flux, retardation coefficient, and moisture content. The physical and conceptual flow models are shown in Fig. 1, where the concentration and flux symbols are defined. An analytic solution for waste package afflux under inventory-limited conditions (everything which is not solubility limited) is evaluated for each nuclide in each waste package type (or release category). This is described in detail elsewhere (2) with the result for the waste package liquid phase compartment concentration, Cw, written as

Eq. 1

with

Eq. 2

The compartment subscripts are shown in Fig. 1, and the l values are 'compartment clearance rates' derived by a control volume integration of the continuity equation defined in terms of the compartment area, A (horizontal planar area), volume, V, Darcy percolation rate, q, dissolution velocity, u, moisture volumetric content, q, and retardation coefficient, R, related in the usual way to the solid-liquid phase partition coefficient, Kd, as R = 1 + rKd/q.

Using this time dependent concentration, the waste compartment afflux, Gw, is

Eq. 3

Nuclide contributions to the same element, including ingrowth from decay chains, are summed to compare to the elemental solubility limits in each

waste package type. Waste package concentrations for each nuclide of the solubility limited elements are partitioned to the contributing nuclides. The solubility limited waste package afflux for a nuclide is

Eq. 4

where Csl is the elemental solubility limit and fsli is the fraction of the solubility limit which is contributed by that specific nuclide. This fraction is assumed equal to the nuclides contribution to the elemental solubility limit at the time when that limit is first exceeded. Solubility limited effluxes continue until the solid phase waste package concentration is depleted to the point the water phase concentration can no longer exceed a solubility limit, and subsequently the waste package is depleted at the inventory-limited release rate (2). An algorithm of how the solubility limits are tracked over nuclides, elements and waste package types is included as Table I.

Table I

Fig. 1a

Fig. 1b

The release model was tested extensively as described previously (1,2). The analytic models described above were supplemented with numerical modeling in two areas. One, the ingrowth of progeny nuclides which can have transport properties different from the parent nuclides makes the analytic solutions to the compartment model impractical. Ingrowth of progeny is calculated numerically in a simple implicit scheme which implements the compartment governing equations with the added complexity of progeny ingrowth. This was used in the present modeling effort to account for 'daughter-product' ingrowth which could contribute to the solubility limit within a waste package. From the perspective of nuclide transport in the site PA work, ingrowth within the packages was considered negligible, and ingrowth was modeled within the disposal unit and in the unsaturated zone within the 2-D unsaturated model (3) because that code has had more extensive QA and verification.

A second numerical modeling effort examined the compartment solutions where the waste package afflux feeds a one-dimensional (vertical axis) representation of the disposal unit. In this case, release from the disposal unit is controlled by the vertical gradient at the disposal unit bottom which evolves consistently along the vertical axis from waste packages homogeneously distributed throughout the disposal unit. This work confirmed that the global or compartment model results were reasonably accurate compared to results with 1-D profile effects in the disposal unit (2).

RESULTS

Figure 2 shows the concentrations verses time inside the waste package matrix. The solid concentration, Cs, is uncorrected for readsorption in the solid phase and corresponds to the case where Kd = 0. The actual solid phase concentration corrected for readsorption inside the waste package and assuming a nuclide with Kd = 0.5 is shown as 'Csmod', based on a governing equation derived elsewhere (2). Csmod decays over the time scale for 'rapid release' to an equilibrium plateau at a fraction, (rmKd/q/(1+ rmKd/q)), of the original solid phase concentration. It then decays to zero over the time scale for percolation from the waste package, along with the waste package liquid phase concentration, Cw. If this nuclide becomes solubility limited, the Csmod concentration is needed to accurately track the solubility limited release through the solubility limited release duration.

For the 'historical inventory' (1988 to the present) and site conditions at Area G, most nuclides are found to be inventory limited. An important result observed from the source release model is that for inventory limited nuclides, it is the component of inventory which is considered to be 'rapid release' which dominates the afflux function of time. All other release forms effect slower release rates which therefore reduce the afflux peak in time. The eventual peak in the aquifer concentration is determined by this release model modified by translation and dispersion during transport through the unsaturated and saturated zones.

Fig. 2

These concepts are illustrated in Fig. 3, which shows the inventory release, Ci/yr, as a function of time for four of the plutonium isotopes in the Area G inventory. The peak release rate for each nuclide is dominated by the 'rapid release' (Kd = 0) waste package source term which occurs at about 3-5 years. Contributions to each nuclide from the other waste package release forms have relatively slower release rates due to effects of solid-liquid partitioning and readsorption (Kd > 0), and are seen in the figure as the 'bumps' which occur later in time from 200-10,000 years. Because these release rates are so small, the contribution from these waste package types to the peak release rate is small even if their inventory is comparable to that in the 'rapid release' category.

Fig. 3

For the Area G inventory and site conditions, the only nuclides which contribute significantly to dose and are solubility limited are uranium and thorium. Uranium is depleted at a solubility limited rate which is sufficient to reduce the concentrations to an inventory-limited release at about 5000 years, after which the source decreases rapidly to negligible levels. This is illustrated in Fig. 4 for three isotopes of uranium. As in the inventory limited case in Fig. 3, there are contributions from multiple waste package types. The most inventory is in the 'rapid release' form which is solubility limited until about 5000 years. Imposed on top of this is a contribution from another waste form (concrete/sludge) which is also solubility limited but is depleted at about 500 years. Most of the uranium in this form is U-235, so only that isotope is noticeably influenced by the second waste form.

Fig. 4

The thorium solubility limit is so small that the release remains solubility limited effectively for all time. Plutonium is close to the solubility limit and would become solubility limited under slightly increased disposal concentrations.

Uncertainties in the results are examined via parametric examination of the input variables, and will be discussed in some detail (1,2). Variability due to transient percolation effects is currently under investigation. Spatial variation in percolation rate or percolation differences within the disposal unit matrix verses within the waste package itself are being evaluated.

CONCLUSIONS

The results of a site Performance Assessment depend critically upon the source release model and its input data. The maximum concentrations in the time dependent afflux from the waste package will be integrated over the disposal unit volume and then translated through the vadoze zone and diluted in the aquifer while being modified slightly for dispersion and diffusion during the aqueous phase transport. Thus, it is a modification

of the time dependent source release which will determine the receptor location concentrations and ingestion dose.

ACKNOWLEDGMENT

This work is sponsored by the U.S. Department of Energy, Waste Management Program Office.

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DEVELOPMENT OF A RISK-BASED APPROACH TO HANFORD SITE CLEANUP

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ABSTRACT

In response to a request from Mr. Thomas Grumbly, Assistant Secretary of Energy for Environmental Management, the Hanford Site contractors developed a set of risk-based cleanup strategies that 1) protect the public, workers, and environment from unacceptable risks; 2) are executable technically; and 3) fit within currently expected annual funding profile.

These strategies were developed because 1) the U.S. Department of Energy and Hanford Site budgets are being reduced, 2) stakeholders are dissatisfied with the perceived rate of cleanup, 3) the U.S. Congress and the U.S. Department of Energy are increasingly focusing on risk and risk-reduction activities, 4) the present strategy is not integrated across the Site and is inconsistent in its treatment of similar hazards, 5) the present cleanup strategy is not cost-effective from a risk-reduction or future land use perspective, and 6) the milestones and activities in the Tri-Party Agreement cannot be achieved with an anticipated funding of 1.05 billion dollars, or less annually.

The risk-based strategies described in this paper were developed through a systems analysis approach that 1) analyzed the cleanup mission; 2) identified cleanup objectives, including risk reduction, land use, and mortgage reduction; 3) analyzed the existing baseline cleanup strategy from a cost and risk perspective; 4) developed alternatives for accomplishing the cleanup mission; 5) compared those alternatives against cleanup objectives; and 6) produced conclusions and recommendations regarding the current strategy and potential risk-based strategies.

This analysis produced a framework and a set of tools that are available for dealing with changes to anticipated funding levels, changes in risk cleanup standards, and Congressional initiatives and inquires. The tools include land-supply curves, cost profiles, risk profiles, mortgage-reduction curves, and minimum operations costs. This paper describes the methodology used to develop mortgage-based, risk-based, and land-based cleanup strategies and how these strategies differ in terms of the work to be performed, its sequence, and the resulting end states.

Some of the principal findings are:

A 50 percent reduction in the cost of cleanup must be immediately achieved and sustained to meet existing commitments and schedules with the project budget. This reduction would require significant savings from some combination of privatization, productivity enhancements, and regulatory relief.

The Hanford Site committed, nondiscretionary costs necessary to maintain safe operations are estimated to be 500 to 700 million dollars annually.

The value of alternative future land uses should be balanced against the cost required to achieve the cleanup levels necessary to enable those uses. Achieving unrestricted use of all Hanford Site land is not

economically feasible given the expected annual funding profile of $1.05 billion (or less) and the limitations of cleanup technology.

Four decisions are critical to any risk-based strategy due to their huge costs and risk implications. These decisions are 1) retrieval/treatment versus in-place disposal of tank waste, 2) retrieval/treatment versus in-place disposal of solid waste, 3) entombment versus removal of major facilities and reactors, and 4) treatment and restoration versus restricted access to groundwater.

If a risk-based strategy is adopted for specific waste materials, the following would apply.

Nuclear Materials. Near-term risks associated with these material would be given priority in the near-term. Given the lack of a national policy for ultimate disposition of these materials, the Hanford Site strategy must plan to manage them safely and efficiently for an indefinite period. An overall strategy for providing long-term, low-cost storage for nuclear material should be developed and implemented.

Tank Waste. The resolution of tank safety issues and prevention of release of contamination would be given priority. Tank waste disposal and related activities would be delayed, pending the determination of the best solution and the resolution of higher-priority cleanup activities. In-situ disposal of tank waste would be the preferred option. Technology for in-situ disposal should be developed.

Solid Waste. A consistent set of standards must be developed and applied for management of solid wastes. In general, buried solid wastes would not be retrieved and stored solid waste would be disposed in-place.

Environmental Contamination (groundwater and soils). Groundwater treatment with present technology is not cost effective and would not be pursued. Access and use restrictions and monitoring would continue while natural processes work to restore the groundwater and technology is developed. Contaminated soils would be capped, covered, and disposed in-place. Environmental restoration activities would be prioritized by benefit per investment dollar.

Facilities. Major facilities would be used for waste disposal and entombed in place. Reactors would be entombed in place. Ongoing mortgage-reduction activities would continue. Other mortgage-reduction activities with high benefit to cost ratios would be accelerated.

The innovative methodology developed in this paper has considerable merit and is recommended for use in development of a Complex-wide cleanup strategy. Hanford managers are applying the results, especially in guidance to contractors for prioritizing the FY96 multi-year program plans and in taking new positions on key decisions, such as cleaning up to less-than-residential standards and treating waste in place instead of retrieving it.

INTRODUCTION

The Hanford Site is at a crossroads. Budgets have been cut substantially and more cuts are expected in the future. The Department of Energy (DOE) is being scrutinized by the U.S. Congress, the Defense Nuclear Facility Safety Board, and the National Academy of Science. Challenges have been made to the Hanford Federal Facility Agreement and Consent Order (Ecology, et. al 1994) (Tri-Party Agreement), and stakeholders and the public are dissatisfied with the rate of progress of the cleanup. Hanford site contractor personnel met with the Assistant Secretary for Environmental Management, Mr. Thomas Grumbly, in December 1994, to discuss the impact of budget cuts on current Hanford Site activities.

During that meeting, Mr. Grumbly requested a risk-based strategy be developed for the Site for the entire cleanup period. Acting on this request, a multi-contractor team from Westinghouse Hanford Company, Pacific Northwest National Laboratory,a and Bechtel Hanford, Incorporated was established to prepare a risk-based cleanup strategy. This paper briefly describes some of the results of that effort, which is formally published in a Pacific Northwest Laboratory Report, Development of a Risk-Based Approach to Hanford Site Cleanup.

Scope of the Study

The team first had to determine the scope of the effort and the elements of a cleanup strategy. The scope of the cleanup mission included those processes that:

Reduce or maintain the health, safety, and environmental hazards to acceptable levels;

Convert hazards to safe, stable forms;

Maintain or disposition existing materials, facilities, and waste inventories; or

Restore the land to enable future uses.

A cleanup strategy is a set of statements that describes the actions that must be taken to accomplish the cleanup mission. A strategy describes the sequence and priority of cleanup activities and the amount of resources that will be applied to those activities over the cleanup period. It specifically describes 1)what will be done, 2)when it will be done, and 3)what the result will be.

Assumptions and Requirements

The development of a risk-based strategy was based upon the following assumptions:

Funding for cleanup would decrease from current levels to about $1.05 billion dollars annually, beginning in fiscal year 1998. Funds could be moved across major program lines.

Regulatory waivers or modifications, including statutory changes, could be obtained, if necessary to execute the strategy.

The Tri-Party Agreement could be renegotiated, if necessary.

All aspects of the current cleanup Hanford Site Strategy could be changed, if warranted.

Given these assumptions, we determined that any cleanup strategy was required only to: 1) protect the public, workers, and environment from unacceptable risks; 2) be executable technically; and 3) fit within the anticipated funding profile. It specifically was not required to meet regulatory requirements.

Then, considering these assumptions and requirements, the approach followed by the study team was to identify the funding available to pay for cleanup activities, determine the highest risks at the site, and build a program that attacked the highest risks first--within allowable funding.

Minimum Safe Conditions

As a first step, we determined how much of the Hanford budget was "nondiscretionary," that is, necessary to protect the public and workers from exposure to stored materials and waste. In this context, minimum operations were defined as "those surveillance, maintenance, and support costs required to control existing material, waste, and facilities in a safe, stable condition." No remediation, stabilization, or disposal costs were included, and new activities required to comply with regulatory agreements would not be completed at the minimum operation funding level.

However, current routine reporting activities necessary to satisfy public and worker safety regulations for surveillance and maintenance were embedded in minimum safe operations estimates.

We determined that Hanford Site nondiscretionary costs to maintain minimum safe operations to be in the range of 500 to 700 million dollars per year. The difference between the committed, nondiscretionary costs and the anticipated budget constraint of 1.05 billion dollars annually represents the funding available to make progress in cleanup. That means the DOE has about 350 million to 550 million dollars annually available to apply to cleanup.

Note that over time nondiscretionary costs would be reduced as "mortgage-reduction" activities were completed. However, some costs would increase due to further deterioration of facilities and associated growth in surveillance and maintenance costs. For study purposes of building the risk-based strategy, we considered a point estimate of minimum safe operations costs of approximately 600 million dollars per year (in current dollars). Next we had to determine what the greatest risks were.

Determination of the Highest Risks

Environmental, worker safety, and public health risks were analyzed to determine their impact on cleanup activities. Risk is the potential of a hazard to cause both immediate and long-term harm to a receptor. It is the product of (1)the consequences resulting from a receptor being exposed to a hazard and (2)the likelihood of an occurrence. A graphical representation of the parameters used in the risk analysis is presented in Fig. 1.

Fig. 1

The harm or consequences to a receptor are related to the hazard source in terms of the quantity of material and its toxicity, form and dispersability. The likelihood of release is related to the available energy sources, release processes, and functioning barriers. Pathways by which a receptor can be exposed to hazards are air, soil, groundwater, and surface water. Since risk occurs over time, we considered 1) current state, 2) the remediation phase of cleanup, and 3) the post-closure phase.

We examined four categories of risks over these three periods. The risks were:

Near-term release hazards with potentially large consequences where release of radionuclides and chemical contaminants could occur through the current or remediation phase. Although these hazards have potentially large consequences, they are relatively unlikely.

Work place hazards associated with all aspects of worker activities.

Long-term hazards where harm to the public results from the transport of radionuclides and chemical contaminants through the groundwater slowly over very long time periods (even hundreds or thousands of years).

Ecosystem hazards where harm results from chemical and radionuclide contamination of plant and animal life and from physical disruption of natural habitats. Hazards to cultural, archeological, and historical resources are included in this category.

Near-Term Release Hazards

Conservative unit risk factors were calculated for an average member of the public, assuming current population distribution and also assuming a calculational receptor 500 meters from the source. These factors were based on inhalation exposure and external exposure from material deposited on the ground. All major facilities were evaluated against the

US DOE Nuclear Safety Policy Criteria of 2x10-6 latent cancer fatalities per year for members of the public.

The largest Hanford Site contributors to the risk to an average member of the public are (not in any particular order): the N fuel stored in the K Basins, the cesium and strontium capsules stored in the Waste Encapsulation and Storage Facility, the debris in the 324 B cell, and the high-level waste in the underground storage tanks. The plutonium in the Plutonium Finishing Plant also represents a significant contributor to the public risk, primarily because of the amount of plutonium and its toxicity.

Work Place Hazards

Key sources of work place hazards include the old reactors, canyon buildings, support facilities, Kbasins, and N reactor. During remediation, the key hazard sources include deactivation and final dispositioning of the old facilities, new facility construction, and cleanup actions involving liquid wastes, buried solid wastes, contaminated soils, and special nuclear materials. Exposures to workers are controlled on an individual basis through procedures, training, and protective equipment. Often, managing worker risk is reflected in increased costs.

In general, it is recognized that it is necessary to increase worker risk (and some public risk) in the short term to achieve a reduction in long-term public and environmental risk. A key decision for managers is whether such increases in worker risks are worth the overall benefit in the long term. Some of the most important results of the work place hazard analysis are:

Overall, worker risks are relatively large compared to public risks, perhaps two to four orders of magnitude greater.

Industrial hazards, especially construction of large facilities, are generally greater risk to workers than routine radiological exposures.

Facility transition can be viewed as work place hazard "mortgage reduction" similar to cost mortgage reduction, in that increased worker risk is the price to be paid for longer-term public and environmental risk reduction. Deactivation of retired facilities results in long-term reduction in worker risks. This means cost-mortgage reduction efforts (e.g., transitioning facilities such as PUREX and the Fast Flux Test Facility) provide risk reduction as well as cost reduction.

There is a fundamental tradeoff between leave versus retrieval policies for tank wastes, buried solid wastes, contaminated soil, and liquid disposal sites. Worker risk is generally greater for retrieval activities than for alternatives which treat the waste in place, such as in situ vitrification and disposal.

Entombment has lower worker risk and is preferred over demolition of old canyon and reactor facilities.

Long-term Release Hazards

The objective of remediation of much of Hanford's wastes is to prevent long-term releases to the public and the environment that exceed acceptable levels. The principal exposure pathway of concern is public consumption of groundwater and contamination of the Columbia River. In the past, airborne and surface water exposure routes were significant, but these have declined substantially with the end of the production mission.

Our study focused on those contaminants of greatest concern, both hazardous chemicals and radionuclides, and three exposure effects: 1)

excess carcinogenic incidence from exposure to radionuclides; 2) carcinogenic incidence from exposure to hazardous chemicals; and 3) non-carcinogenic adverse health effects from hazardous chemicals. Unit exposure factors were calculated for a residential scenario with exposure only through the use of contaminated groundwater. Carcinogenic effects were determined assuming exposure to the peak concentration over the lifetime of the exposed individual.

The summary results of the analysis of long-term risks are:

Long-term release hazards through the groundwater pathway are dominated by seven long-lived, mobile constituents: C-14, Tc-99, I-129, Np-237, carbon tetrachloride, nitrate, and uranium.

Potential contributions from tank waste, buried solid wastes, and existing environmental releases are roughly of the same order of magnitude.

The long-term release hazard posed by existing low-level waste burial grounds is roughly two orders of magnitude higher than that posed by the pre-1970 unsegregated buried waste and the post-1970 retrievably stored transuranic waste. The policy of retrieval of transuranic and suspect transuranic wastes is questioned.

For tank waste, roughly one-fourth of the tanks contribute more than 99% of the long-term release risk. Also, the residual contamination following retrieval and the contamination from prior tank leaks place a floor below which risk cannot be reduced without some form of in situ treatment.

Ecosystem Hazards

We reviewed prior studies and analyses for results that would be applicable to the development of a risk-based strategy. We learned that the isolation of Hanford and its restricted access has resulted in a habitat that is relatively undisturbed. However, the potential for adverse ecosystem impacts from remediation activities is substantial, especially from physical disruption of the ecosystem by soil and waste evacuation and construction of new facilities. Like worker risk, there are tradeoffs for some environmental management activities between near-term damage to the environment and longer-term reductions in public risk.

Elements of a Risk-Based Strategy

Considering the risks described above, we developed a cleanup strategy based upon a single objective: carry out those activities which are the most effective from a risk-reduction perspective, while ensuring the risks are reduced to acceptable levels for the public, workers, and environment. Four considerations are primary: 1) near-term releases which pose an unacceptable risk to the public, workers, and ecosystem will be prevented; 2) workers will be protected from unacceptable hazards; 3) the ecosystem and cultural resources will be protected during remediation; and 4) the Site will minimize long-term releases that might affect the public and ecosystem.

The key elements of the Risk Strategy are:

All reactors would be entombed as permanent disposal sites.

Liquid and solid waste sites outside the central plateau would be capped, covered, and disposed in place.

The central plateau would be a permanent waste disposal site for entombed nuclear facilities, liquid and solid waste disposal sites (capped and covered), and pre- and post-1970 transuranic waste and tank waste (both of which would be disposed in-place). Access to the central

plateau would be controlled indefinitely, and access to groundwater would be controlled for the foreseeable future.

Spent nuclear fuel and special nuclear materials would be removed from current locations and placed in interim storage in the central plateau, pending final disposition.

Facilities outside the central plateau would be dismantled, as appropriate, or consolidated to reduce operations and maintenance costs.

Groundwater would not be treated, and public access would be controlled.

Execution of a Risk-Based Strategy

Execution of this strategy would result in 1) nuclear material consolidated and stored on-site pending final disposal, 2) liquid tank waste disposed in-situ, 3) solid waste disposed in-situ, and 4)major facilities entombed in place. This strategy would leave the environmental contamination in place, but capped and covered, as necessary.

The costs to implement the Risk Strategy would be considerably lower than the costs to implement the current baseline strategy because much of the waste would be disposed in-place. Assuming a 10 billion dollar in-situ cost of disposal of tank waste and nine billion dollars for environmental restoration, the total cleanup cost is estimated at 44 billion dollars--saving perhaps 20 to 30 billion dollars as compared to the current strategy. Figure 2 illustrates the funding profile over the next 75 years.

Recall that a strategy must say what will be done, when it will be done, and what the result will be. Figure 2 addresses the second part of the strategy. Adoption of the risk-based strategy described above means that first priority would be given to preventing near-term, potentially catastrophic releases. Thus, removing the spent fuel from K-Basins and stabilizing underground storage tanks would be given high priority. So would cleaning out the 324 building B cell, cleaning out the Plutonium Finishing Plant, and moving the cesium and strontium capsules to a more stable storage as soon as practical.

Second priority would be given to mortgage-reduction efforts which also reduce worker risk. Facility transition of PUREX, N reactor, Fast Flux Test Facility, and others would be funded as soon as possible.

Fig. 2

As money becomes available from the first two priorities, priority would be given to tank waste disposal, solid waste disposal, and, especially, environmental restoration work. Technology for groundwater treatment would be funded, but not current pump and treat activities.

Of course, all support activities needed to execute these programs, e.g., landlord, would be funded at those levels needed to carry out the above functions.

Adoption of a Risk-Based Strategy

Adoption of a risk-based strategy like the one described in this paper (and it is only one of several strategies which could be considered "risk-based") would require regulatory modification and relief, including renegotiation of the Tri-Party Agreement. It would significantly impact the ultimate use of the land and would considerably increase the waste left in the ground at Hanford. Many of Hanford's stakeholders, especially the Environmental Protection Agency and the Washington State Department of Ecology may oppose such a strategy, and certainly it could not be adopted without their involvement and approval.

Despite local opposition, however, the Department of Energy and Congress may support a risk-based strategy because it is financially affordable while still addressing the highest risks. But if it is to be adopted, the DOE and its regulators should begin negotiations now, before an increasingly skeptical Congress reduces funding and limits options. It is also important for DOE Headquarters to understand the significant implications of adopting risk-based (vice compliance-based) strategies for the Complex. While it sounds good to Congress and others, our study indicates that adoption of a risk-based approach to cleanup is a major change from the current way of doing business.

Session 52 -- SELECTION OF DISPOSAL METHODS AND DISPOSAL SITES FOR LOW-LEVEL, INTERMEDIATE-LEVEL AND TRU WASTES
Co-chairs: Ted McIntosh, USDOE;
Kathryn Tominey, Battelle-PNL
52-1
THE NATIONAL ACADEMY OF SCIENCES REVIEW OF THE PROPOSED LOW-LEVEL RADIOACTIVE WASTE DISPOSAL SITE AT WARD VALLEY, CALIFORNIA - PART II

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ABSTRACT
The National Academy of Sciences' review of geological issues related to the technical suitability of the proposed Ward Valley, California site for a low-level radioactive waste disposal facility has important, broadly applicable generic lessons for situations where a scientific panel is on the critical path for a specific project which is governed by a body of existing law and regulation. This paper will describe these generic lessons and offer recommendations for avoiding pitfalls at the interface of science and public policy to the end that the proper role of independent scientific panels in controversial siting cases may be better defined and implemented.

INTRODUCTION

In a paper presented last year at Waste Management '95, we reported on the National Academy of Sciences/National Research Council's (NAS) review of the proposed low-level radioactive waste (LLRW) disposal site at Ward Valley, California. ("The National Academy of Sciences as 'Science Court': Review of the Ward Valley Site by the Board on Radioactive Waste Management.") In March, 1994, Secretary of the Interior Bruce Babbitt requested the Board on Radioactive Waste Management (BRWM) of the National Research Council to review the "earth sciences concerns" in a critical report on Ward Valley written by three geologists, the so-called "Wilshire Report." (The scope of the review was later expanded to add two environmental issues, revegetation and protection of the desert tortoise, a threatened species). Release of the Academy's report was delayed until May, 1995, several months after Waste Management '95. Therefore, last year's paper was limited to discussions of the scope of the NAS study, the committee's procedures, political antecedents to the Secretary's request, and several issues raised by the Academy's review procedures

including charges of bias made by Ward Valley opponents about the composition of the review committee and the Academy's response to those charges. Other issues identified were uncertainty as to the role of the U.S. Nuclear Regulatory Commission's regulations in the deliberations of the committee and the Secretary's failure to say how the Academy report would be used in the federal administration's decision-making on the State of California's request to purchase 1,000 acres at Ward Valley from the U.S. Bureau of Land Management for use as the site of the Southwestern Compact's regional LLRW disposal facility. It was also noted that the Academy's review process had encouraged the U.S. Geological Survey to accelerate completion of a report on characteristics of the unsaturated zones at Beatty, Nevada and Ward Valley and had stimulated the U.S. Nuclear Regulatory Commission (USNRC) and the Idaho National Engineering Laboratory to review estimates of the amounts of plutonium expected to be disposed of at Ward Valley over its thirty-year operational life. (The new estimates are considerably lower than earlier estimates and also considerably less than experience at old disposal facilities.)

HIGHLIGHTS OF THE ACADEMY REPORT

With the release of the NAS report on May 11, 1995, ("Ward Valley -- An Examination of Seven Issues in Earth Sciences and Ecology"), it is now possible to summarize and analyze its findings and to comment on the role of the report in the ongoing Ward Valley land sale (federal government to the State of California) and project development processes. The NAS report lays to rest charges by Ward Valley opponents that the disposal facility will pollute the Colorado River. ("...we believe it highly unlikely that significant amounts or radioactive material from the site would reach the ground water or the Colorado River.") The findings of the Academy report are consistent with decisions made in September, 1993 by the California Department of Health Services to certify the Environmental Impact Report and to issue a license to US Ecology, Inc. to construct and operate a LLRW disposal facility at Ward Valley.

At the news conference in Washington, DC at which the Academy report was released, George A. Thompson, Professor of Geophysics at Stanford University and Chair of the NAS Committee to Review Specific Scientific and Technical Aspects Related to the Ward Valley, California Low-Level Radioactive Waste Site made an opening statement which included the following summary conclusions.

"...the committee concluded that the potential transfer of contaminants through the unsaturated zone to the water table is highly unlikely. In the report we discuss a number of aspects of this potential transfer, and in none of these cases could we find a likely mechanism that would allow contaminants to reach the water table." "...even if all the plutonium at Ward Valley reached the river at the same rate it was disposed of, effects on the river-water quality would be insignificant relative to the background radiation levels currently found in the river. In other words, it would be well below accepted regulatory health standards. And I would emphasize again that we believe it highly unlikely that significant amounts of radioactive material from the site would reach the ground water or the Colorado River."

In responding to charges of committee bias in the selection and composition of the committee, Professor Thompson noted:

"In the past year, we have come under fire -- indirectly in most cases and directly in some -- for having conflicts of interest on the issue of

whether or not low-level nuclear waste should be stored at Ward Valley. It is certainly very true that most of us have been called upon in the past for advice pertaining to the implications of geologic processes for the safety of nuclear power plants and the disposal of radioactive wastes. For this very reason -- our scientific knowledge and experience with these issues -- we were asked to participate in the study. In order to properly evaluate the geologic and hydrologic issues under question at Ward Valley, it was immediately apparent that scientists with expertise on these specific issues would be needed.

"The committee was assembled by the National Research Council after having received names of experts from a range of organizations, including the Committee to Bridge the Gap, a public-interest organization that opposes the low-level waste site at Ward Valley. I should also note that the Research Council did not solicit suggestions for committee members from industry or organizations in favor of siting the waste facility at Ward Valley.

"Having said that, most of us at one time or another have received research funding from the Department of Energy, which as you know, was not the sponsor of this study, but is the primary agency for administering the federal government's nuclear weapons facilities. And several of us at one time or another have conducted or reviewed studies for the Nuclear Regulatory Commission or have worked for private industry on nuclear power and radioactive waste issues. One would be hard pressed, in fact, to find an expert on radioactive waste, or on hydrology and geology issues surrounding radionuclides, who has not lent their scientific expertise to industry or to government agencies."

In his summary statement, Professor Thompson touched on the fact that two of the seventeen members of the committee filed dissenting points of view in the report appendices:

"Two committee members dissented from the general conclusion regarding the potential transfer of contaminants through the unsaturated zone. One of them, Professor Oberdorfer, is here with us today. Their points of view can be found in Appendices E and F in the report. They emphasize the remaining uncertainties about the movement of water through the unsaturated zone, and as a result they were not willing to judge the likelihood of contaminants reaching the ground water as highly unlikely. In addition, the second dissenter, Dr. Mifflin, took a somewhat different view on the issue of revegetation at the site, which I'll discuss in a moment. On the other points in the report, these two members joined with the committee consensus."a

Professor Thompson noted that one of the committee's recommendations is for "ongoing data collection and continued evaluation." Some Ward Valley opponents have seized upon this recommendation to argue that additional data must be taken prior to transfer of the Ward Valley site to the State of California. But this was not the view of the committee majority:

"The entire committee agrees that additional analysis needs to be done. The majority of the committee believes that this could be done during construction and operation of the site. Ongoing review of the site, we feel, would build scientific assurance, credibility and public confidence in the monitoring program."

STATUS OF THE WARD VALLEY LAND SALE AND POSSIBLE CONGRESSIONAL ACTION

As of February, 1996, the Interior Department has not yet sold the Ward Valley land to the State of California. Last May, Interior Secretary Babbitt announced his intention to go ahead with the land sale -- but

with conditions. Some, but not all, of his proposed conditions are based on recommendations of the NAS report. A summary of those recommendations and of California's responses to them is attached to this paper as an appendix. California Governor Pete Wilson has objected to the Secretary's proposed conditions as intruding upon California's regulatory authority. In particular, Interior has insisted that it have a perpetual enforcement role with respect to the conditions and has demanded that the conditions be enforceable in court. California has argued that such a role for Interior is not appropriate for two reasons: one, USNRC, not the Department of the Interior, is the responsible federal agency which exercises oversight of Agreement State radiological health and safety programs, and, two, Interior lacks the necessary expertise. The state has also asserted that, as a regulator, it has available administrative remedies sufficient to assure safe operation of the disposal facility without resorting to litigation. Interior's response in an October, 1995 press statement was that "...the agreement must be binding and enforceable..." because "trust us will not suffice..."

The U.S. Department of Energy, responsible under the Act for providing technical assistance to the states for development of LLRW disposal facilities, has taken a very different view than that of the Department of Interior with respect to the state and federal roles in assuring safe disposal of LLRW. When requested by California Senator Barbara Boxer to assign a National Laboratory to undertake soil tests recommended by the NAS, Energy Secretary 'Leary declined unless the state decides it needs, and requests, analytical services "unique to the Department of Energy." "We believe the State of California, in its licensing role as authorized by the Nuclear Regulatory Commission, should determine how to implement the National Academy of Sciences' recommendation." The Academy report recommended that additional testing be done during the operational life of the disposal facility. Senator Boxer asked that the testing be done before sale of the Ward Valley land to the State.

Following release of the Academy report and Secretary Babbitt's decision to impose conditions on the land sale, negotiations between the California Department of Health Services and the Department of Interior took place. By September, it was clear that the state and federal governments were at an impasse. In September, California Governor Pete Wilson asked Congress for relief. In letters to Senator Frank Murkowski (R-Alaska), Chairman of the Senate Energy and Natural Resources Committee, and to Congressman Don Young (R-Alaska), Chairman of the House Resources Committee, Governor Wilson asked that Congress convey the Ward Valley site to California by statute. The Governor's request was considered and approved by both policy committees and a Ward Valley conveyance provision was included in the Budget Reconciliation Act approved by the Congress but vetoed by the President in December, 1995. The President's veto message includes a reference to Ward Valley which says that legislative conveyance of the land would be "without public safeguards." The President has been misinformed. All requirements of federal and state regulations will apply to disposal of low-level waste at Ward Valley along with the state's 130 license conditions. The fate of both the federal budget and the Ward Valley conveyance provision are uncertain as of February, 1996.

OBSERVATIONS

The Academy's Ward Valley review was on the project critical path for fifteen months from March, 1994 to May, 1995. While the NAS report was

pending, it was clear that the federal government would not act to sell the Ward Valley site to California. Indeed, a reasonable political interpretation of events is that the report was requested because the administration wanted to delay a decision on the State of California's application to purchase the land for the site. In a sense, the Administration's insistence that NAS "recommendations" be elevated to the equivalent of regulations (along with additional conditions invented by Interior), enforceable in the courts by Interior and perhaps by third parties as well, has kept the NAS review on the critical path delaying administrative action by the federal government to sell the land to the state.

At the time the NAS accepted Secretary Babbitt's request for a review of the issues in the Wilshire report, a license for the Ward Valley project had already been issued by the legally responsible state agency. Furthermore, the license and EIR certification were in litigation in state court. It appears that the Academy was not aware of these facts regarding the status of the project, and, equally disturbing from the point of view of predictable process, it is not clear that it would have made any difference to the Academy had it known this status. Litigation on the license and EIR certification was completed on January 18, 1996 when the California Supreme Court denied opponents' petition for review of an Appellate Court decision (October, 1995) upholding the license and rejecting all opponents' claims.

Both the California Department of Health Services and Cal Rad Forum urged the Academy's Ward Valley review committee to utilize the USNRC's regulations for LLRW disposal (Title 10 Part 61 of the Code of Federal Regulations) as the framework for their inquiry. This recommendation drew a chilly reception. When this same recommendation was made on December 15, 1994 to the BRWM at a meeting in Irvine, California, the response was that the Board does science, not regulations. It is only one of several ironies associated with the federal government's handling of the Ward Valley land transfer that the Department of the Interior has attempted to elevate the Academy's recommendations to the status of regulations. By entering a politically controversial siting case late in the game, and utilizing a review framework different from the legally applicable standards applied during license application and judicial review, the independent scientific advisory panel placed the licensee, other project proponents, and the state regulatory agency in a difficult situation. Under these circumstances, the project was subject to two, possibly different, standards of review, one established by statute and regulation, the other established ad hoc by the scientific panel.

Some have argued that independent scientific review can enhance public confidence in the scientific validity of a controversial technical project. However, this benefit can be limited in situations where the scientific issue of safety is not what really motivates opposition. It is an observation of the authors that opposition to LLRW disposal is often really a surrogate for opposition to the use of radioactive materials.

RECOMMENDATIONS

Independent scientific panels can play an important role at the intersection of science and public policy. This role is more appropriately undertaken early in a process when public policy is being formulated generically and regulatory standards to implement that policy are being considered, or in early review and evaluation of data, rather than later, after legal regulatory standards have been adopted and

applied in specific cases of license application review. Certainly, to undertake independent scientific review after a license has been issued and is in judicial review is to risk serving the promoters of delay. Independent scientific groups should be wary of situations where they may find their review on the critical path of a specific project or where their work effectively constitutes an appellate review competing with the courts.

With respect to specific projects, and from a structural point of view, independent scientific advisory panels can work most effectively if they are advisory to the license applicant or to the regulatory agency rather than to agencies outside of the legally established regulatory framework. This will maintain political accountability with those organizations to whom responsibility and authority have been assigned by law.

While it is often argued that additional scientific review will allay the fears of the public and political leaders in controversial decisions, science can only inform, but it cannot make, decisions legally reserved to the political leaders. A surfeit of science is no substitute for political will.

When called upon to resolve controversy that is essentially political, scientific groups would do well to heed the advice of Hippocrates (Epidemics, Book 1, Chapter 11): "First, do no harm."

Summary of California's Response to National Academy of Sciences' recommendations

Recommendation: Install monitoring stations in both the unsaturated zone and saturated zones to supplement the site characterization database as well as to expand the baseline data for operational and post-closure monitoring.

Response: The state plans for operational and post-closure monitoring programs are consistent with this recommendation.

Recommendation: Make Cl-36 determinations and add three groundwater monitoring wells (for a total of eleven) to planned monitoring program.

Response: The recommendation will be incorporated into programs for monitoring for water content, water potential, tritium, groundwater levels, vertical hydraulic gradient, groundwater quality, etc.

Recommendation: Avoid conditions that could cause lateral flow in and surrounding the disposal trenches.

Response: The state has already required US Ecology to remove shallow caliche layers under each of the perimeter berms during construction in order to disrupt the lateral continuity of the caliche layers and prevent lateral flow.

Recommendation: Re-initiate formal consultation with the U.S. Fish and Wildlife Service on the desert tortoise.

Response: Such formal consultation has already begun and will be completed within a few weeks.

Recommendations: The NAS Committee made several recommendations concerning investigational and action levels for unsaturated zone monitoring.

Response: An already established license condition requires the licensee to develop investigation and action levels for the unsaturated and saturated zone monitoring programs, and the Environmental Monitoring Report establishes the investigation and action levels for tritium. Operational procedures approved by the state address potential accidental releases and appropriate remedies during the operation of the disposal facility. Performance assessments of the postclosure period indicate that

off-the-shelf remediation plans are not needed, rather, there is sufficient opportunity to develop and implement remedial measures specifically for any actual, post-closure release of radionuclides, should an unlikely event occur. These existing and proposed requirements fully address the committee's recommendations in this area.
Recommendations: Give significant emphasis to the design and performance testing of the trench cover demonstration units and the revegetation program.
Response: The state agrees that significant emphasis be given to the trench cover demonstration program, and many of the specific recommendations of the committee, such as testing of irrigated and non-irrigated areas, are already in the state plans. However, the state opposes a general research and development program (and an increase in the number of trench cover demonstration units) because of the increased amount of land that would be impacted by such a program.
Recommendation: Establish an independent third-party review group as a means of increasing the level of public confidence in the program.
Response: The process already laid out provides ample opportunity for all interested and affected parties, as well as the general public, to participate in an open and timely manner. The state's periodic evaluations of the site monitoring data will be publicly available. The state will continue to seek independent peer review on specific issues or concerns, including those mentioned by the NAS committee.
Recommendation: Monitor the stability of the site surface, specifically for erosion, ponding, and any subsidence of trench covers, and take appropriate maintenance actions.
Response: The state has always planned to take such actions, with monitoring to continue for at least 100 years after the site is closed, and the state will continue the monitoring and maintenance activities for as long as there is an observed need for such activities, even beyond the 100 year period if necessary.
Recommendation: Incorporate an engineered slope and lined channel for conveying storm water around the west, north, and south sides and corners of the flood protection berm.
Response: The state prefers to allow any infrequent runoff to spread across the existing, shallow alluvial surface for removal by evapotranspiration rather than to create and area of concentrated flow and potentially concentrated infiltration, as the synthetic liner would fail over time.
Reprinted from Straight Talk About Low-Level Radioactive Waste for June, 1995. Published by Cal Rad Forum. Nicki Hobson, Editor.

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SIGNIFICANT PROGRESS TOWARDS DEVELOPMENT OF THE LOW-LEVEL RADIOACTIVE WASTE
DISPOSAL FACILITY IN ILLINOIS
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ABSTRACT
Development of disposal sites for low-level radioactive waste is a complicated legal, regulatory and public sector process. Development of the low-level radioactive waste disposal facility to support generators in Illinois and Kentucky is well under way. Significant progress has been made to re-engineer the siting development process capitalizing on prior lessons learned and a recommitment from Illinois state leadership assuring the future success of the program. Comparisons of why this new process will succeed are the major focus of this paper. Specific changes in approach from the previous process including changes in the Illinois Management Act (Management Act), creation of the Illinois Low-Level Radioactive Waste Siting Task Group (Task Group), new roles for the Illinois State Geologic Survey and Illinois State Water Survey (Scientific Surveys) and the Illinois Department of Nuclear Safety (IDNS), a new contractor reliance approach and increased confidence on the "science" are the major contrasts between the previous process and the new process currently underway.

INTRODUCTION

Prior to 1980, all nondefense low-level radioactive waste (LLRW) generated in the U.S. was shipped for disposal to six commercial LLRW facilities, including the LLRW facility near Sheffield, Illinois. By 1979, only three of these disposal facilities were still in operation. The facility near Sheffield was closed in 1978 following a refusal by the NRC to grant a license for an expansion of the site. The remaining facilities were located in Nevada, Washington and South Carolina. Following a series of packaging and transportation incidents at each of the sites, the governors of those states took steps to reduce the volume of waste coming into their respective states and put pressure on Congress to create a more equitable system for disposal of national LLRW.

In response, Congress passed the Low-Level Radioactive Waste Policy Act (Policy Act) of 1980. This act established a federal policy that each state should be responsible for providing disposal of commercial LLRW generated within its borders by 1986. The Policy Act also encouraged the formation of interstate compacts to manage LLRW on a regional basis.

In April 1980, then Governor James Thompson signed an executive order creating the Illinois Department of Nuclear Safety (IDNS). The mission of this newly created department was to protect the public health and safety of the Illinois citizens from the potential hazards of ionizing radiation.

By 1985, it was apparent that no state or compact would be able to meet the 1986 deadline for LLRW disposal, so Congress passed the Low-Level Radioactive Waste Policy Amendments Act (Amendments Act) of 1985. This new act extended the deadline for providing final LLRW disposal to 1996 with interim disposal provided by 1993. The act further established a system of milestones and penalties for the states and compact to meet.

Between 1984 and December of 1992, the Department followed a LLRW management system which included a siting process, public participation plans, community benefit plans, research studies to develop siting criteria, contractor qualification and hiring process, and development of licensing regulations compatible with federal guidelines. The reasons the first process failed was that the initial process was bounded by weaknesses in the Illinois Management Act, dependance on the Department to serve as both site promoter and site regulator, and attempts by the Department to accelerate selection and licensing of the Martinsville

site. As determined by the Illinois Low-Level Radioactive Waste Disposal Facility Siting Commission (Siting Commission) after its 72 days of hearings, the Department's rush to judgement suggested that politics appeared to take the lead over science in the selection of the site. A thorough evaluation of the failed process would need to be undertaken and changes made for any new siting process to succeed.

From January 1992 until late summer 1993, the Department along with its contractor, and major waste generators reviewed siting processes being developed and siting activities underway in other host states nation-wide. Out of this review grew the newly re-engineered and redefined process being implemented today.

THE ORIGINAL PROCESS

The Illinois Low-Level Radioactive Waste Management Act of 1983, as amended, made the Illinois Department of Nuclear Safety responsible for selecting a site for disposal of LLRW in conjunction with licensing a disposal facility. The Department had developed a program and adopted a schedule to allow the new regional LLRW disposal facility to begin receiving LLRW by 1993. To fulfill this mandate the Department, as shown on the Process Flowchart, Fig. 1, planned to locate several large candidate areas with technically excellent features for disposal of LLRW. To identify from these areas several potential alternative sites. Designate from the alternative sites, a couple of sites for detailed geotechnical characterization and environmental study. Evaluate the results of these investigations and select one site for the development of the disposal facility. The selection of a final site had to meet all of the following criteria as set forth in the Management Act:

Fig. 1

1) The site shall be located so that the public health, safety, and welfare will be protected.

2) The site shall be located in a suitable geologic and hydrological medium.

3) The site shall be located so as to minimize the possibility of radioactive releases into groundwater utilized as public water supplies.

4) The site shall be located outside the boundary of the 100 year flood plane as determined by the Department of Transportation.

5) The site shall be located so as to consider the distance necessary for transportation of low-level wastes and so that the impact on existing traffic flows is minimized.

6) No low-level radioactive waste disposal facility shall be located in or within 1 1/2 miles of the boundaries of any municipality unless approval is given by the governing body of that municipally.

7) No low-level radioactive waste disposal facility shall be located in an area of a county situated more than 1 1/2 miles beyond the boundaries of a municipality unless approval is given by the governing body of that county.

The Department had virtually unlimited authority to pursue this process. It could choose its contractors for the detailed studies, it could choose its siting criteria, it could ultimately choose its own site. As this process began to bog down in technical disputes and semantics, it became apparent the original time table would not be met. The acceleration of the entire process to stay on schedule may have been responsible for untimely management decisions, which, when viewed externally from the process, appeared to call the entire siting process into question.

Following an investigation by the Illinois Senate Executive Committee of the LLRW siting process in the spring of 1990, the Illinois General Assembly amended the Management Act to create the Illinois Low-Level Radioactive Waste Disposal Facility Siting Commission (Siting Commission). The commissions directive was to evaluate the safety and suitability of any site proposed by the director of the Department of Nuclear Safety. In August 1990, the Siting Commission held its first meeting. On October 9, 1992 after 72 days of public hearings and volumes of testimony by all parties involved, the Siting Commission issued its final order relative to Martinsville Alternative Site, "...the Commission finds and determines that the site with respect to a facility of the proposed design does not meet all the criteria set forth in subsection (b) of Section 12 of the Management Act by a preponderance of the evidence." The looseness of the Management Act had allowed the process to become very disjointed, leading to its final end with the Martinville hearings, and Siting Commission ruling.

THE NEW PROCESS

During the spring Legislative session in 1993, significant changes were made to the Illinois Management Act. Among these important changes the establishment of the Illinois Low-Level Radioactive Waste Siting Task Group (Task Group) as well as doing away with original seven criteria. The Management Act as amended (August 1993) authorized a second site-selection process. Responsibility for the initial part of the process, up to site characterization and licensing, was assigned to the following entities: the Task Group, the Illinois State Geological Survey and the Illinois State Water Survey, and the contractor chosen by the Department. This new process set forth decreet steps to follow to ensure a successful conclusion. The Illinois LLRW Siting Process flowchart, Fig. 2, depicts how these steps during this endeavor will proceed.

The Task Group was mandated to assume responsibility for two critical steps:

Fig. 2

1) Development of site-selection criteria for the LLRW disposal facility and

2) Determination of whether three proposed sites meet the site-selection criteria.

The Management Act emphasized the importance of science above other considerations in the siting process by requiring (1) that four of the nine members of the Task Group "shall have expertise in geology, hydrogeology, or hydrology" and (2) that the "[p]rinciple criteria shall relate to the geographic, geologic, seismologic, tectonic, hydrologic and other scientific conditions best suited" to an LLRW disposal facility. The Act also directs the Task Group to consider supplemental criteria, such as "land use, economic, transportation, and any other matter identified by the Task Group as relating to desirable conditions for an LLRW disposal facility."

It is important that the roles of the entities involved in the site-selection process not be confused. In particular, the role of the Task Group in developing criteria for screening the state and narrowing the available choices should carefully distinguished from the role of the Department in licensing the disposal facility and regulating its operation and closure.

Once the Task Group adopts its final set of criteria, the Governor appoints a replacement for the Director of the Illinois Department of

Nuclear Safety who is now removed from the Task Group. These final criteria are then turned over to the Scientific Surveys who apply them to screen the entire state and identify at least 10 locations of at least 640 acres each that appear likely to meet the siting criteria. During this process any volunteer locations presented will be included for evaluation. A written report will be issued to the Task Group identifying the locations within the state.

After the Scientific Surveys identify at least ten or more locations, this list is then passed to the facility contractor, Chem-Nuclear Systems, Inc. (CNSI) who will evaluate the locations and winnow the list to three promising sites. During this process limited intrusive field investigations will be conducted when required. Once three site are identified, the contractor will prepare a report and the Task Group will hold public meetings to discuss the issue of whether the three sites meet the siting criteria. Once concurrence and approval is given by the Task Group that the three sites indeed meet the criteria, they will be passed back to the contractor who will select one site for final evaluation, and will notify the Task Group of the Site selected. At this time, the Illinois LLRW Siting Task Group is abolished, having finished their task. CNSI will then conduct intensive field investigations and site characterization of the selected site. Concurrently, CNSI will prepare a license application for the site. Once this phase draws to a close the contractor will submit to the Illinois Department of Nuclear Safety, Licensing Division its license application for review.

The Department will review the application following the sequences shown in the license application review process, Fig. 3. Once all licensing concerns are addressed, a Notice of Intent to Issue License will be given. The process will now proceed in one of two avenues. First, if there is an objection to the issuance of a license then a hearing will be conducted. During this process a hearing officer will be appointed to preside over an adjudicatory hearing where the burden of proof is on the objector. The hearing officer will recommend to the Director of the Department whether or not a license should be granted. Second, if no objection is forth coming, the Department will issue a license for the construction. During the construction, the Department inspects all phases of work for conformity with the license application. Once construction is completed, the Department will amend the license to authorize receipt and disposal of waste. Where upon LLRW disposal operations begin.

Fig. 3

CONCLUSIONS

Since 1980 a great deal has been accomplished toward successfully siting a LLRW facility in Illinois. The first process failed for several reasons including the Siting Commissions perception that politics and the need for local approval overrode the scientific investigations into site suitability; the Department's dual role of site promoter and regulator; and the apparent rush to meet federally mandated deadlines. A relational chart of the New process and the Original process depicting similar siting activities is shown in Fig. 4. As determined by the Siting Commission after its 70 plus days of hearings, the Departments rush to judgement appeared to caused politics to take the lead over science in selection of the site.

Fig. 4

Illinois, after the failed first siting process and the experience of the Siting Commission, evaluated the short comings of its first process and

set in motion a comprehensive program to ensure the ultimate success for LLRW disposal. This process includes a definitive "road map" for the siting activities. Eliminates the Department from the roles of site promoter; and focuses the Department on its role as regulator. This process also places specific emphasis on scientific discipline, while allowing private enterprise to develop a sound working philosophy for the safe management of low-level radioactive waste disposal in the state of Illinois.

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THE NEW PROCESS FOR SELECTION OF A SITE FOR LOW-LEVEL RADIOACTIVE WASTE DISPOSAL FOR ILLINOIS

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ABSTRACT

Chem-Nuclear Systems, Inc. and its technical contractors Golder Associates, MK-Company, and Hanson Engineers have developed a new process for selection of a Low-Level Radioactive Waste Disposal Facility in Illinois. Regulatory framework for the new process are contained in changes to the Illinois Management Act approved by the Governor in 1993. These changes created the Illinois Task Group which adopts criteria, assign's the Illinois Scientific Surveys the responsibility to initially screen the state, gives Chem-Nuclear responsibility to assess locations in Illinois against criteria and select and characterize a site to serve the generators in Illinois and Kentucky for the next 50 years, and allows the Illinois Department of Nuclear Safety to resume its mandated role as regulator of Illinois generators, users, and disposers. The new process concedes the old process to history and starts afresh with many new concepts. Putting the site selection, characterization and licensing responsibility totally in Chem-Nuclear hands allows the company and its contractors to develop an integrated process which will use newly proven technical and analytical processes and will use performance assessment to guide the siting decision process and support the licensing process. Unlike the previous effort to site a facility in Illinois, the new process is clear, unambiguous and explicit in its intent to have science rather than politics as the driver and to incorporate lessons learned from the previous siting effort. This paper describes the new process from Management Act amendment to selection of the site by Chem-Nuclear.

INTRODUCTION

The State of Illinois has dramatically improved the methodology that will be employed to select the site for development of a low-level radioactive waste disposal facility to serve the disposal needs of Illinois and Kentucky. The site selection process developed by Chem-Nuclear Systems, Inc. (Chem-Nuclear), and its project team of Golder Associates Inc., Hanson Engineers Inc., and Morrison Knudsen Corporation is founded on the positive aspects of the lessons learned from the previous siting effort. The basis for the new process will be the utilization of performance assessment modeling and decision analysis methodology to perform Value-of-Information analyses during its five-step site selection process. Use

of performance assessment modeling and decision analysis methodology will lead to a more focused management and refined utilization of resources dedicated to field investigations that will be used in selection of the site for the Illinois Low-Level Radioactive Waste Disposal Facility (ILLRWDF).

BACKGROUND

The site selection process Illinois employed in the late 1980s, during its first effort to identify a site for the development of the ILLRWDF, was a process based on acceptance of the site by the unit of local government (either a municipality if within 1.5 miles or the county if not located within 1.5 miles of a municipality). This process lead to the identification of the Martinsville Alternative Site. The site selected was ultimately eliminated from consideration after the Siting Commission reviewed the site selection process and the technical merits of the MAS and found them to be unacceptable. As a result of this first siting effort, the Illinois Legislature, through the Illinois Department of Nuclear Safety, established a revised siting process to identify a site suitable for the ILLRWDF.

The revised site selection process is substantially different from the previous one that resulted in the selection of the MAS. The major changes embodied in the new process:

Establish a Task Group to develop site selection criteria;

Divide the siting process into two distinct phases;

Removes acceptance of the site by local government; and,

Establishes a process allowing for the acceptance and evaluation of volunteered lands.

Task Group - Formed by the amendment of the Management Act, the Task Group consists of nine voting members, three of whom are State officials (the Directors of the Illinois Department of Nuclear Safety, the Illinois Environmental Protection Agency and the Illinois Department of Natural Resources). Four of the additional members have expertise in the fields of geology, hydrogeology or hydrology. Of the two remaining additional members, one is a member of the public with environmental experience and the other has at least five years experience in local government. In addition, there is one non-voting member.

The Task Group has been given the responsibility to accomplish several important tasks including:

Development (including public review and comment and holding public hearings) of the site selection criteria;

Acceptance of the locations identified by the Scientific Surveys; and,

Conducting meetings on, and approval of the three sites identified by Chem-Nuclear;

After identification of one site for characterization, the Task Group will be abolished.

Site Selection Process - The site selection process will be conducted in two distinct phases. The first, conducted by the Scientific Surveys, will involve Statewide screening. The Surveys will conduct a broad screening of the State of Illinois using the criteria developed by the Task Group. Additionally, the Scientific Surveys will evaluate all lands volunteered against the site selection criteria. The result of the Statewide screening will be the identification of at least ten locations, each of at least 640 acres in size. The results of the Statewide screening and evaluation of volunteered locations will be published in a report and submitted to the Task Group.

The second phase of the site selection process will be conducted by the Chem-Nuclear team. After identification of at least ten locations by the Scientific Surveys, Chem-Nuclear will conduct evaluations, including possible field investigations, of the locations. Upon completion of these evaluations, Chem-Nuclear will select three sites, of at least 640 acres that appear promising for development of the ILLRWDF. In selecting the three sites, Chem-Nuclear will give preference to sites in Locations that were volunteered, unless those sites are clearly less promising than sites in other locations. Chem-Nuclear will then prepare a report fully describing the siting evaluations and submit the report to the Task Group.

Upon receipt of the siting report, the Task Group will conduct public meetings on three sites selected. At the public meetings, Chem-Nuclear will present the results of the siting evaluations. Also during the meetings, the Task Group will receive oral or written information from the public regarding the sites under consideration. Following the meetings, the Task Group will determine if the three sites satisfy the site selection criteria adopted by the Task Group.

Following the Task Group's decision that the three sites meet the site selection criteria, Chem-Nuclear will select one of the three sites for characterization and notify the Task Group of the site selected. Upon receipt of the notification of a site selected, the Task Group will be abolished and all of its records transferred to IDNS.

Public Acceptance - The new site selection process does not require that formal acceptance by the public nor local governing bodies be obtained prior to selecting a site. The new process is based on selecting the premise that site selection should be driven by the technical merits of the sites rather than by the public's or government's acceptance of the sites.

Volunteered Lands - During the new siting process, lands that are volunteered by land owners or units of local government will be given selection preference unless they are clearly less promising than other lands identified as meeting the Task Group's site selection criteria. Lands may be volunteered at two times during the site selection process. The first period for accepting volunteered lands will be within 45 days of the formal adoption of the Task Group's site selection process. The second period will be during the first 90 days of Chem-Nuclear's site selection activities.

IMPROVEMENTS TO SITE SELECTION IN ILLINOIS

Unlike the previous "greenfield" effort to select a site in Illinois, the new site selection process which the Illinois legislature has promulgated and the Governor has signed into law is clear, unambiguous, and explicit in its intent to lay out a process which capitalizes on the issues raised and lessons learned in the previous effort. The new improvements in the Illinois Process explicitly altered inherent conflicts in the Act and will lead to a safe, suitable and environmentally sound site for Illinois and Kentucky generators to use for the next 50 years.

In the new process, the contractor, Chem-Nuclear, plays a significantly enlarged role, the Illinois Scientific Surveys play a new and expanded role, the Department of Nuclear Safety's role is aligned strictly with its regulatory mission, the Task Group is formed, serves, and is abolished and the general public has an opportunity to participate in the process from criteria development to site operations. Figure 1 shows a brief view of the major process milestones divided into six project life

phases. As of February 1996, the project is in phase I and is beginning the process of performing statewide screening to identify 10 or more locations in Illinois which appear suitable to host the LLRW Facility.
Fig. 1
IMPROVEMENTS IN THE ACT
The Illinois Management Act , amended by House Bill 1918 and signed in March of 1993, was streamlined and strengthened to assure that it did not fail to guide the Illinois participants to a successful conclusion in this, the second attempt to develop a LLRW disposal facility in Illinois. Assurance for success, especially in LLRWDF Development processes, can be and is defined as "incorporation of change based on past failures to improve probabilities of success". In Illinois, this is exactly what has transpired and what is being implemented.
The new process concedes the old process to history and starts afresh with several new concepts: The initial process in Illinois put the Illinois Department of Nuclear Safety in a position where it was both site developer and site regulator. In this obviously conflicting role, the Department had to promote, seek, find, market, and regulate the process. In the new process, the Department is a participant until the criteria are adopted. At that point, the Department plays no active role in site selection and is not a process player until Chem-Nuclear submits a license application to them.
The previous Management Act established general criteria that provided minimal technical direction to use to establish a site. Unfortunately, the criteria that were developed under the Act fell short of the technical specificity needed to guide the process participants to find a suitable site. Hence, IDNS, the regulator, had to step up and assume responsibility to develop the site selection framework including specific technical criteria to use to qualify sites. Revisions to the Management Act created the Task Group which has the responsibility to develop the needed technically specific criteria. The process employed by the Task Group as specified in the Management Act to develop these criteria, specifically lays out technical discipline experts to sit on the Task Group, specifies technical areas which must be addressed by criteria and mandates public participation into the entire process.
The Act, as a result of detaching the IDNS from the process, separated the site selection decision from the more rigorous health and safety decisions made during the licensing process. This in fact puts the responsibility to make decisions directly where it should be. The Task Group makes criteria decisions, the Surveys make broad statewide screening decisions, Chem-Nuclear must make precise technical comparisons of sites to selection criteria to find licensable sites, and the Department makes licensing decisions.
The Act dealt the responsibility for statewide screening and broad application of the Task Group's criteria to the State Scientific Surveys. This action eliminated the Department's role in the previously controversial technical part of the process and as well put the State's trusted technical body in charge of one of the most crucial parts of the process. The Surveys will eliminate broad areas in the state from further consideration based on their many years of experience in statewide studies and investigations, and will provide to Chem-Nuclear those areas that appear likely to meet the Task Group criteria.
Another significant change in the Act, is to delegate to the contractor, Chem-Nuclear, the direct responsibility to select a site which it will

have to characterize, defend during licensing and eventually operate. This puts the responsibility to perform directly on the company who stands to capitalize the most from the success of the siting process. The original process had the drawback of putting the IDNS in the role of needing to market the siting. The Department had to try to attract communities to volunteer since a community selected without consent could easily pass a resolution stating that the community did not want the site and they were no longer in consideration. This "promotional" role for the Department was one of the most significant changes in the Act.

A final change that the Act brought about is the removal of politics from the process and the inclusion of the public into all phases of site development, from Task Group proceedings and Scientific Surveys screening to Chem-Nuclear selecting a single 640 acre parcel of land in Illinois to license. This was accomplished by removing local referendum which restricts the ability of a city management to override the wishes of the community, and opened the process to all members of the public, anywhere, anytime.

THE IMPROVED SITE SELECTION PROCESS

The project participants as shown on Fig. 2 each have a defined role as outlined in Section 10 of the Management Act. The new and improved process includes formation of the Task Group, the Task Group developing criteria with which to screen the state, and Chem-Nuclear developing criteria with which to evaluate safety/licensing, environmental, and operational qualifications of the locations determined by the Surveys.

Fig. 2

The Task Group Criteria are intended to serve as a reasonable and objective basis to eliminate certain lands of the state from consideration, to select other lands for siting a disposal facility, and to determine if the three sites offered by Chem-Nuclear are promising for the development of a disposal facility. However, the Task Group Criteria do not address all necessary requirements that the site and disposal facility will have to meet.

Chem-Nuclear will develop additional safety/licensing, environmental and operational criteria to use in selecting three sites from the Scientific Surveys multiple locations. These criteria include Mandatory Criteria , which define characteristics a site must (or must not) have to be licensed; Avoidance Criteria which define less desirable conditions, and Preference Criteria which define desirable conditions. Chem-Nuclear Avoidance and Preference Criteria are comparative in nature and can only be applied to land which meets both the Task Group Criteria and the Chem-Nuclear Criteria.

USE OF PERFORMANCE ASSESSMENT FOR SITE SELECTION AND CHARACTERIZATION

The general practice, in the past, for site selection and characterization of nuclear waste disposal facilities, was to gather as much information on the site and region surrounding the site as possible, analyze the information, and then make a choice of a site. The initial appearance of performance assessment (PA) was usually in a safety report or the license application following the detailed characterization of the site.

This approach to siting underwent a change, starting in the mid-1980's, as part of the site characterization program for a high level nuclear waste site. There were three factors that drove this change: first, the testing program had grown so large that it was too costly and time consuming to perform all the tests that had been identified by the

researchers; second, there was a need to identify those tests that were important to determining whether or not a site was suitable; and third, the period of performance was so long that using a deterministic approach was not suitable in a licensing environment. The combination of these three factors resulted in PA assuming an important role in the siting process. For the Illinois siting process, PA plays a role from the start of siting to both guide the siting process and support the licensing proceedings to help ensure that the testing and data needs are well defined and consistent with what is needed to ultimately site a disposal facility. The Illinois siting program must acquire all the data needed to make a siting decision but without engaging in a testing program that is far in excess of the requirements. PA will greatly assist the project in obtaining only the appropriate data.

PERFORMANCE ASSESSMENT IN THE SITING PROCESS

As noted in the "Improvements to Site Selection" section, The Illinois Scientific Surveys engage in Statewide screening to find 10 or more locations that appear likely to meet the Task Group Criteria. Once this operation is complete, Chem-Nuclear then proceeds to reduce these 10 or more locations to three (3) sites that appear promising for development of a disposal facility. It is during this stage where the locations are being reduced to 3 sites and one of the sites is selected for detailed site characterization, that PA is applied.

APPLICATION OF PERFORMANCE ASSESSMENT

In simple terms, PA is a computational method that models various parameters which are important to determining the suitability of a site and ties the parameters together in a relational manner to predict overall system performance. The important parameters are:

- Disruptive events (human intrusion, earthquakes, etc.) that can affect the disposal site performance;
- Performance of the engineered barrier system;
- Radionuclide transport pathways through the geosphere; and
- Radiation dose at the disposal site boundary.

Initially, following the identification of the 10 or more locations by the Illinois Surveys, the locations will be evaluated using the Task Group and Chem-Nuclear Criteria. However, there are criteria that cannot be evaluated without site-specific field data, especially, criteria related to subsurface geology and hydrologic conditions. It is the availability of these additional site-specific data that will serve as the initial input into the preliminary PA model. The model, however, at this point in the process has only limited field data which can be used for evaluation of the locations. Nevertheless, PA will be used to gain information on the total system performance, and aided by Value-of-Information analyses, will guide the subsequent collection of field data.

VALUE-OF-INFORMATION ANALYSES

Value-of-Information Analyses (VOI) will be used to assist in the design of data collection activities at various stages of the siting program. A VOI analysis explicitly compares proposed testing decisions to ensure that the testing selected offers the value in return of useful data. This technique is based on an iterative PA process where as more data is collected it is added to the data base of the model and the program is run with the enhanced data base. By examining the output and performing sensitivity analyses, it will be possible to identify which activities (data collection) will be most critical to rapid and accurate determination of site and design suitability and provide the greatest

reduction in overall uncertainty, particularly in the overall uncertainty of the preliminary PA results.

DECISION ANALYSIS AND PEER REVIEW

The use of PA and VOI, as noted above, will provide a framework for constructing and implementing the siting program. In addition, PA will provide a basis for selection of sites to present to the Task Group, and ultimately aid in the selection of one site for development as a disposal facility. The PA will provide the decision maker with the probabilities of a particular set of sites or a site having a "likelihood of success". However, PA alone will not necessarily provide a clear path to selection of site(s) since there will be uncertainties associated with the collection of data and the probabilities that a site will perform as predicted for many hundreds of years.

Because of the uncertainties in performance, the Program will use a decision-analysis approach, possibly using techniques such as the multi-attribute utility method where the data on the site(s) is subjected to an elicitation routine to provide additional information on the expected site(s) performance and the "likelihood of success". In addition to a formal decision-analysis approach, the Project will make use of peer review to provide the decision maker with independent evaluation of specific technical issues. The ultimate decision maker on selection of three (3) sites, and subsequently, a single site for licensing, is the Project manager.

Making the Site Selection Decision

The process which Chem-Nuclear will implement to make the site selection decision can be described in general terms as follows: Using an iterative procedure, the 10 or more locations provided by the Surveys will be examined using both non-intrusive and intrusive testing to enlarge and enhance the data base, and with the assistance of PA and decision-analysis, reduce the locations to three (3) sites and ultimately to one site. A typical process is shown in Fig. 3. This figure shows the iterative process as the large land areas, associated with Locations, are successively reduced to smaller areas, then to three (3) sites. A similar process is used as the three (3) sites are reduced to a single site which is then subjected to site characterization and licensing. As the process unfolds, an evaluation is made of the probability that each area/site will comply with the criteria and the licensing requirements. Decision analysis and PA will be used to guide the testing, as well as to identify areas and sites which should be dropped from the list of candidates or retained. This iterative process continues throughout the siting stage of the program.

Fig. 3

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A DETAILED COMPARATIVE EVALUATION OF SIX LOW-LEVEL WASTE DISPOSAL METHODS

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ABSTRACT
The New York State Low-Level Radioactive Waste Management Act, as amended, requires that the New York State Low-Level Radioactive Waste Siting Commission (Siting Commission) select a disposal method or methods before selecting a disposal site. The Siting Commission chose to evaluate six candidate disposal methods -- above-grade vaults without an earthen cover, covered above-grade vaults, below-grade vaults, augered holes, a vertical shaft mine, and a drift mine -- in great detail before making a choice. The paper describes the evaluations and comparisons of the six methods, with emphasis on the performance measures of most interest in making such comparisons. The Siting Commission conducted the most detailed comparison of low-level radioactive waste disposal methods to date. In July 1995, it selected covered above-grade vaults, with both a drift mine or a vertical shaft mine as alternatives.
INTRODUCTION
In response to the federal laws on low-level radioactive waste disposal, New York enacted the Low-Level Radioactive Waste Management Act in July 1986 and amended it in 1990. The Act created a Low-Level Radioactive Waste Siting Commission with responsibility for selecting both a preferred method and a site for disposal of New York's low-level radioactive wastes. Under the amended Act, the Siting Commission was required to evaluate alternative disposal methods and select a preferred disposal method (or methods) before proceeding to select a site. The process for method evaluation and other steps in the method selection process are described in some detail in Ref. 1-3. It is summarized in the following paragraphs.
Based on the input received during a public outreach program and the review of previous work by the Commission and others, six candidate disposal methods were evaluated in the disposal method selection process:
- Above-grade vaults without an earthen cover (uncovered above-grade vaults).
- Covered above-grade vaults.
- Below-grade vaults.
- A vertical shaft mine.
- A drift mine.
- Augered holes.

This group of candidate disposal methods covers the full range of major conditions affecting permanent low-level radioactive waste disposal. It also includes the two alternative methods specifically mandated for consideration in the 1990 amendments to the 1986 Act: deep vertical shaft mined disposal and above-ground monitored retrievable disposal (represented by above-grade vaults without an earthen cover).
The Commission evaluated the six candidate disposal methods with respect to the seven evaluation considerations listed in Table I. The relative abilities of each of the disposal methods to satisfy criteria that were formed from the evaluation considerations were assessed. Factors for use in developing criteria and in evaluating the disposal methods are also shown in Table I. This paper summarizes the results of the evaluation and indicates the methods preferred by the Commission.
Table I
In the spring of 1995 the New York legislature reduced the Commission's funding to a point where it was no longer able to operate. Prior to

ceasing operations, the Commission chose the tentative preferred disposal methods (the wording comes from the 1990 Amendments) described here. Presently there are no plans to continue to look for a disposal site in New York.

THE DISPOSAL METHODS EVALUATED

To perform comparisons of alternative disposal methods, the Siting Commission prepared conceptual designs for the six candidate methods. Those conceptual designs were intended to meet design requirements specified in New York regulations and to be of sufficient detail to enable objective comparisons of the six methods with the aim of selecting a preferred method (or methods). In fact, it was often necessary to add more detail to the designs than is typical for a conceptual design, in order to enhance public understanding of how a disposal facility would function. The conceptual designs are described in Ref. 4-5.

Some features are common to all six disposal facility designs. These include features such as buildings for the receipt of waste, the placement of waste in concrete overpacks called modular disposal units (MDUs), and conduct of administrative functions. The designs were based on the receipt of a total of 5.5 million cubic feet of low-level radioactive waste over 60 years. This waste volume corresponds to the most likely or "expected" case described in Ref. 6.

Conceptual designs were developed for generic disposal facility sites with characteristics based on Ref. 7, which provides information about the geology and hydrology of sites that are typical of the non-excluded areas in the State of New York. For the four near-surface methods (i.e., not mines), the generic site characteristics do not have a major effect on the designs, except that generally shallow groundwater conditions limit the depths to which waste can be placed below grade. Designs were prepared for the mine disposal facilities in four geologic media: shale, salt, limestone, and igneous/metamorphic rock. The designs for covered above-grade vaults, below-grade vaults, and augered holes have earthen covers placed over the concrete vaults or holes. Those earthen covers are 7 feet thick and consist of various layers of natural and man-made materials that serve functions such as prevention of water infiltration, prevention of animal and plant intrusion, and support of short-rooted plants at the surface. The conceptual designs were developed to allow comparison of the inherent features of the six candidate disposal methods. The designs were not necessarily optimized and, since potential sites were not known, they will probably be changed considerably if they are developed further. Table II provides information about some characteristic parameters of the conceptual designs. The parameters shown are some of those that were used in the detailed comparisons.

Table II

EVALUATIONS

Comparative evaluations were conducted of the ability of the six candidate disposal methods to satisfy the considerations and factors given in Table I. Both qualitative and quantitative evaluations of the methods were used, depending on the consideration or factor. Some factors were divided into several parts called measures, the measures were evaluated separately, and then the evaluations of the measures for each factor were combined. All evaluations were presented in the visual form illustrated in Fig. 1, showing the relative ranking for each of the six candidate disposal methods. Each ranking also contained a brief statement of the level of uncertainty associated with the evaluations. The ranking

in Fig. 1 is for Factor 5B, Confidence in Licensability. The evaluation for that factor was a qualitative one and above-grade vaults without covers were rated significantly lower than the other methods because they would probably not meet the present New York regulations for low-level radioactive waste disposal. The Siting Commission was unanimous, however, in its view that the most important factor in selecting a disposal method was the protection of the public health and safety; in this regard all but the Uncovered Above-Grade Vaults were rated favorably.

Fig. 1

Figure 2 shows the visual presentation of the relative ranking of the disposal methods for a measure used for Factor 1A, Protecting the Health and Safety of the Public. The evaluation for this measure, potential peak radiological dose to the public after closure, was quantitative. It was the result of detailed dose assessments for the 10,000 years after disposal facility closure, based on the conceptual designs and the generic site descriptions (8). Figure 2 shows that the uncovered above-grade vaults and mines in igneous or metamorphic rocks were estimated to give doses in excess of the limits prescribed by New York regulations. The poor projected performance of the mines in igneous and metamorphic rocks was based on geologic data available for New York. Dose projections for mines in limestone were very dependent on future drilling activities. Since limestone formations in New York are expected to always be overlain by shale or salt (7), no further analysis was made of this measure for limestone mines. Mines in salt and shale were projected to give very low doses.

Fig. 2

The abilities of the six candidate disposal methods to meet all 19 Factors listed in Table I were evaluated. However, not all of the factors or measures used to evaluate factors were found to be helpful in discriminating among the methods. In some cases, the measures were very similar for all six methods. In other cases, the impacts of the methods (for example, radiation doses to the public during normal operations, one of the measures for Factor 1A), while different, were so small that there was no significance to the differences. All of the measures and factors were divided into three categories after the evaluations were completed: those judged to be of little or no value in discriminating among methods, partial discriminators, and major discriminating factors. Table III shows the way the evaluations for the latter category were summarized. A similar table that was prepared for the partial discriminators is shown in Table IV. The entire set of evaluations is summarized in Ref. 9.

Table III

Table IV

SELECTION OF THE TENTATIVE PREFERRED DISPOSAL METHOD

The comparative evaluations were presented at three public meetings in March and April of 1995. The Commission decided not to assign numerical weights to the considerations and factors, or to give numerical ratings representing the abilities of the disposal methods to satisfy the considerations and factors. It was felt that numerical weights and ratings can portray a degree of objectivity that is not always supported by the available data and the level of design detail. Also, use of numerical weights can result in a lack of flexibility that makes it difficult to respond to public concerns. In its selection of the preferred methods, the Commission used a qualitative approach that represented the individual and collective judgements of its members. The

Commission members each weighed the advantages and disadvantages of the methods separately; any two Commissioners could have chosen the same method for somewhat different reasons. Although other methods (such as multi-attribute utility analysis) could have been used to compare the candidate disposal methods, it is believed that the final choice of preferred methods would be essentially the same.

The Siting Commission chose Covered Above-Grade Vaults as the Tentative Preferred Disposal Method. For the major discriminating factors (Table III), the Commission determined that Covered Above-Grade Vaults was rated more favorable for all factors. Covered Above-Grade Vaults would protect the public from radiological impacts (Factor 1A). There is confidence that the construction materials used would maintain their essential physical and chemical properties as long as necessary (Factor 3A). Monitoring to assure compliance with regulatory standards could be conducted without difficulty (Factor 4A). This method would allow for the retrieval of waste, either the entire volume of waste or only Class B and C waste (Factor 4B). The method offers flexibility in siting options; it could be used in a variety of locales and conditions (Factor 5A). There is confidence that the method could meet regulatory requirements and be licensed (Factor 5B). Finally, the projected total life cycle costs for this method compared favorably with other methods (Factor 6A).

The Covered Above-Grade Vaults were also rated more favorable for fifteen of the seventeen partial discriminating factors (Table IV). The only two factors where this method rated less favorable were relative resistance to inadvertent intrusion after facility closure (Factor 1C) and increased truck traffic in the host community because of the importation of materials for construction of the cover system (Factor 7A). Neither of these two factors was considered decisive.

Recognizing the need for flexibility in fitting a method to a site to be determined later, the Commission chose Mine Disposal (i.e., either a Drift Mine or a Vertical Shaft Mine) as an alternative Tentative Preferred Disposal Method. The two mine methods shared many similarities in terms of their design, function, and favorability ratings and can be considered variations of one method. If placed in certain rock media, such as shale or salt, both mine methods rated more favorable than three of the near surface methods for most of the comparative factors in Tables III and IV. A mine in shale or salt performed very effectively to isolate radioactive waste and protect public health and safety and the environment (Factors 1A, 2A, and 3B). A mine in igneous or metamorphic rock of the type typically found in New York was considered to be unsuitable, however, because of the potential for high postclosure peak radiation dose to the public.

The mine methods ranked less favorable than the Covered Above-Grade Vaults for a number of factors that the Commission judged to be important. As can be seen from Table III, the mine methods were rated less favorable than covered above-grade vaults with regard to the ability to monitor compliance (Factor 4A) and, for the vertical shaft mine, the cost of recovering or retrieving all Class B and C waste (Factor 4B). Factor 4B, Ability to Recover or Retrieve Waste, was evaluated separately for recovery or retrieval of all waste and for Class B and C waste only. They were also rated less favorable than covered above-grade vaults with regard to protection of health and safety of workers (Factor 1B), and operational cost sensitivity (Factor 6B). On the other hand, mines were rated more favorable than covered above-grade vaults in terms of their

resistance to inadvertent intruders (Factor 1C) and the amount of truck traffic near the disposal site (Factor 7A). The principal deficiencies in the other disposal methods are described in the following paragraphs.
Uncovered above-grade vaults were considered less suitable because the estimated potential doses to the public after closure indicated that this method may not be able to meet the regulatory performance objectives (Factor 1A); the method is not licensable under current New York regulations (Factor 5B); the vault concrete and reinforcing steel may not last long enough to provide adequate containment of radioactive contaminants (Factors 3A and 3B). This method is not as capable of protecting air, water, and biota (Factor 2A) and it is more vulnerable to earthquakes and other extreme events (Factor 3C).
Augered holes were judged to be less suitable for low-level radioactive waste disposal because they require a much larger land area, making siting much more difficult (Factor 5A); have higher costs as reflected in the present value of the facility (Factor 6A); have higher costs to retrieve or recover wastes in the postclosure period (Factor 4B). They are also less resistant to inadvertent intruders (Factor 1C) and, due to the large site size, have larger nonradiological impacts on land, water, and any nearby community (Factors 2B, 2C, and 7A).
Below-grade vaults were considered less suitable than the methods chosen because of the potential difficulty of finding a site in New York with a water table deep enough to accommodate the conceptual design for this method (Factor 5A); a depth of at least 30 feet is needed. Below-grade vaults were also found to offer less resistance to inadvertent intruders (Factor 1C). The possibility of covered vaults that are partially above and partially below grade was discussed but this approach was not evaluated. Had the method selection process continued beyond selection of a tentative preferred method, an evaluation of this hybrid approach may have been pursued.

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Session 53 -- PACKAGING & TRANSPORTATION II
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53-1
FUNDING OPTIONS FOR TRANSPORTATION PLANNING ASSISTANCE TO STATES AND TRIBES

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ABSTRACT
The projected volume of radioactive and mixed-waste material shipped by the U.S. Department of Energy is expected to grow significantly in the future as DOE waste management programs are fully implemented. Concerns raised by affected jurisdictions related to DOE transportation activities are carrier qualifications, vehicle inspection, enforcement of applicable regulations, transportation operational procedures, and support for emergency preparedness and response activities. States have requested that DOE provide funds for all transportation-related activities, not just selected shipping campaigns. There are a number of existing mechanisms to fund these activities in affected states, however, most of these take a 'piecemeal' approach. This paper examines current DOE and other federal agency funding for transportation and emergency management planning and explores options for streamlining and rationalizing DOE's approach toward funding states and Tribes.

INTRODUCTION

In recent years there has been an increasing focus on cooperative transportation planning, technical assistance and training between DOE and states, Tribes and local governments affected by DOE shipments of radioactive material. The department recognizes the benefits of coordinated planning and anticipates greater demand for technical assistance and training from host and corridor jurisdictions as DOE's radioactive shipments increase through the next decade (see Fig. 1)a. By one estimate, up to 40 states will be directly affected as either hosts or along the transportation routes. Shipments of material managed by the Office of Environmental Management (EM) constitute a large majority of the projected increase.

The Office of Civilian Radioactive Waste Management (RW), the Waste Isolation Pilot Plant (WIPP) and other EM programs are successfully engaging states, Tribes, local governments, regional organizations, and professional associations in all levels of transportation planning including transportation program development and specific shipping campaigns. The RW and WIPP programs are required, by statute, to provide transportation-related technical assistance and training to state, tribal and local jurisdictions affected by their shipments. Both programs have several cooperative agreements in place with states, Tribes, and regional and national organizations to provide transportation-related planning assistance and training. Other EM programs have used these agreements to channel funds to state and tribal jurisdictions for planning and training

activities related to specific shipment campaigns. States, Tribes and local governments also receive funds through other non-transportation-related DOE and federal agency programs (Federal Emergency Management Agency (FEMA) and the Department of Transportation (DOT)) which indirectly benefit their transportation-related planning and training. States also provide a major portion of emergency preparedness funding through fees or state appropriations.

Fig. 1

Providing assistance to states, Tribes, local governments and other interested parties to support their involvement in general transportation program development and specific shipment planning has paid dividends. The Transportation External Coordination Working Group (TECWG) has helped to build a consensus on what issues DOE must address in developing its transportation programs and is now recognized as a model of public involvement for other non-transportation EM programs. Also, coordinating with affected jurisdictions played an important role in successfully planning and implementing the Cesium-137 Capsule Return Program and the Foreign Research Spent Nuclear Fuel Urgent Relief Program.

Still, recent budget cuts and subsequent programmatic reprioritization have focused attention on the effectiveness and efficiency of the department's transportation-related planning assistance and training efforts. Clearly, some of the arrangements for engaging states, Tribes, local governments and other interested parties in transportation planning are redundant. For instance, several of the RW and WIPP cooperative agreement groups have the same or similar membership and frequent single issue meetings place heavy administrative burdens on all participants. It is evident that DOE is no longer able to fund all transportation planning efforts that have been developed over recent years.

To continue capitalizing on the benefits of cooperative transportation planning, meet the likely demand for greater technical assistance and training generated by anticipated increases in DOE shipments, and reduce administrative and program inefficiency, DOE must develop a more rational and consistent approach to funding transportation planning assistance and training. This paper examines the current method for delivering funds and proposes a more streamlined approach.

CURRENT APPROACH

Funding for planning assistance and training provided by each DOE program to states, Tribes and local governments is based on their program mission. DOE recognizes states have base programs and DOE's responsibility is to assist states in planning and preparing for the transportation of DOE materials. For instance, the WIPP and RW cooperative agreements provide recipient organizations and their members with the funds and training they need to help DOE plan and implement the shipping programs for transuranic waste and commercial spent nuclear fuel, respectively.

Fig. 2

Although other EM offices use the WIPP and RW agreements, funds are also provided only for the material managed by the particular program office. Figure 2 shows the sources and delivery mechanism of funding assistance for planning and training provided by EM programs to states and Tribes during FY1994. A comprehensive picture of transportation-related financial assistance to states and Tribes from all federal sources can be found in Bradbury and Jones (1994).

The current DOE approach is largely a reflection of the legislative, administrative, and organizational differences among traditionally separate and functionally distinct DOE transportation programs. As part of her Strategic Alignment Initiative, Secretary O'Leary has indicated that minimizing DOE's "stovepiping" of programs is a key component of her efforts to increase the department's administrative efficiency. Although transportation programs have long been seen as separate and distinct from various DOE program offices, missions, within the department, their transportation missions are cross-cutting. They involve or affect many of the same people outside DOE (States, Tribes, localities). Moreover, the operational and public safety responsibilities of state, tribal or local personnel are not likely to differ significantly for the bulk of DOE shipments. In addition to the obvious administrative and resource inefficiency of sometimes redundant DOE transportation assistance and training programs, the current fragmented approach also inhibits recipient jurisdictions from pooling DOE assistance with other federal assistance or resources to efficiently plan for and meet the entire spectrum of their public safety needs.

POSSIBLE OPTIONS

There are three basic options for funding coordinated transportation planning, technical assistance and training: national; regional; and "local". Each option has advantages and disadvantages for achieving various aspects of a coordinated approach to transportation planning and implementation.

A National Option

Through a national approach, DOE would continue providing funds for transportation planning assistance to organizations with a national perspective on issues related to the transportation of radioactive and other hazardous materials. Since its inception in 1992, the TECWG has worked with RW and EM to identify and resolve issues in 6 broad transportation-related areas: 1) General Program Planning: 2) Transportation Operations; 3) Emergency Management; 4) Technical Assistance; 5) Training; and 6) Public Information & Education. The group meets twice annually and maintains a rigorous process for incorporating participant input into transportation program development. Individual TECWG members serve as liaisons between the group and their broader organizations. Many TECWG member organizations have significant influence on the course of various issues of concern to DOE transportation programs.

In addition to the TECWG, EM sponsors the Local Government Network (LGN) through a cooperative agreement with the Urban Energy & Transportation Corporation. The network is a loose affiliation of senior local government officials, from around the country, emergency management, fire and rescue, law enforcement, and environmental health officials. Participants in LGN regional and national meetings provide DOE transportation managers with a local perspective on hazardous material transportation.

Like TECWG, LGN participants help shape DOE transportation program development by applying their knowledge and experience towards transportation-related issues such as technical assistance and developing and shaping training. Both groups have been instrumental in developing many of the ideas and issues raised in this paper. They will continue to work with DOE to develop and implement more efficient and effective transportation programs. As groups, neither the TECWG nor the LGN are

directly involved in planning specific DOE transportation actions, although individual members can and have served as valuable contacts for DOE programs and field offices during shipping campaign planning.

A Regional Option

As part of a regional funding option, DOE would continue sponsoring various regional or specific national organizations through cooperative agreements to convene their respective members to address operational and planning issues related to specific DOE shipments. RW currently has agreements in place with the Southern States Energy Board (SSEB), Western Interstate Energy Board (WIEB), The Midwest Office and Easter Regional Conference of the Council of State Governments (MOCSG and ERC) , the National Conference of State Legislatures (NCSL), the National Congress of American Indians (NCAI), the Conference of Radiation Control Program Directors (CRCPD), and the Commercial Vehicle Safety Alliance (CVSA). WIPP has agreements with SSEB, the Western Governors' Association (WGA), the Confederated Tribes of the Umatilla Indian Reservation, the Shosone-Bannock Tribe of the Fort Hall Indian Reservation, and the State of New Mexico.

These regional government organizations provide a neutral forum for discussing issues of regional concern, developing regional response or solutions such as a mutual aid agreements, and moderating between DOE programs and individual states. In addition to funds for administering the cooperative agreement and convening meetings, DOE could provide "pass-through" funds to the states through regional groups for, among other things: 1) DOE trainer "consultations" with affected state jurisdictions; 2) training curriculum development and distribution; and 3) train-the-trainer courses. States would then be responsible for providing training to appropriate local personnel.

There are several drawbacks to working through regional groups. Experience with the Cesium-137 Capsule Return Program and the Foreign Research SNF Urgent Relief Program has shown that obtaining needed authorizations and adding incremental funding to existing agreements can be awkward and time-consuming. Tribal interests are not represented in the current cooperative agreements with these regional groups. In addition, existing cooperative agreements do not represent all the jurisdictions likely to be affected by future DOE shipments. Additional State-by-State or Tribe-by-Tribe agreements would be required. Implementing additional agreements to cover the remaining affected jurisdictions would result in additional administrative costs for both the federal government and the States and/or Tribes. The "pass-through" nature of the regional approach to program implementation may not result in the most effective or consistent application of resources by recipient organizations because potential differences in the interests of the regional groups and the constituents they represent.

A "Local" Option

The "local" or jurisdiction-by-jurisdiction option is the traditional federal approach for directly aiding states and Tribes and is most effective when dealing with a limited number of affected jurisdictions. DOE-EM has experience in direct funding through several cooperative agreements with states and Tribes, and through its Agreements in Principle (AIP) with several states hosting DOE facilities. Other federal programs (FEMA, DOT) providing funds directly to states and Tribes may provide an avenue for DOE to do the same. However, some programmatic control and objectives might be lost.

EM has several cooperative agreements in place with Tribes to participate in environmental monitoring activities, to conduct facility ground water testing, and to participate in a dose reconstruction project. One such agreement with the Confederated Tribes of the Umatilla was used to fund transportation-related emergency response activities as a pilot project.
EM's AIP process provides more than $20 million per year to 13 states for technical and financial support for independent oversight of facility environmental monitoring and emergency response planning. AIPs are funded by EM and administered by DOE field offices located in or near recipient states. Current guidance to participating states emphasizes that AIP funds are for emergency response must be directly related to off-site consequences of on-site emergencies. AIP guidance and procurement procedures would need to be amended to include transportation-related activities. However, it is not clear in every case that the state agency receiving AIP funds is the appropriate recipient of transportation-related funds.
The Federal Emergency Management Agency (FEMA), the Department of Transportation (DOT), and the Environmental Protection Agency (EPA) also are significant sources of planning assistance and training for state, tribal and local governments.
FEMA is responsible for coordinating and distributing federal assistance for emergency planning, preparedness, response, and mitigation for civil defense and peaceful radiological incidents (for both fixed and transport-related), and natural disasters. FEMA has used its Comprehensive Cooperative Agreement (CCA) mechanism to consolidate and channel financial assistance from various federal sources through the governor's office to the emergency management agency in each state. Tribes can participate in CCA programs either through separate cooperative agreements or (at their option) by receiving funds passed through state agreements.b
Currently, 13 programs are funded through the CCA including FEMA's Emergency Management Assistance Program, Emergency Management Training Program and Hazardous Materials Program, and the U.S. Army's Chemical Stockpile Emergency Preparedness Program (CSEPP). Each sponsoring agency governs the content and scope of the specific program while FEMA administers the overall agreements with participating states and provides technical assistance. Thus, each program has different eligibility and reporting requirements to meet specific program and information needs, and each comprehensive CCA is tailored to meet the needs of the state.
The DOT also provides assistance through training and planning grants authorized by the Hazardous Materials Transportation Uniform Safety Act (HMTUSA) of 1990 and through its Motor Carrier Safety Assistance Program (MCSAP). Both states and Tribes are eligible for the HMTUSA grants which are intended to assist in training public sector employees in responding to hazardous material incidents and planning grants for developing, improving, and implementing emergency plans required under Title III of the Superfund Amendments and Reautherization Act (SARA) of 1986. Grant applicants are required to identify fees assessed on hazardous material shipments and the purpose of those fees. The MCSAP program provides funds to assist states in covering costs for roadside inspections, safety compliance reviews, and follow-up enforcement actions.
Transportation-related assistance from the EPA is provided through a competitive grant process designed to improve state, tribal and local emergency planning and right-to-know programs established under SARA

Title III. The grants are focused on geographic areas that have high risks for a chemical accident including transportation intersections and corridors.

By channeling its assistance through an existing federal program, DOE may be able to reduce its overall administrative costs by utilizing established procedures and personnel (both federal and state) experienced in administering assistance programs. Also, this approach would enable jurisdictions to better leverage the assistance provided by DOE against the assistance they receive from other sources and their own resources for routine transportation and emergency preparedness activities. DOE would, however, risk losing some administrative and programmatic focus and program recognition among recipients

Of the existing federal assistance programs discussed here, only FEMA's CCA mechanism has been used by other federal agencies to implement their respective assistance programs. The CSEPP program is an example. A March 1995 General Accounting Office (GAO) report on the CSEPP program found inadequate financial data and management controls by both the Army and FEMA despite an attempt in 1994 to restructure the program's management procedures.c

The GAO report also questions FEMA's plans to replace its current system with Performance Partnership Agreement (PPA) beginning in FY96. According to FEMA the new mechanism will consolidate funding streams into a multi-year performance agreement which will enable states greater flexibility to allocate resources to address their particular hazards. It will also "reduce substantially micro-management and current reporting requirements." In its analysis, the GAO cautions that "CSEPP funds may lose their identity if funds for many programs are awarded to states by functional categories."

Given the GAO's concerns about of the CSEPP program and the fact that FEMA is planning to replace its current system with a new block-grant-type assistance program, channeling DOE's transportation-related assistance through FEMA, at this time, does not appear to be prudent.

A Hybrid Option

To this point this paper has examined the current funding system for transportation-related planning assistance and training and several other options. Each of these options, by themselves, have disadvantages that might jeopardize the effective completion of DOE program office missions. Therefore, we propose a hybrid option. The hybrid option incorporates pieces from all of the other previously mentioned options.

Recognizing that transportation-related planning assistance and training are cross-cutting needs; every DOE program will now, or in the future, require these services and expertise. DOE should maintain these services in a "corporate" level, rather than have each program office develop, implement, and support a transportation assistance and training program. There are several reasons why this makes sense especially in light of the current budget situation. First, external parties have requested that DOE consolidate its training support and provide a consistent approach that can be incorporated into existing state, tribal, and local programs. Second, DOE can no longer afford to maintain separate, but similar, training and assistance for each DOE program office. Finally, from a program management point of view, a "corporate" office is better able to integrate ongoing transportation and training activities throughout the complex and between the field offices and DOE HQ.

After consultation between DOE-HQ program offices, the DOE "corporate" transportation office, and with input from the site programs and DOE support offices, would pool assistance funding and or training for field programs that would be managed at DOE field offices. DOE-HQ maintains its policy and coordination role. As the field offices and the site programs are the people implementing shipping campaigns and interacting with the first responders, state, tribal, and local officials the operational decisions should be made closer to "home." DOE-HQ would continue to work with the TECWG and LGN and the Transportation Internal Working Group (TICWG) to address broader program and policy issues. (The TICWG is the internal DOE working group that brings together program people to discuss transportation issues.) In the near term DOE-HQ would continue to work with the regional groups of states, with an eye towards having them work more closely with the DOE sites (site programs) in the states they represent to address operational planning. See Fig. 3.

There would be an eventual devolution of technical assistance and training to be handled by either the States and/or the DOE sites. In some cases the regional groups can serve as a repository for information and/or trainers. Technical assistance might flow through the DOE operations offices to the States. We refer to the successful Idaho example, in which DOE-HQ provides the materials and the funding to DOE-Idaho and they in turn provide their training, exercises, outreach, and education. At some point we might consider funding states directly, given the legislative precedent seen in the 180 (c) provision of the National Waste Policy Act.

The primary advantage of this hybrid option is that it provides for direct program office involvement in field implementation while providing the field office system with authority to plan and implement shipments. It strongly encourages "corporate" integrated transportation planning, not simply on a program-by-program basis. It also enhances DOE's ability to fulfill program objectives by crafting agreements which focus on the specific technical assistance and training needs in transportation corridors.

Transitioning from the current fragmented approach to the proposed approach will take time. Successfully implementing the proposed option rests largely on the willingness of HQ programs to consent to relatively decentralized program implementation and the ability of all parties to agree upon an appropriate formula for determining DOE program funding contributions. These and other internal DOE issues are being address by the Transportation Internal Coordination Working Group (TICWG).

Fig. 3

CONCLUSION

This paper has presented a variety of funding options for providing transportation-related assistance and training to States, Tribes, and localities. Each of these options has advantages and disadvantages. One is able to argue the merits of pieces of all of these options. However, the expected increase in the number of radioactive material shipments during the next decade and the current budget situation are providing strong incentives for DOE to streamline its fragmented approach to funding transportation planning assistance and training. If done correctly, a more effective and efficient funding and planning process, one that increases the likelihood of a safe, successful and timely movements, can result.

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THE TRUPACT-II MATRIX DEPLETION PROGRAM

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ABSTRACT

Contact-handled transuranic (CH-TRU) wastes will be shipped and disposed at the Waste Isolation Pilot Plant (WIPP) repository in the Transuranic Package Transporter-II (TRUPACT-II) shipping package. A primary transportation requirement for the TRUPACT-II is that the concentration of potentially flammable gases (i.e., hydrogen and methane) must not exceed 5 percent by volume in the package or the payload during a 60-day shipping period. Decomposition of waste materials by radiation, or radiolysis, is the predominant mechanism of gas generation during transport. The gas generation potential of a target waste material is characterized by a G-value, which is the number of molecules of gas generated per 100 eV of ionizing radiation absorbed by the target material.

To demonstrate compliance with the flammable gas concentration requirement, theoretical worst-case calculations were performed to establish allowable wattage (decay heat) limits for waste containers. The calculations were based on the G-value for the waste material with the highest potential for flammable gas generation. The calculations also made no allowances for decreases of the G-value over time due to matrix

depletion phenomena that have been observed by many experimenters. Matrix depletion occurs over time when an alpha-generating source particle alters the target material (by evaporation, reaction, or decomposition) into a material of lower gas generating potential. The net effect of these alterations is represented by the "effective G-value".

Based on existing allowable wattage limits, it is estimated that a large portion (approximately 34 percent) of the CH-TRU waste cannot be shipped. The TRUPACT-II Matrix Depletion Program (MDP) has been established with the objective to investigate the phenomena of matrix depletion to support more realistic, age-dependent effective G-values. The MDP is a cooperative effort involving the U.S. Department of Energy Carlsbad Area Office National TRU Program, the Idaho National Engineering Laboratory, the Los Alamos National Laboratory, and the Rocky Flats Environmental Technology Site. The MDP is comprised of experiments designed to examine the behavior of effective G-values over time for different waste materials and the effects of isotope, agitation, and heating. The experimental data will be evaluated in conjunction with waste container headspace gas sampling and theoretical and predictive modeling to formulate bounding effective G-values for each simulated waste material and time segment. This paper describes the objectives, scope, components and preliminary results of the MDP that are expected to provide the justification for greater wattage limits for CH-TRU waste containers. The increased wattage will ultimately allow shipment of a much greater portion of CH-TRU waste.

BACKGROUND

The inventory of contact-handled transuranic (CH-TRU) waste, currently in retrievable storage at U.S. Department of Energy (DOE) sites, is planned for shipment to, and disposal at, the Waste Isolation Pilot Plant (WIPP). The Transuranic Package Transporter-II (TRUPACT-II) is a reusable shipping package designed for the transportation of CH-TRU waste containers to WIPP. Waste containers are 55-gallon drums, standard waste boxes, and ten-drum overpacks.

The TRUPACT-II was designed in accordance with the requirements for Type B packaging found in Title 10, Code of Federal Regulations Part 71 (1). Upon completion of the design and the required testing, the TRUPACT-II Safety Analysis Report for Packaging (SARP) was submitted to the U.S. Nuclear Regulatory Commission (NRC) in 1989. Based on the analyses presented in the SARP, the NRC issued Certificate of Compliance No. 9218 for the package in August of 1989 (8).

A major transportation requirement for the TRUPACT-II is that the concentration of potentially flammable gases must not exceed 5 percent (by volume) in the package or the payload during a 60-day shipping period after the TRUPACT-II is sealed. Decomposition of materials caused by radiation, or radiolysis, is the predominant mechanism of gas generation during transport.

CH-TRU waste is classified into four major types (I, II, III, and IV) based on chemical and physical characteristics and further subdivided into waste material types (I.1, I.2, I.3, II.1, II.2, and III.1) based on bounding flammable gas generation potential as shown in Tables I-1 and I-2 (8). The gas generation potential of a target material is characterized by its G-value, which is the number of molecular or ionic products (usually gaseous products) generated or consumed per 100 eV of ionizing radiation absorbed by the target material. Each CH-TRU waste container is assigned a TRUPACT-II shipping category, which is based on a combination

of waste material type and the packaging (number and type of plastic layers of confinement) of the waste materials within the waste container. To demonstrate compliance with the flammable gas requirement, theoretical worst-case calculations were performed using G-values to establish allowable wattage (decay heat) limits for each TRUPACT-II shipping category.

The maximum allowable wattage limits for each shipping category and for the TRUPACT-II were based on the initial G-values observed during experiments on the irradiation of materials found in TRU waste. The wattage limit calculations assumed a constant G-value from the time the waste was packaged until emplacement at the WIPP facility. The calculations made no allowance for decreases of the G-value over time, a phenomenon observed by many experimenters (5,6,7,9,10,11,12). Because the majority of CH-TRU waste retrievably stored at DOE sites is 5 to 22 years old, wattage limits based on initial G-values are extremely conservative. Taking into account the existing TRUPACT-II wattage limits, it is currently estimated that a large portion of the CH-TRU waste container inventory cannot be shipped. A joint effort was conducted at the Idaho National Engineering Laboratory (INEL), Los Alamos National Laboratory (LANL), and Rocky Flats Environmental Technology Site (RFETS) to determine the impact of existing wattage limits on the shipability of CH-TRU waste stored at those sites. Of the total volume certifiable CH-TRU waste (i.e., waste that meets the requirements of the Waste Acceptance Criteria for the Waste Isolation Pilot Plant (3)), it was determined that approximately 34 percent of the waste would not be shippable based on its failure to meet the TRUPACT-II wattage limits. Although the effort was performed at the three named sites, the results are applicable to all DOE sites that plan to ship CH-TRU waste to WIPP. The TRUPACT-II SARP contains a Gas Generation Test Plan; however, implementing the plan for 34% of the CH-TRU waste would be prohibitively expensive and time consuming.

All TRU waste will ultimately have to be transported to the WIPP; therefore, a method to provide for its acceptance for shipment in the TRUPACT-II is needed. The solution may lie in determining effective G-values as a function of time and thus more realistic TRUPACT-II wattage limits, primarily by accounting for age-dependence of the waste. A cost-effective method for arriving at age-dependent G-values and revised TRUPACT-II wattage limits is matrix depletion testing.

Matrix depletion occurs over time when an alpha-generating source particle alters the target material (by evaporation, reaction, or decomposition) into a material of lower gas generation potential. The matrix depletion process decreases gas generation by reducing the availability of the target material, thereby resulting in a decrease in the G-value with time. The fraction of energy emitted that is absorbed by the target material varies with time and is related to the short mean free path of alpha particles. When the alpha-generating source is dispersed in or on the target material in a particulate form, it will affect only that target material in a small semi-spherical area around each source particle. Additionally, some energy will be lost in the source particle itself and in surrounding nonhydrogeneous material, such as air, that is not available for the liberation of hydrogen. The net effect of these reactions is represented by the "effective G-value."

Two observations of CH-TRU waste over time support the concept of matrix depletion: 1) a consistent decrease in the effective G-value, and 2)

darkening of plastic surfaces (i.e., charring of the waste matrix). The effective G-value has been found to decrease exponentially with time. In fact, within 2 years the effective G-value for Pu-238 contaminated materials was found to approach an asymptotic value several times lower than the initial effective G-value. Based on the available experimental and empirical results, matrix depletion acts to decrease the rate of flammable gas (specifically hydrogen and potentially methane) generation inside CH-TRU waste containers.

OBJECTIVES AND COMPONENTS

The Matrix Depletion Program (MDP) was established with the objective to investigate the phenomenon of matrix depletion and to arrive at age-dependent bounding effective G-values. The activities under the MDP include matrix depletion experiments, headspace gas sampling and analysis, data management and analysis, and documentation. In order to specify the quality of data from the data collection activities under the MDP, a formal procedure based on U.S. Environmental Protection Agency (EPA) guidance was used in formulating data quality objectives (DQOs). The DOE Carlsbad Area Office (CAO) is responsible for the overall management of the MDP. CAO is also responsible for performing audits of MDP activities at INEL.

In the past, LANL has conducted a number of investigations of gas generation in CH-TRU waste to support the WIPP (10,6,11,12). Recently, a limited number of experiments were conducted to determine the matrix depletion effects of Pu-238 on cellulose and polyethylene (7,9). Because of this experience and the availability of the necessary equipment, LANL will be responsible for conducting the matrix depletion experiments. The experiments were designed to quantify the matrix depletion phenomenon using a variety of simulated TRU waste matrix materials (i.e., polyethylene [PE], dry cellulose, wet cellulose, polyvinyl chloride [PVC]), and solidified aqueous or homogeneous inorganic solids that contain water (i.e., cemented waste forms). The total duration of the MDP will be 2 years, with cylinder sampling occurring every 2 weeks. The sampling frequency was determined by the requirement for an adequate number of samples, so as to ensure accurate statistical analysis. Based on mechanistic arguments, the effective G-values for drums older than 2 years will be similar, but lower than those at 2 years; therefore, the value for 2 years will be conservative for older drums. The experiments will also quantify the effects of agitation and heating. Agitation levels to be used in the study are those that simulate transportation and handling events. The heating temperature is based on the highest values used in SARP analyses (140F 5F). Two isotopic sources of plutonium, a blend predominantly Pu-238 and a second blend predominantly Pu-239, will be used in the experiments because Pu-238 has higher decay heat production that causes many drums in the TRU waste inventory to exceed the established wattage limits and Pu-239 is the primary isotope present in TRU waste. The experiments will comprise 60 test cylinders split up into four groups: normal, heated, agitated, and heated/agitated. These groups are designed to simulate the range of conditions encountered during TRU waste transportation and handling operations. Effective G-values will be calculated for each test cylinder from the raw data derived from the experiments. The effect of each isotope, agitation, and heating will then be evaluated to formulate bounding effective G-values for each simulated waste material and time segment (2).

Headspace samples will be collected and analyzed from a representative sub-population of existing TRU waste containers at ambient temperatures. Samples will be taken from both drum and inner confinement layer. These activities are being performed by the INEL and the RFETS under the WIPP Transuranic Waste Characterization Program (TWCP) (4). Headspace sampling includes the drum headspace and the headspace of inner layers of confinement. In the TWCP, headspace gases of all CH-TRU waste drums will be sampled and certain drums will undergo innermost layers of confinement sampling. QA requirements specific to the MDP are described in the MDP QAPP, which is consistent with the TWCP requirements (3). Headspace gases will be analyzed for hydrogen and methane. The TWCP and MDP are coordinating efforts to make a variety of waste types and drum ages available to the MDP. A total of 740 drums will be examined prior to the completion of the matrix depletion experiments. Headspace gas sampling and analysis will be accomplished in accordance with the TWCP.

Drum headspace gas samples will be collected and analyzed from a limited number of existing TRU waste containers at elevated temperatures. This effort is being performed at the INEL under the TRUPACT-II Gas Generation Test Program (GGTP). The GGTP consists of controlled tests with actual containers of CH-TRU waste to determine gas generation rates under simulated transportation conditions. CH-TRU waste drums of waste types I, II, or III will be selected for testing if the decay heat loading of the container exceeds the TRUPACT-II wattage limit for the shipping category of the container. The drum must also meet other requirements before being tested, including having a fissile gram equivalents (FGE) less than 200 grams, weights less than 1,000 pounds, surface dose rates less than 200 mrem.

Data management and analysis involves several key elements. First, data obtained in each portion of the MDP must be validated to ensure that quality assurance requirements have been met and that the data are suitable for use in the MDP. Second, individual effective G-values must be calculated from raw data collected in the matrix depletion experiments. Third, the individual effective G-values must be summarized appropriately to formulate the bounding effective G-values. Fourth, the experimentally derived bounding effective G-values and drum headspace flammable gas (i.e., hydrogen and methane) concentrations predicted from the bounding values must be calculated and compared with respective actual waste drum values.

As noted above, data collected in matrix depletion experiments will be reduced to formulate bounding effective G-values. This first step involves calculating effective G-values for each sampling time and test cylinder. Second, individual effective G-values will be analyzed and used to formulate bounding values for each simulated waste material and time segment of interest; these values will be 95% upper confidence levels based on matrix depletion experimental effective G-values. A set of bounding effective G-values will be determined for PE, wet cellulose, dry cellulose, and cement for time segments of interest.

Existing mathematical models for predicting gas concentrations in waste containers will be finalized. The models are based on the aspiration model described in the TRUPACT-II SARP and simulate the time-dependent generation of flammable gas within the innermost confinement layer and subsequent time-dependent transport across the various confinement layers of TRU waste containers. The models will then be applied using the bounding effective G-values determined from the matrix depletion

experiments to predict flammable gas concentrations within actual TRU waste containers up to the age of the container when sampled. A theoretical model that simulates matrix depletion effects in an idealized geometry will also be developed to further illustrate the phenomenon. Development of the models will be led by INEL. However, it is anticipated that the model development will be completed with a high degree of collaboration drawing on the expertise from the INEL, the LANL and the RFETS.There will be two types of comparisons made using the bounding effective G-values derived from matrix depletion experiments. The first will compare the bounding values with the GGTP results and the second will compare hydrogen and methane concentrations predicted from the bounding values with hydrogen and methane concentrations measured in the TWCP.

Statistical comparisons will be made of bounding effective G-values derived from the matrix depletion experiments with the measurements from actual TRU waste drums (i.e., TWCP and GGTP) in order to show that the effective G-values from the matrix depletion experiments are in fact bounding. Because of the simulated waste materials used in the MDP, it is expected that headspace flammable gas concentrations predicted from the MDP bounding effective G-values will be greater than measured drum headspace flammable gas concentration at ambient temperatures. It is also expected that the bounding effective G-values measured in the MDP will be greater than the effective G-values calculated from sampling TRU waste containers at elevated temperatures.

In both cases, MDP effective G-values are expected to be greater due to the nature of the target material and the geometry of the cylinder contents. While other non hydrogen-generating materials are present in actual CH-TRU waste, these are not included in the MDP. In addition, the target material will be directly sprinkled with plutonium, allowing it to be embedded in the target. In actual CH-TRU waste, the plutonium is dispersed and not always in such direct contact with hydrogen-generating materials.

Quantification of the time-dependent behavior of the effective G-values (i.e., flammable gas generation rates within waste containers) is expected to support justifications for greater wattage limits for CH-TRU waste. The increased wattage limits will allow for shipment of a much greater portion of certifiable TRU waste without increased risk. Assuming the data justifies an application for higher TRUPACT-II wattage limits, the documentation to support an application to the NRC will be prepared and submitted to the CAO and National TRU Program.

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ISOTOPE PRODUCTION POTENTIAL AT SANDIA NATIONAL LABORATORIES: PRODUCT, WASTE, PACKAGING, AND TRANSPORTATION*

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ABSTRACT

The U.S. Congress directed the U.S. Department of Energy to establish a domestic source of molybdenum-99, an essential isotope used in nuclear medicine and radiopharmacology. An Environmental Impact Statement for production of 99Mo at one of four candidate sites is being prepared. As one of the candidate sites, Sandia National Laboratories is developing the Isotope Production Project. Using federally approved processes and procedures now owned by the U.S. Department of Energy, and existing facilities that would be modified to meet the production requirements, the Sandia National Laboratories' Isotope Project would manufacture up to 30 percent of the U.S. market, with the capacity to meet 100 percent of the domestic need if necessary. This paper provides a brief overview of the facility, equipment, and processes required to produce isotopes. Packaging and transportation issues affecting both product and waste are addressed, and the storage and disposal of the four low-level radioactive waste types generated by the production program are considered. Recommendations for future development are provided.

PROJECT NEED AND BACKGROUND

Nuclear medicine is an expanding segment of today's medical and pharmaceutical communities. Specific radioactive isotopes are vital, with molybdenum-99 (99Mo) being the most important medical isotope. The decay product of 99Mo is the isotope technetium-99m (99mTc), which is the medically useful element in its metastable form. The metastable state of a nucleus is at a higher energy potential than the ground state (1); this property produces gamma rays upon transition that are detected by medical equipment.

Nordion International Inc. manufactures much of the world supply of the 99Mo used for medical purposes in a reactor operated by Atomic Energy of Canada, Ltd. The reactor's useful life is expected to expire by the end of this century. The possibility of this reactor being shut down prompted the U.S. Congress to direct the U.S. Department of Energy (DOE) to provide for a domestic backup source for this essential isotope. An Environmental Impact Statement for the production of 99Mo at one of four alternative sites is being prepared by the DOE. Although the final selection of a site awaits the completion of the National Environmental Protection Act process, Sandia National Laboratories (SNL) has been indicated as the preferred site.

The SNL Annular Core Research Reactor (ACRR) was evaluated as a facility to produce 99Mo (and later other isotopes that can be economically extracted from the process). Medical isotope production at SNL is a new manufacturing venture. Should SNL be selected and the project achieve the manufacturing stage, the SNL facility would serve as a backup to the Canadian supply to provide up to 30 percent of the U.S. market under normal circumstances, with the capability to produce 100 percent of the domestic 99Mo requirement should the need arise.

The DOE has produced radioisotopes for users, both public and private, for decades. In the private sector, Cintichem, Inc., manufactured 99Mo and several other isotopes for a market segment in the U.S. as recently as 1989. The Cintichem process was approved by the U.S. Food and Drug Administration (FDA). Rights to the patented process (and the Drug Master File documenting the process) were acquired by DOE in 1991; DOE proposes to use this process to avoid the development time and expense of qualifying a new process. Ancillary equipment acquired by the DOE included packages for product, wastes, and spent fuel.

The proposed project would fabricate unirradiated targets to Cintichem specifications, followed by irradiation in the SNL ACRR. The irradiated targets (contained in a transfer cask) would be moved to the adjacent Hot Cell Facility (HCF) for processing. At the HCF, the radioisotopes of interest would be separated from the fission product inventory. The short half-life product, 99Mo in NaOH, would then be transported by air to U.S. radiopharmaceutical manufacturers. Fabrication process wastes would be temporarily stored at SNL for later disposal at authorized waste facilities.

THE REACTOR

DOE views SNL's ACRR, HCF, and other associated facilities to be a promising site for this isotope production program (2) for several reasons. The ACRR is a modern facility in an operational state with characteristics that are compatible with radioisotope production. It is capable of being dedicated to continuous isotope production, which is necessary to meet the demands for short-lived medical-use isotopes. The ACRR is collocated with the HCF and both can be modified with relative

ease. The ACRR at SNL is in proximity to excellent air transportation facilities (the Albuquerque International Sunport) for radioisotope shipments.

The ACRR facility includes the reactor and all support systems required for its operation. The reactor core is installed in a large open tank filled with about 10 meters of water to provide both core cooling and radiation shielding. The core is cooled by natural convection in an open water pool, and the water pool is cooled by an external heat exchanger. The current ACRR configuration consists of an annular array of UO2-BeO-fueled elements with an active fuel height of 52 centimeters. The dry, steel-lined, control cavity would be removed from the center of the core to provide a flooded region for target irradiation. Two configurations, one with a maximum of 19 targets and the other with a maximum of 37 targets at a time, are planned. The targets used to form the isotopes would almost completely fuel the reactor. With targets installed, only 180 or 130 conventional fuel assemblies, depending on the target configuration selected, would be required to operate the reactor. With installation of additional heat exchangers or cooling towers, the improved heat rejection capacity would allow the reactor to run at 4 MW. For isotope production, the ACRR would be operated in the steady-state mode at or below 4 MW.

TARGETS AND PROCESSING

Because the ACRR is a pool reactor, targets and fuel elements would be readily accessible for removal. Targets consist of stainless-steel tubes approximately 45 centimeters long and 3.18 centimeters in diameter, containing highly enriched uranium-235 (235U) as a 50-micron layer of uranium oxide (U3O8) electroplated onto the inside surface of the tube. Targets will be irradiated to provide a range of fission products that includes isotopes of molybdenum. Targets irradiated for several days would be removed from the core and transferred (using pass-through ports) to a rack in the adjacent Gamma Irradiation Facility (GIF) pool. A transfer cask would be lowered into the GIF pool and the irradiated target(s) would be loaded into the cask and transferred to the HCF using a manned transport vehicle.

The HCF will be reconfigured to streamline the process of irradiated target processing. One proposed HCF modification is the addition of new steel confinement boxes (SCBs) that would result in safer, more reliable, and more versatile extraction operations. The new SCBs would provide complete process control, including waste minimization and management. The units would collect byproducts from the radioisotope extraction, process the byproducts, and package them into waste containers. Modular design would allow easy replacement of components. Another important addition to the HCF is the Quality Control Laboratory, which is required by the approved FDA procedure.Irradiated targets containing almost 7.4E+14 becquerels (Bq) of fission products would be processed within the SCBs. The desired isotopes would be extracted from the fission product spectrum by chemical dissolution and precipitation procedures. First the noble gases and iodine would be condensed from the target fill gas and the fission products would be dissolved from the inside of the target. Then chemicals would be added to maintain specific fission products in solution and the molybdenum would be precipitated, filtered, and cleansed. Finally, the precipitated molybdenum would be redissolved for shipment to radiopharmaceutical companies.

Although 99Mo is the initial product of interest, iodine-131 (131I) and xenon-133 (133Xe) may also be directly extracted from the processing line as additional medically valuable products. The isotope iodine-125 may also be processed from xenon-124, a nonradioactive isotope of xenon; however, this process requires additional apparatus and would only be explored after sufficient success is achieved in 99Mo processing.

Each target would yield up to 29.6E+12 Bq of 99Mo after discharge from the reactor. The isotopes would be further purified to meet FDA standards. The isotopes would then be packaged and shipped in shielded casks by air freight to radiopharmaceutical companies. Approximately 20 to 25 targets per week can meet all U.S. demand with proportionately fewer targets for the standby level of 10 to 30 percent U.S. demand. The ACRR will have the capability to irradiate up to 37 targets continuously; however, this level would be used only under extraordinary conditions of national need and would place a greater burden on the processing facility.

WASTES

The production program will generate low-level radioactive waste primarily consisting of four types. The first type is a high-activity acidic liquid that will contain the bulk of fission products and unfissioned uranium. The second type of low-level radioactive waste is hardware process stream waste generated from isotope separation and purification. This waste includes copper, stainless steel, glass, plastic, and aluminum. Also generated is soft waste, such as personnel protective clothing and lay down material. The final type of waste is spent resins generated during the operation of the ACRR and GIF pool.

The 99Mo isotope extraction process wastes would account for the majority of the low-level radioactive waste volume generated by the proposed program. The extraction of isotopes from irradiated targets involves a number of wet chemical processes. These production activity chemical processes are benchtop processes that are conducted using small (< 500 milliliter) laboratory containers and other equipment of various forms, typically glassware. Because the isotopes that would be produced are used as radiopharmaceuticals for human use, the extraction process is closely controlled and monitored for contamination by chemical remnants and unwanted radioisotopes. Hence, all laboratory equipment such as glass flasks, tubing, and the like would be replaced after a single use to prevent possible contamination of future batches. This equipment would be placed in small containers (about the size of a one-gallon paint can) that, when full, would be placed in a waste container.

The acidic liquid process solutions that remain after the radioisotopes are extracted contain uranium and other radioactive elements called fission products. These liquid solutions would be neutralized as a final process step, rendering the solution nonacidic, solidified with an agent such as portland cement, and then placed in a waste container. The uranium would not be recovered from the solution nor from the solidified waste because it is not economically feasible to recover it at SNL or other DOE recovery facilities.

The waste container used during storage and eventual disposal for this solidified process waste would have a volume of about 0.21 m3 (55 gallons). The radioactive process equipment discussed above and the solidified process waste may be placed either in the same container or in separate containers, based on operational and disposal considerations.

The filled waste containers would be stored in a shielded area. The solidified process waste would be stored on site for approximately 6 to 12 months, by which time the radioactivity of the waste would have been reduced significantly as shown in Fig. 1. The process hardware waste is expected to have a lower level of radioactivity and thus could be transported to the selected waste disposal site after final packaging at a time that is being determined.

Fig. 1

The proposed program would generate between 24 and 180 containers (similar in size to a standard 55-gallon drum) of low-level waste (both process hardware and solidified process waste) from the isotope extraction process each year. These waste containers would contain no more than 7.4E+13 Bq of radioactive waste when shipped to the disposal site in B-3 waste transport packages. Fission products would be responsible for nearly 99 percent of the activity level. The remaining activity would be mostly from the activation of the stainless-steel target shell.Although solidified process waste shipments could be made to the disposal site as soon as 6 months after generation, the HCF has sufficient shielded storage area to store waste generated by 2 years of maximum (100 percent of U.S. need) production. Table I identifies most of the waste materials and quantities that are known to be included in the extraction process waste stream.

Table I

Past assessments indicate that the wastes produced in the extraction process would not generate radioactive mixed wastes. This means that the radioactive waste would not contain nor be mixed with waste that is considered hazardous according to the Resource Conservation and Recovery Act.

PRODUCT AND WASTE PACKAGING

Table II shows the general characteristics of the three Type B packages that will be used in the Isotope Production Project. The DOT- and NRC-certified Type B (3) package designated for use in transporting 99Mo and 131I is the CI-20WC-2 or -2A (4). The primary difference between the two models is the size and the amount of shielding. Both CI-20WC (Fig. 2) models are steel-encased with wooden outer protective jackets, a depleted uranium shielded cask, and an inner steel containment vessel. The protective jackets are contained within an 18-gauge steel drum. The inner containment vessel is a 7-cm outer diameter by 14.26-cm-long, 416 stainless-steel, gasketed and threaded container. The product packages are certified for 99Mo/99mTc in normal form as solids or liquids with a maximum quantity of material per package of 3.7E+13 Bq, and 131I in normal form or liquids with a maximum quantity of material per package of 7.4E+12 Bq.

Table II

Fig. 2

The package designated for transportation of product waste is the NRC-certified B-3 Type B package (5). The packaging (Fig. 3) consists of a 15.38-cm lead-shielded steel weldment in the shape of a right hollow cylinder with a bottom containing a drain assembly and a recessed, plug-type gasketed and bolted lid. Packaging features include lifting and tie down devices and a drain to the central cavity. The maximum weight of the loaded package is 13,636 kg.

Fig. 3

Spent nuclear fuel shipments are not expected to occur for several years. Such fuel will be stored until DOE designates a repository. When shipments are to be made, the BMI-1 package will likely be used. The package assembly consists of five major components: the stainless-steel enclosed lead-shielded cask, the stainless-steel encased lead-shielded cover and gasket, the radioactive material, internal canister or basket, and a custom matching skid on which the cask rests vertically.

PRODUCT AND WASTE SHIPMENT

Isotope 99Mo decays at the rate of about 1 percent per hour (half-life 66 hr). Consequently, shipment of the product must be expedited to prevent needless decay of the product. Nordion, which is the only North American company presently shipping bulk 99Mo, uses a combination of commercial and chartered air flights.

SNL expects to ship up to 29.6E+12 Bq of 99Mo per processed target and about 6 to 7 packages per week at the nominal 30 percent production level. The 99Mo product will be pharmaceutical quality, and FDA-approved procedures will be used for its production. The 99Mo is expected to be shipped initially on a daily basis to one of three different locations: St. Louis, Chicago, or Boston. Air freight express class of shipments are planned. If a stop is required, the shortest routing time from Albuquerque to the customer city will be preferable. Product movement from the SNL reactor area to the airport transfer point using Kirtland Air Force Base and Albuquerque International Sunport access roads is the preferred route, avoiding public roads. Product quality assurance may occur during the time the product is in shipment.

The primary waste disposal site designated for production and laboratory wastes is the Nevada Test Site facilities north of Las Vegas, Nevada (6). The site is compatible for the classes of waste generated by the production processes, and the site is operational. Two alternative waste disposal facilities are located at the Hanford Site near Richland, Washington. The 200-West Waste Generation Facility is quite extensive and has several current burial sites as well as some older burial grounds that are monitored. A second, alternate site is operated by U.S. Ecology as a commercial facility.

If the waste containers were to be transported by truck, it is most likely that one truck would carry one B-3 package per shipment, because of the weight of the B-3 package. Two packages per truck may be possible; however, depending on the exact payload weight, two packages would likely exceed the maximum gross weight allowed for one truck. The shipments would go directly from the HCF to either the primary site or to one of the alternative sites using the most direct route selected by the motor transport company. Approximately 85 shipments per year will be required. All packaging used to store and transport waste generated by the isotope production processes will adhere to DOT requirements as specified in the applicable parts of 49 CFR (7).

FUTURE DEVELOPMENT

Nuclear medicine is growing in importance and versatility, leading to an expanding market both in the U.S. and worldwide. The need for nuclear medicine will continue to grow as populations increase. New radiopharmaceutical and medical substances are increasing the spectrum of these items available to the medical community. Concurrently, the number of countries acquiring such technologies is expected to increase, perhaps dramatically. A substantially increased demand for radiopharmaceuticals is likely to result.

Future production, storage, loading, handling, and movement considerations of both isotope products and waste should focus on the need for modified or new containers and those that are suitable for automation. Significant benefits can accumulate by minimizing the human element in isotope production operations. Robotics Address As Low As Reasonably Achievable concerns and reduce the possibility of human error. In addition, real-time cask identification survey and real-time radiation survey could be safely and quickly carried out mechanically. SNL's expertise in robotics and automated waste handling should be applied to the Isotope Production Project to increase production, safety, and waste management.

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Session 54 -- STREAMLINING DATA MANAGEMENT AND PRIVATIZATION: CONTINUING PROGRESS WITH A SMALLER USDOE BUDGET

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54-1

FEASIBILITY STUDY FOR PRIVATE-SECTOR TREATMENT SERVICES FOR ALPHA-CONTAMINATED LOW-LEVEL MIXED WASTES

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ABSTRACT

Rust Federal Services, under contract to the United States Department of Energy (DOE), Idaho Operations Office, performed a study to develop and evaluate the feasibility of a suggested private sector solution for the treatment of alpha-contaminated low-level mixed waste (ALLMW) stored or produced at the Idaho National Engineering Laboratory (INEL). The feasibility study is an initial step in the potential procurement of privatized treatment services for these wastes.

Rust's derived objective of the feasibility study was to define an optimal treatment system and analyze the feasibility of that system for accomplishing the processing objectives specified by DOE. All aspects of the selected treatment system were addressed in the feasibility study, including technical, regulatory, public involvement, and financial considerations. Two central elements of the study were a technology screening task to select the optimal treatment system and an analysis of the institutional, business, financial, and contractual issues that are likely to accompany the privatization of treatment services for DOE.

INTRODUCTION

As a result of past nuclear defense program activities, a considerable volume of waste containing both radioactive and hazardous contaminants ("mixed waste") has been buried and/or stored at the INEL. Much of this waste is contact-handled heterogeneous material such as paper, plastic (chlorinated and nonchlorinated), cloth, wood, sludges, bulk metal, etc. This waste contains less than 100 nCi/g TRU and is therefore classified as low-level waste. The INEL has approximately 27,000 m3 of ALLMW. Approximately 95 percent of the waste is contaminated with Resource Conversation and Recovery Act (RCRA) hazardous material, and some waste contains Toxic Substance Control Act (TSCA) polychlorinated biphenyls. However, EPA regulations prohibit the disposal of material that contains RCRA constituents directly on or in the land without prior treatment to remove and/or destroy the hazardous constituents. Therefore, treatment of the ALLMW is required to remove or destroy the hazardous constituents (i.e., those regulated under RCRA).

This paper first describes the technical and institutional issues that were considered in developing and evaluating the optimal system for treating the ALLMW. It then addresses the business, financial, and contractual considerations involved in obtaining private sector participation in design, construction, and operation of the selected treatment facility.

TECHNOLOGY SELECTION AND EVALUATION PROCESS

The process used by the Rust team for selecting and evaluating the optimal treatment solution for INEL ALLMW focused on a technology screening and selection task to define the treatment technology system that could most cost-effectively meet all treatment objectives defined in the Feasibility Study Statement of Work (SOW, Ref. 1) for the specified waste streams. This task was followed by an analysis of the regulatory and other institutional requirements and constraints that could affect the treatment facility and its operations. The steps in the technology screening and selection task are summarized below.

Step 1. Identify types and quantities of waste media/contaminants requiring treatment. Approximately 27,000 m3 of wastes are stored at the INEL that meet the criteria for treatment in the described facility. Five ALLMW streams (metals, combustibles, heterogeneous, sludges, and concrete/bricks) account for approximately 94% of the total waste volume. The remaining waste streams, which account for approximately 6% of the total waste volume, include glass, particulate matter, plexiglass, non-metal molds/crucibles, resins, evaporator salts, gloves, scintillation cocktails, radioactive sources, and unknown wastes. The data in Table I is provided to help summarize the volume of each ALLMW waste stream and the relative percent of the total volume that they represent.

Table I

As the first step in the technology screening and selection process, characteristics of these wastes were tentatively identified from information contained in waste profile sheets. Many of the waste profile sheets included only information about the radioactive components of the waste, so information about the hazardous constituents was incomplete or inaccurate. As the project progressed, additional information became available and was incorporated into decisions.

Step 2. Define waste treatment objectives. The following Rust waste treatment processing system objectives were specified in the SOW:

Meet RCRA Land Disposal Restrictions for destruction or removal of hazardous constituents.

Meet waste acceptance criteria provided in the SOW for durability, compressive strength, leachability, resistance to immersion and biodegradation, and radiation stability.

Maximize final waste form durability.

Maximize options for treating a variety of contaminants and/or types of waste.

Step 3. Establish screening criteria for candidate technologies. Two go/no go criteria were identified to immediately eliminate from consideration those treatment technologies that could not support the overall waste treatment objectives. To be considered, an ALLMW treatment technology had to have been successfully demonstrated on at least a pilot-scale basis and had to be applicable to at least 10% of the targeted waste streams.

Subsequently, the Rust evaluation team produced a definitive list of ranking criteria against which to evaluate candidate treatment technologies that passed the go/no go decision point. Each criterion was assigned a weighting factor consistent with the level of importance of that criterion to the overall technology selection process. The five ranking criteria and their respective weighting factors (wf) were:

- Operability/Maintainability/Reliability wf - 5
- Properties of Secondary Waste Streams Produced wf - 4
- Waste Feed Characterization and Pretreatment Requirements wf - 4
- Nuclear Criticality Issues wf - 3
- Permitting Issues and Public Acceptability wf - 2

These criteria were used for two purposes: first, to identify technologies with the highest overall scores for further consideration, and later (in Step 6) to establish relative rankings for each criterion among the candidate technologies.

Step 4. Identify candidate treatment technologies. Using the criteria developed in Step 3, the team identified a list of candidate technologies that met all ranking criteria. The candidate technologies, identified for further consideration by type, are listed below.

Thermal Processes

- Plasma Hearth
- Joule Melter
- Controlled-Air Incinerator
- Fluidized Bed
- Metal Melter
- Vortec Incinerator
- Rotary Kiln

Chemical Treatment Processes

- SOIL*EXSM with Wet Air Oxidation

Decontamination Processes

Acid Rinse
CO2 Blasting
Chemical Rinse

Stabilization/Encapsulation Processes
Polyethylene Encapsulation
Sulfur Polymer Cement Stabilization.

Step 5. Establish treatment system selection criteria. Other criteria deemed important, as indicated by the SOW, were also identified and used as factors for evaluating the candidate technologies. These criteria were:

Versatility/robustness of system (number of waste streams treated, pretreatment requirements)
Waste volume reduction
Operational safety risks (e.g., use of high-pressure systems, waste handling requirements)
Life cycle cost
Final waste form characteristics.

Step 6. Evaluate and rank the treatment technologies using the weighted selection criteria. In this step, applicable waste streams were identified for each candidate technology and each technology was evaluated against the selection criteria for the applicable waste streams. Within each waste stream, a numerical ranking was assigned to each technology versus each selection criterion reflecting the degree to which the technology satisfied the criterion for that waste stream. The numerical values ranged from 3 (high) to 1 (low). The numerical values for each waste treatment/waste stream combination were tallied and reported as a final score. The score represented the applicability of each waste treatment technology to each target waste stream.

Step 7. Define potential treatment system implementation scenarios. No single technology can effectively treat all of the ALLMW that is the subject of this study; a suite of treatment technologies is required. Therefore, from each of the five waste stream treatment technology scoring matrices, the two highest-scoring waste treatment technologies for each individual waste type were identified. Various combinations of the high-scoring technologies were evaluated to predict their effectiveness when used together to treat the total INEL ALLMW inventory. In total, five waste treatment systems were evaluated.

Step 8. Select the treatment system with the best combination of advantages considering all selection criteria, evaluation factors, and implementation scenarios. Using the waste treatment objectives and evaluation criteria defined earlier, the five waste treatment technology systems were evaluated to select the most advantageous overall treatment system. The waste treatment system that was recommended consisted of a plasma hearth process, an evaporation system, a thermal desorption and mercury amalgamation system, and a series of decontamination systems. This combination was selected because of its ability to handle the extremely heterogeneous assortment of waste materials to be treated, as well as the glass waste form produced and other advantages of the plasma hearth technology.

Regulatory Considerations. Because the feasibility study examined every aspect of siting, designing, constructing, and operating an ALLMW treatment facility, it was necessary to define the regulatory requirements and constraints that could impact the facility's cost,

completion schedule, and operations. Plans were developed for implementing:

- A licensing and permitting strategy and schedule
- A public acceptance program
- Waste transportation
- Safety Analysis Report requirements
- Support for National Environmental Policy Act requirements.

In addition, a siting analysis was performed to identify the most cost-effective location for the facility. This analysis considered the public acceptance, licensing, permitting, waste transportation, and safety aspects of building the facility at an on-site (i.e., INEL) location versus an off-site location.

BUSINESS, FINANCIAL, AND CONTRACTING CONSIDERATIONS

As part of the feasibility study, Rust prepared a business plan for developing, constructing, and operating the ALLMW Treatment Facility under a prime contract with DOE. The business plan included a market assessment, an analysis of risks and liabilities, a plan for providing treatment services, contractual performance terms and conditions, pricing, and other contractual considerations.

Rust's ALLMW Treatment Facility would be planned as a turnkey project, design through D&D, performed by a prime contractor. Regardless of whether the prime contractor enters into a joint venture, partnership, or teaming arrangement with other organizations, the requirement for ensuring a reasonable return on investment will exist. Since the majority of the waste that could be treated _s under DOE's jurisdiction, the prime contractor will need DOE's commitment to make a minimum quantity of waste available for processing. The contractor will then establish an equitable return on investment based on that minimum waste quantity. Given this expectation, Rust examined various investment scenarios and concluded that three are most likely to meet the needs of DOE and the contractor, as follows:

Commercial Partnership Model. Rust examined a facility designed, constructed, and operated under commercial-type fixed price contracts, but owned by the government. This investment scenario provides for shared risk between DOE and the contractor, a reasonable return on investment for the contractor, and, potentially, low overall cost to the government. With this alternative, pre-operational costs would be reimbursed as incurred on a fixed-price basis, except permitting, which would be performed on a cost-plus-incentive-fee basis. Operating costs would be recovered through fixed-unit-price processing revenues. The government would own the treatment facility and would obtain treatment services from the contractor on a fixed-unit-price basis under a multi-year performance-based contract.

To maximize the benefits of this contracting arrangement, the terms and conditions should be as definitive as possible to avoid contingencies, the concept of pay-for-performance should be incorporated wherever possible, and incentives should be considered where appropriate.

Privatized Model. Rust suggested that a "totally privatized" investment scenario, where the contractor would pay all pre-operational costs with the expectation of recovering those costs through operation of the facility during the base period of performance, is the preferred approach. This approach would save the government time and money by optimizing contractor performance and requiring only a limited DOE staff (project manager and contract specialist) to oversee contractor

activities. This investment scenario presents some risks that would be borne almost exclusively by the private contractor. These risks would potentially consist of little to no cost recovery until operational start-up, potential increases in the rate of interest on borrowed money, and changes in strategy or objectives, either by DOE or the regulatory community, during the pre-operational period that may adversely affect the contractor's ability to recover pre-operational costs.

However, these risks could be mitigated by inflation adjustments, clear definition of DOE requirements in the contract, and provision for unit price adjustments as necessary. For example, annual reviews of unit prices in relation to interest/inflation rates and state taxes, with unit price adjustments to compensate for increases, would alleviate some risks. Clear definition of DOE requirements in the contract scope will aid in preparing accurate lump sum estimates, evaluating them, and recognizing when changes are necessary and valid.

Government Model. The "government model" would be a conventional cost reimbursable contract with an incentive fee to incorporate the pay-for-performance concept. As in the other approaches, a performance bond would be required. This method would likely result in more competition (i.e., more bidders) due to the virtual elimination of risk for the contractor; the lowest fee; and the lowest contractor-applied contingency. It would involve more oversight and administrative costs for the DOE. It could result in additional contractor cost as additional DOE intervention potentially could occur. The contractor would not have to add substantial contingency for cost of money, regulatory delays, etc. However, DOE would assume the majority of the risk with this approach.

CONCLUSION

Treatment of the ALLMW at INEL can be achieved in the most cost-effective and timely manner by locating the treatment facility on the INEL site and using the plasma hearth process (PHP) technology as the centerpiece of this facility. This facility will cost substantially less than is projected in current government estimates for an ALLMW treatment facility. The "totally privatized" investment scenario would offer the most cost effective treatment services to the DOE by allowing the private sector contractor to cover all pre-operational costs and subsequently recover these costs through operation of the facility during the base period of performance.

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54-2

NATIONAL MIXED WASTE TREATMENT STRATEGY - AN INDUSTRIAL PERSPECTIVE

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ABSTRACT

The Federal Facility Compliance Act (FFCA) required that each site develop an inventory of its mixed waste volumes and a plan for implementing treatment capacity. These plans identified available capacities of commercial facilities; however, they were unable to predict future commercial facilities since the private sector was generally

unwilling to invest in new facilities without a committed volume of waste. A review of the draft site treatment plans indicates that the DOE will need eighty-seven new treatment facilities to manage its mixed wastes.

Many sites are now considering privatization as an alternative to meeting FFCA requirements. INEL, Oak Ridge and Hanford are expected to release privatization solicitations in late 1995 or early 1996. These privatization initiatives represent the opportunity to serve as a core for a national mixed waste strategy which can potentially save the DOE millions of dollars. By expanding the treatment capabilities and capacities of these three privatization initiatives, the DOE can avoid capital investments at multiple sites. It is expected that this concept could handle 90% of the DOE's mixed waste. The remaining 10% could be managed by transportable equipment mobilized from site to site or to one or more of the privatized fixed facilities.

This paper will expand on this proposed concept and will include an update on the DOE's mixed waste privatization initiatives. It will also provide an industry perspective on how to structure privatization initiatives that will attract industry.

BACKGROUND

The Federal Facility Compliance Act of 1992 established requirements for U.S. Department of Energy (DOE) sites to achieve compliance with the Resource Conservation Recovery Act (RCRA) for mixed wastes. Sovereign immunity was granted until October 1995 to allow the DOE to devise a plan to ensure that adequate treatment capacity would be available. Each site was required to inventory its stored waste and predict volumes for newly generated wastes. Furthermore, it required that each site evaluate treatment options and establish a schedule for treating mixed wastes. These requirements led to each site preparing a site treatment plan for submittal to their respective states or EPA for approval. In general, these plans identified existing treatment capabilities or described technology development and treatment capacity activities needed to achieve compliance. Evaluations of existing treatment capabilities were limited to permitted or planned treatment facilities either operated at a DOE site or within the commercial sector. These activities generated the data required to quantify the volumes and characteristics of mixed wastes. However, the data did not provide an evaluation of private sector capabilities for adding mixed waste treatment capacity. Nor did it provide the market demand information required for the private sector to invest in new treatment capacity. In traditional environmental markets, market demand is largely created by enforcement of regulations. Since DOE sites were operating under sovereign immunity, the incentive or market demand for creating commercial mixed waste treatment facilities was eliminated. Furthermore, since each site developed its own plan, it was impossible to determine the economies of scale that could be achieved by combining waste streams from multiple sites with common characteristics. To achieve compliance, the DOE identified numerous treatment systems at an estimated cost exceeding $7B.

Realizing the need to "do more with less," the DOE began aggressively pursuing privatization for treatment of mixed wastes. Privatization will indeed save monies. However, if the DOE is to benefit from privatization, procurements must be devised that share risk to attract the maximum number of competitors. In general, high risk will result in few

competitors, and thus higher cost to the government. This relationship is depicted below.

Fig. 1

STATUS OF CURRENT PRIVATIZATION ACTIVITIES

In 1995, the DOE moved forward with privatization plans as evidenced by solicitations issued by INEL and Hanford. The following is a summary of the terms and conditions and assessment of the associated business risk:

Table I

RECOMMENDATIONS FOR IMPROVING PRIVATIZATION

Private sector companies are not averse to assuming risk if the risk can be quantified to allow a business decision. For example, our parent company, WMX Technologies, Inc., is aggressively pursuing privatization of POTWs. These opportunities represent large capital investments and risk. However, the main differences between the POTW market and the DOE mixed waste market is that in the former we are guaranteed long term contracts. Liability is shared and rewards match risks. On the other hand, the DOE is priding itself on shifting risk to the private sector. The following must be accomplished to maximize private sector investment and participation in DOE mixed waste treatment opportunities:

- Provide adequate performance specifications
- Guarantee minimum mixed waste volumes - combine waste streams of common characteristics that can be treated by a single treatment train
- Fund initial technology demonstrations so that the private sector can gain information on waste characteristics and treatment technology efficiency
- Change terms & conditions to avoid risk language such as "termination for convenience"
- Allow waste to be treated by NRC & EPA regulations and avoid transfer of DOE orders
- Commit to schedule for delivery of waste
- Shift control to private sector and avoid having cost plus award fee contractor overseeing fixed price contractor

NATIONAL MIXED WASTE TREATMENT STRATEGY

An innovative approach for privatizing mixed waste treatment has been proposed by Lockheed Martin's Center for Waste Management at Oak Ridge. The core of this program is to establish national contracts for mixed waste treatment technologies. This approach offers the benefit of defining market demand for the private sector and allowing the DOE to take advantage of economies of scale that would be realized by combining wastes from multiple sites. In order for this approach to be successful, the DOE must receive cooperation from multiple sites. Furthermore, it will be necessary for the sites to agree on contract terms and have funding available simultaneously. One alternative would be to pool waste management funds from multiple sites and establish a single national contract for treatment technologies. A roadmap must be developed, clearly defining market demand, contract requirements and delivery of waste volumes so that the private sector can quantify risk and potential return on investment. Finally, this roadmap should include a strategy depicting which waste streams will be managed at regionalized treatment facilities and by mobile treatment systems. Regionalized privatization facilities should be procured in a standard fashion.

ALTERNATIVE TO NATIONAL PROCUREMENTS

The DOE does not need a new national procurement initiative to realize the benefits of privatization. Since procurements are moving forward at

INEL and Hanford, a new national procurement would be disruptive. Instead, to avoid redundancy the DOE should attempt to coordinate privatization procurements to ensure inclusion of wastes treatable at other sites. Secondly, standard terms and conditions, performance specifications, and other contract requirements should be structured to ensure adequate competition. By coordinating these large regional privatization initiatives (INEL, Hanford, Oak Ridge), it is anticipated that 90% of mixed waste by volume could be managed. The remaining 10% could be treated using the Oak Ridge model of combining like waste streams and procuring single vendors for each waste type. These technologies could be offered as mobile services for treatment of wastes at individual DOE sites. The services could be mobilized to one or more of the large privatization complexes or set up at a vendor's own site.

CONCLUSION

Privatization of mixed waste treatment can reduce the DOE's cost by at least half. If the DOE develops sensible contracts and defines market demand, private industry will, in the spirit of competition, make treatment capacity available. A national strategy is therefore recommended. The strategy should include privatization of regionalized facilities encompassing the maximum number of waste streams from multiple sites. It should also provide for procurement of niche technologies and services for remaining wastes. If DOE mixed waste privatization efforts are not designed correctly, the DOE will end up with few competitors, thus negating the benefits of privatization. Furthermore, private firms that accept unrealistic risk will run the chance of failure, possibly resulting in default or renegotiation of contract.

Session 55 -- WORKSHOP - REMEDIATION WASTE MANAGEMENT UNDER THE CAMU RULE AND RCRA/CERCLA INTEGRATION

Co-chairs: J. Erich Evered, Jacobs Engr.

55-2

CAMU ISSUE PAPER

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ABSTRACT

The Environmental Protection Agency (EPA) has promulgated regulations governing the creation and designation of Corrective Action Management Units (CAMUs) to add flexibility to the Resource Conservation and Recovery Act (RCRA) corrective action process. In the final rule,

remediation waste is subject to land disposal restrictions (LDRs) and minimum technology requirements (MTRs) in a less limiting way than has been the case under existing regulations. Is the CAMU concept useful? This paper will explore the use of the CAMU concept and practical applications. The implementation of CAMUs will be discussed for each EPA Region and associated states. Currently, a few industrial sites have obtained CAMUs through a Class III permit modification. Details of these permits and discussions of permit applications being finalized will be presented. Draft Order modifications have been finalized for several facilities, and these will also be discussed. Some sites have presented the CAMU as an applicable or relevant and appropriate requirement (ARAR), and the difficulties encountered will be described. The possibility for application at the U.S. Department of Energy (DOE) sites will be detailed. Discussions will include the criterion for designation and "Keys for Success." Applications for CAMUs have been tracked for over two years, and specific projects, both commercial and government, will be described.

CAMU ISSUE PAPER

The Environmental Protection Agency (EPA) has promulgated regulations governing the creation and designation of Corrective Action Management Units (CAMUs) to add flexibility to the Resource Conservation and Recovery Act (RCRA) corrective action process. The CAMU is similar to the Superfund concept of "the area of contamination," in which broad areas of contamination, often including specific subunits, are considered a single land disposal unit for remedial purposes. The CAMU provisions allow for corrective actions that may not be subjected to all RCRA requirements. Rules for CAMUs achieved final status on February 16, 1993 (FR 8658), although the original corrective action rule has not been finalized in its entirety. In the final rule, remediation waste is subject to land disposal restrictions (LDRs) and minimum technology requirements (MTRs) in a less limiting way than has been the case under existing regulations. A summary of the advantages of the CAMU rule are as follows:

Remediation wastes, including hazardous wastes, may be placed in a CAMU without triggering LDRs.

Because of the flexibility in moving and placement of wastes, hazardous waste remediations may be completed in less time than corrective actions subjected to all RCRA requirements.

Cost savings can be realized by avoiding the expense of off-site incineration.

More treatment of waste is expected, resulting in remediation that is equally or more protective than current standards.

MTRs may not apply to CAMUs.

KEYS TO SUCCESS

Is it protective?

Successful CAMU applications should include a clear statement of how the CAMU will be more protective of health and the environment.

What is the timing?

CAMU applications where corrective actions will be completed before March 1997 have a better chance of success.

Is the application complete?

Failure to provide a complete response will cause delays and may jeopardize the decision process.

Fig. 1

CAMU ISSUES

A lawsuit was filed against EPA in May 1993 by the Environmental Defense Fund (EDF), Natural Resources Defense Council, and Hazardous Waste Treatment Council. The lawsuit alleges that the provisions of the CAMU rule are not protective of human health and the environment and that process waste may be mixed with cleanup waste in a CAMU. A settlement is being sought through modifications of the Hazardous Waste Identification Rule (HWIR). The first part of the proposed HWIR was signed November 13, 1995. There will be separate rulemaking for HWIR media with final action anticipated by March 1997. The HWIR is expected to divide contaminated media into two categories: highly contaminated material and less contaminated material. As defined under the rule, highly contaminated waste will be treated under Subtitle C, whereas less contaminated wastes would be exempted from Subtitle C as long as they are adequately managed under a state program.

DISPOSAL OF CURRENTLY STORED WASTES

While stored waste that is not remediation waste (i.e., process waste or other waste not generated as a result of corrective action activities) cannot be disposed of in a CAMU, it is unclear whether currently stored remediation waste could be disposed of in a CAMU. The final rule does not address this issue. A discussion with EPA indicated that it may be possible, although difficult, to obtain approval for the disposal of previously stored remediation wastes in a CAMU. The primary obstacle to gaining approval for the placement of stored remediation wastes in a CAMU appears to be centered on EPA and public concerns regarding the difficulty associated with ensuring that stored, nonremediation wastes are not unknowingly (or knowingly) placed in the CAMU. This concern appears to be strong enough that it is unlikely that facilities will be able to treat and/or dispose of previously stored remediation wastes within a CAMU.

IMPLEMENTATION POLICY

The basis for implementing the CAMU provisions rests in EPA's findings that RCRA Subtitle C requirements, when applied to existing contamination problems at a facility, can limit the flexibility of the decision maker, act as a disincentive to more innovative remedies, and provide strong incentives for leaving wastes in place. The desire to expedite corrective action and achieve reasonable cleanup solutions led EPA to develop the concept of CAMUs in the 1990 proposed corrective action rule. Because corrective action at RCRA facilities often addresses broad areas of contamination containing discrete waste management units, the consideration of a contaminated area as a whole and selection of a remedy that best addresses the entire area of contamination is appropriate. In these situations, EPA believes that the entire area of contamination can properly be considered as a waste management unit.

If the Regional Administrator determines that a CAMU designation should be made, the facility's permit or order must be modified to incorporate the CAMU. EPA intends to implement the CAMU rule in all states in which the Agency currently administers the Hazardous Waste Solid Amendment Sections 3004(u) and (v) corrective action authority. No formal CAMU guidance document exists, and the U.S. Department of Energy Headquarters does not plan to issue amendments or interpretations to the ruling. The tracking of CAMU requests is handled by the regional offices. The status is as follows:

Region I

Remington Arms (Connecticut) established a CAMU under a modification to a consent Order signed by EPA on October 1994. This CAMU is for the soil washing residue.

Region II

The Region does not have a formalized Standardized Operating Procedure (SOP) for CAMU application. EPA has developed a policy that requires initial treatment, a waste collection system, biomonitoring, and triple lining of the landfill.

American Cyanamid in New Jersey finalized a CAMU in December 1993. This was a Class III permit modification. The CAMU was a consolidation of metal sludge from small landfills into a triple lined landfill.

Region III

The Region received the following CAMU applications.

Sci-Tech (formally American Cyanamid) has finalized a Class III permit modification for the use of a CAMU at the Willow Island, West Virginia facility. An existing basin was the designated CAMU and will receive potentially contaminated soil from the Underground Storage Tank removal operations.

Standard Chlorine Superfund Site in New Castle, Delaware, is an operating RCRA facility. EPA, as the lead agency, generated a proposed plan for public comment. The facility commented that the CAMU provision should be considered as an applicable or relevant and appropriate requirement under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). The Record of Decision did not include the CAMU. It was reported that the facility had selected an inappropriate area.

Region IV

Several facilities within the Region have submitted CAMU applications.

Air Products (Florida) submitted a proposal for a Statement of Basis in which wastes from several areas contaminated with dinitrotoluene would be consolidated into a CAMU and biotreated. This was finalized August 8, 1994. EPA signed this order September 19, 1995.

Southern Piedmont (Georgia) has submitted a proposal for a Class III permit modification for a CAMU that would be used for bioremediation of contaminated soils.

U.S. Electrical Motors (Mississippi) received a CAMU designation as part of the original Part B permit for the facility. The CAMU will be used to store soil excavated during construction of interceptor trenches.

North Carolina granted a Class III permit modification to General Timber on June 5, 1995.

Caven Ham Forestry (Mississippi) has proposed a CAMU for inclusion in the reissuance of the permit.

Region V

Ford Motor Company (Michigan) has a CAMU. Earlier the Michigan authorities had discussed this facility, but EPA had been unwilling. EPA approved the CAMU application after the CAMU Rule. There are approximately five proposals for CAMUs in this region, including Ohio and Indiana.

In Ohio, an earlier petition for a facility in Cincinnati was not granted, but one application is close. A federal site (Fernald) is in the "talking" stages at this writing.

A commercial site is being discussed in Indiana.

Region VI

The Region has adopted a formal SOP to maximize the quality of review for CAMU requests. A regional CAMU Working Group reviews applications at Peer Review meetings led by the applicant's EPA facility manager. The following facilities have received or are pursuing CAMUs.

Louisiana will require consolidation waste to meet a standard other than the best demonstrated available technology.

The W.J. Smith facility in Texas will employ a unique technology.

EPA Region VI is issuing an executive order to WITCO of Louisiana.

Region VII

All states in the Region have expressed favorable interest; however, there are no CAMUs or final applications in this Region.

Region VIII

Regional personnel identified as current policy the expedient approval of CAMU applications where remediation and completion of the corrective action is likely by 1996. The time frames used in this verbal directive coincide with the expected duration of the EDF lawsuit.

The Flying J Petroleum Refinery (Williston, North Dakota) has identified the location of a proposed CAMU in the Corrective Measures Work Plan for the Facility. These work plans have been approved. Two CAMUs are designated at this facility. One CAMU will occupy three quarters of the property. The other CAMU is for lead-contaminated soil being held for off-site shipment.

Union Pacific (Wyoming) had a CAMU designated in the work plans approved by EPA Region VIII on June 30, 1995.

For Rocky Flats (Colorado), there are early discussions concerning a CAMU. Details will be presented in a case study.

Region IX

California reported the following activities.

IT, Vinehill has requested a CAMU designation for a landfill. This landfill has gone beyond the boundaries of the county, and the facility proposes to consolidate the waste into a permanent landfill.

California has prepared an executive order for this facility.California has a two-step process: (1) designation of a CAMU and (2) preparation of an enforceable document.

Region X

Except Alaska, state interest has been favorable. The Region suggested those issues to be identified in the application are: (1) Is it justified? and (2) Will it facilitate treatment?

The Hanford facility in Washington has considered a CAMU application, but decided on a CERCLA designation.

Envirosafe Services of Idaho had a CAMU designated in the RCRA permit. The CAMU is a Titan Missile silo.

Table I

With the exception of Region VI and California, there is little formal guidance for CAMU applications. As demonstrated above, activities vary from Region to Region. The informal applications are presented to indicate the variety in timing of CAMU applications with respect to other corrective action activities.

55-3

INTEGRATING CERCLA AND RCRA TO BETTER MANAGE HANFORD'S ENVIRONMENTAL RESTORATION WASTE

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ABSTRACT
The Environmental Restoration (ER) Program at the Richland Operations Office and the Office of Northwestern Area Programs at Headquarters, EM-44, have worked together successfully with regulators and stakeholders to establish a disposal facility at the Hanford site capable of managing Hanford's environmental restoration waste. The Environmental Restoration Disposal Facility (ERDF) is a technically sound, cost-effective, and "environmentally-friendly" disposal facility that is the result of an innovative integration of CERCLA and RCRA requirements. Most importantly, in this instance, the integration of CERCLA and RCRA requirements represents a new paradigm in project management, one that exemplifies the phrase "better, faster, cheaper".

INTRODUCTION
Successfully integrating the requirements of the Comprehensive Environmental Response, Compensation and Liability Act (CERCLA) and the Resource Conservation and Recovery Act (RCRA) has led to the establishment of an innovative disposal facility for managing remediation wastes at Hanford. Hanford's Environmental Restoration Disposal Facility (ERDF) is the result of a three to four-year effort between the U.S. Department of Energy (DOE), the U.S. Environmental Protection Agency (EPA), and the Washington State Department of Ecology to construct an on-site facility for managing wastes from Hanford's environmental restoration program. During that time, the three parties considered several alternatives to determine which regulatory pathway would expedite the construction and ultimate operation of the ERDF.
The ERDF is the primary disposal facility for wastes generated under Hanford's Environmental Restoration (ER) program. It is a large-scale expanding landfill, authorized under the CERCLA. Designed and constructed to comply with the technical requirements of the RCRA, the ERDF occupies a 1.6 square mile area on the central plateau of the Hanford site, approximately 200 feet above groundwater. The ERDF includes a double liner and leachate collection system. The initial trench is composed of two cells, each 500 feet wide x 500 feet long and 70 feet deep, and is designed so that it can be expanded in two-cell increments, while still allowing disposal operations to continue. This "expanding trench concept" allowed the DOE to significantly reduce overall project costs by shrinking the initial facility footprint from 6 square miles to 1.6 square miles, while still providing adequate disposal capacity for the restoration program.(1)

INTEGRATION OF CERCLA AND RCRA FOR THE ERDF
The ERDF project is authorized for operation under CERCLA. The project was designated as the DOE pilot project to integrate NEPA values into the CERCLA regulatory process. Authorization of a waste disposal facility under CERCLA, and integration of NEPA values into the CERCLA process, set this project apart from others at Hanford.(1)
The first regulatory pathway considered for the ERDF, in late 1992, was to include the facility as a RCRA land disposal unit in the overall Hanford site-wide RCRA permit, which at the time was undergoing review by the Washington State Department of Ecology. However, there was almost immediate concern among the three parties that this option was unsatisfactory, because in addition to the lengthy RCRA permitting

process, that by itself could take several years, most wastes being placed into the ERDF would likely be subject to the land disposal restrictions (LDRs), and having to treat all of them prior to disposal would be cost prohibitive.

Shortly thereafter, in May 1993, the regulators informed DOE that their preferred regulatory approach for ERDF was going to be application of the newly-issued (April 1993) CAMU rule. The CAMU rule amounted to a more streamlined approach to RCRA landfill design and construction, due to some regulatory relief from the LDRs and minimum technological requirements (MTRs). At this point, the DOE believed that the CAMU Rule was a better alternative than the usual RCRA permitting scheme because at least the CAMU Rule offered the DOE some regulatory flexibility. The terms of NEPA and its state counterpart, the Washington State Environmental Policy Act, were to be addressed in tandem with those of the CAMU. CERCLA was to be addressed, but only to the extent necessary to fulfill the requirements of NEPA. Thus, at that time, the ERDF regulatory package was to consist of a CERCLA Proposed Plan to cover NEPA, a CAMU permit application, and a draft NEPA Environmental Impact Statement.

This CAMU strategy was developed to take advantage of the teaming arrangement between the Washington State Department of Ecology and the EPA. Under the arrangement, Ecology would regulate ERDF as a CAMU under the Hanford Site-Wide RCRA permit, and under the Washington State Environmental Protection Act (SEPA), via the State Environmental Impact Statement (SEIS) process. EPA would provide oversight and issue a Record of Decision (ROD).(1)

However, by the spring of 1994, it became clear that even with the greater flexibility of the CAMU strategy, the approach could not overcome the time constraints associated with the ERDF schedule. Since the CAMU process would still require the DOE to submit extensive geotechnical data and design plans for the facility, similar to following the full RCRA process, it would likely take several years before the ERDF could be approved. There was a very real prospect of missing ERDF's TPA milestone.With Hanford's stakeholders voicing concerns about the lengthy process and potential for a missed milestone, the three parties quickly came to recognize that the solution for getting the ERDF constructed lay in CERCLA's advantages in terms of speed of implementation over the RCRA/CAMU process. With the speedier CERCLA process allowing for the technical requirements of both RCRA and CAMU to be integrated as ARARs, while also integrating NEPA values into CERCLA, the state and the EPA decided that they wanted the ERDF to be designated as a CERCLA facility.

The final CERCLA-only ERDF regulatory package was submitted in September 1994. It included the RI/FS, a Proposed Plan, a NEPA roadmap describing where in the regulatory package the NEPA values were captured, and the ROD, which was issued on January 20, 1995. The public involvement schedule, beginning with the initial siting meetings and finishing with the approval of the ROD, encompassed less then 15 months. This compares to a schedule that normally takes up to three years to complete.

LESSONS LEARNED

The ERDF project is an excellent example of reinventing government, or what the DOE-EM program refers to as "thinking out of the box". From the very start of the project, the major focus was to depart from the conventional thinking that surrounds DOE projects. Changes to regulatory programs and their standards was not considered off-limits. Moreover, when accounting for the ERDF's tight project schedule and impending TPA

milestone, new ideas for expediting ERDF construction were encouraged. With the help and cooperation of the EPA and the State Department of Ecology, as well as Hanford stakeholders, normal regulatory processes were streamlined. By integrating the technical requirements of RCRA and NEPA with the regulatory flexibility of CERCLA, the DOE was able to design and construct the most environmentally sound and cost-effective facility for Hanford's remediation waste. In hindsight, the decision to depart from conventional project management thinking has proven to be a very wise one, especially in light of shrinking EM program budgets. Without this new approach to integrate CERCLA and RCRA requirements, the ERDF would not exist.

At the present time, construction of the ERDF facility is well advanced. It is likely, weather-permitting, that the facility will be completed several months early, and begin receiving waste in July 1996. In any case, the ERDF will be ready for operations on September 30, 1996, satisfying its Tri-Party Agreement (TPA) milestone.

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55-4

CAMU EQUALS FASTER, BETTER, CHEAPER REMEDIATION AT THE FERNALD ENVIRONMENTAL MANAGEMENT PROJECT

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ABSTRACT

A 1,050 acre Corrective Action Management Unit (CAMU) was approved for the Fernald Environmental Management Project (FEMP) by the U.S. Environmental Protection Agency (USEPA) to manage environmental media remediation waste in the Operable Unit 5 Record of Decision, 1995. Debris is also proposed for management as remediation waste under the CAMU Rule in the Operable Unit 3 Remedial Investigation/Feasibility Study (RI/FS) Report, as of December 1995. Application of the CAMU Rule at the FEMP

will allow consolidation of low-level mixed waste and hazardous waste that presents minimal threat from these two operable units in an on-property engineered disposal facility without triggering land disposal restrictions (LDRs). The waste acceptance criteria for the on-property disposal facility are based on a combination of site-specific risk-based concentration standards, as opposed to non-site-specific requirements imposed by regulatory classifications.

The designation of the CAMU was proposed because the Department of Energy (DOE) will manage low-level radioactive waste, hazardous substances, hazardous wastes and/or mixed wastes as remediation wastes pursuant to Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) response actions at this former uranium processing facility. Certain regulations promulgated under the Resource Conservation and Recovery Act (RCRA) were evaluated as applicable or relevant and appropriate requirements (ARARs) for remediation of the FEMP, including the CAMU Rule. Therefore, the CAMU will add a measure of flexibility in order to expedite and improve FEMP remedial actions. Compliance with these ARARs would have increased the cost and time of the remedial projects without providing any additional protective measures.

Specific aspects of remediation will be expedited under the CAMU at the FEMP, based on negotiations with the Ohio Environmental Protection Agency (OEPA) and the USEPA, Region V. Environmental media and other remediation waste that may contain listed hazardous wastes may be managed in the on-property disposal facility that are below the site-specific waste acceptance criteria (WAC). The OEPA, in supporting this concept, has expressed a desire to limit placement of characteristic waste in the on-site disposal facility. Listed waste from any area will not invoke treatment standards because the regulatory status of the waste will change from "listed" to "remediation" waste. For example, if an area of concern is identified through field monitoring instrumentation, then treatment will only be required if the media affected by a source of released contaminants exhibits toxicity characteristic concentrations.

Temporary units (TUs) and existing facilities will be designated under the CAMU in remedial action work plans when needed to facilitate remediation. These standards will allow for more flexibility in using the minimum technology requirements (MTRs) so that existing structures may be used to facilitate remediation.

WHY USE A CAMU?

Historically, joint CERCLA-RCRA guidance dictated that hazardous waste could not be treated or moved out of the designated area of contiguous contamination (AOC) without triggering LDRs or MTRs. The Corrective Action Management Unit (CAMU) Final Rule (58 FR 8658, Vol. 58, No. 29), promulgated on February 16, 1993, provides facilities undergoing RCRA corrective action with greater flexibility to move, treat, and dispose of wastes on site without triggering LDRs or MTRs, thereby encouraging application of innovative technologies and more protective remedies.

If on-property disposal is selected as part of the preferred alternative for a CERCLA site, there are three possible options for on-site management, treatment, and disposal:

1) comply with LDRs and possibly request any combination of the following: a no migration petition, a treatability variance, a treatment and storage facility variance, or a delisting petition; or

2) application of the "CAMU Rule"; or

3) request an ARARs waiver under CERCLA.

Management and treatment of low-level mixed waste (LLMW) at the FEMP was proposed using the "CAMU Rule" because the soil and debris containing hazardous waste are remediation wastes. In addition, the other options cited above under option 1 would prove to be more costly to meet treatment requirements and time-consuming to receive approval for variances, waivers, or petitions that do not improve the degree of protection to human health or the environment.

The "CAMU Rule" can be invoked only if the waste to be managed is a remediation waste (i.e., not part of an "as generated" process). A "remediation waste" is defined in 40 CFR 260.10 [58 FR 8683] as:

"all solid and hazardous wastes, and all media (including groundwater, surface water, soils, and sediments) and debris, which contain listed hazardous wastes or which themselves exhibit a hazardous waste characteristic, that are managed for the purpose of implementing corrective action requirements under 264.101 and RCRA section 3008(h). For a given facility, remediation wastes may originate only from within the facility boundary, but may include waste managed in implementing RCRA section 3004(v) or 3008(h) for releases beyond the facility boundary."

Disposal of hazardous waste constituents during Superfund actions requires compliance with several potential ARARs under RCRA. Once waste is picked up under the CERCLA area of contamination (AOC), or from the RCRA unit, the requirements for waste disposal are triggered. The LDR treatment standards often cause increased cost and time for remediation. For this reason, many facility owners have historically opted to cap contaminated soil in place and avoid triggering waste placement standards. With the promulgation of the CAMU Rule remediation waste can be managed based on protective site-specific standards and at a lower cost to the remediation project.

Compliance with the LDRs presents the most stringent potential waste placement RCRA ARARs. LDRs can be triggered as applicable requirements by "placement" of restricted RCRA hazardous wastes in land-based units. Land-based units include landfills, surface impoundments, waste piles, and land treatment facilities.

CAMU RULE CRITERIA

EPA promulgated the "CAMU Rule" under RCRA to promote the most efficient and cost-effective remediation possible. In the absence of the CAMU Rule, LDRs are triggered when "placement" occurs, as described above. In promulgating the "CAMU Rule", the EPA provided a separate regulatory framework to manage remediation waste, judiciously expedite cleanups, and reduce costs. In this respect, CAMUs can only be used for management of remediation waste, not for "as generated" hazardous wastes from ongoing production processes or other industrial activities.

The CAMU designation criteria are related to the practical necessities of managing remediation wastes on site during cleanup, rather than to the areal extent and the contiguousness of the contamination prior to cleanup. A CAMU can be designated to include the entire facility, but cannot be extended beyond the facility property boundary even if the contamination release has migrated beyond the facility boundary. However, remediation wastes, especially environmental media, can be managed within the CAMU even if they are associated with a release that has migrated beyond the facility boundary.

The "CAMU Rule" also created Temporary Unit (TU) provisions [40 CFR 264.553, 58 FR 8684] that can be applied to treatment or storage of remediation wastes during remedial activities. TUs can be located inside

or outside the physical boundaries of a CAMU; however, like CAMUs they must be located at the facility. The EPA Regional Administrator determines the requirements for siting, operating, monitoring and closing a TU. Like CAMUs, TUs are also not subject to LDRs and MTRs. There is a one-year time limit on the use of the TU which can only be extended an additional year if the wastes have to remain in the unit due to "unforeseen, temporary, and uncontrollable" circumstances.

According to 40 CFR 264.552(c), seven criteria are to be considered to designate and approve CAMUs for purposes of managing remediation waste:

1) Facilitate the implementation of reliable, effective, protective, and cost-effective remedies.

2) Waste management activities will not create unacceptable risks to humans or to the environment resulting from exposure to hazardous wastes or hazardous constituents.

3) Include uncontaminated areas of the facility, only if including such areas for the purpose of managing remediation waste is more protective than management of such wastes at contaminated areas of the facility.

4) Wastes that remain in place after "closure" of the CAMU shall be managed and contained so as to minimize future releases, to the extent practicable.

5) Expedite the timing of remedial activity implementation when appropriate and practicable.

6) Use treatment technologies (including innovative technologies) to enhance the long-term effectiveness of remedial actions by reducing the toxicity, mobility, or volume of wastes that will remain in place after "closure" of the CAMU.

7) To the extent practicable, minimize the land area of the facility upon which wastes will remain in place after "closure" of the CAMU.

FEMP SITE BACKGROUND

The Fernald Environmental Management Project (FEMP) is a facility owned by the U. S. Department of Energy (DOE). The Fernald facility occupies approximately 1050 acres in a rural area approximately 18 miles northwest of downtown Cincinnati, Ohio. The facility was operated for production of purified uranium metal from 1952 until 1989, when operations were suspended. In July 1986, a Federal Facility Compliance Agreement was jointly signed by the U.S. Environmental Protection Agency (EPA) and the DOE to perform site characterization. In April 1990, the EPA and DOE entered a Consent Agreement for cleanup of Fernald as a Superfund site under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA). This agreement has been revised several times, and is now referred to as the Amended Consent Agreement (ACA). In 1988, a Consent Decree was jointly signed by the Ohio Environmental Protection Agency (OEPA) and the DOE, under Clean Water Act and RCRA authorities, that provides for the management of water pollution and hazardous wastes, including closure of hazardous waste management units (HWMUs). This Consent Decree was amended in January 1993, and together they are collectively referred to as the Stipulated Amendments to the Consent Decree.

Several RCRA-regulated hazardous wastes were generated during the production of uranium. Since the shutdown of production operations at the facility, several HWMUs have been identified. Knowledge of releases from the HWMUs will necessitate compliance with RCRA during the remediation of building debris, and soil and groundwater impacted by these releases. All environmental media (soil, groundwater, and sediment) which contain

hazardous waste constituents are anticipated to also be low-level mixed waste (LLMW) due to pervasive low-level radioactive contamination at the FEMP. These remediation wastes will be managed under the provisions set forth by DOE, EPA, and OEPA to designate the CAMU at the FEMP.

The ACA divided the site into the following five operable units (OUs) based upon their location or the potential for similar response actions:

- OU1 Waste Pit Area Waste Pits 1 - 6, Burn Pit, Clearwell
- OU2 Other Waste Units Solid Waste Landfill, Southfield Disposal Areas, Flyash Piles, Lime Sludge Ponds
- OU3 Former Production Area production area and production-associated facilities and equipment
- OU4 Silos 1 - 4
- OU5 Environmental Media soil, groundwater, surface water and sediments, flora and fauna

In accordance with their signed Records of Decision (RODs), OUs 1 and 2 will ship their RCRA-regulated remediation wastes off-site, which will require complying with the RCRA LDRs for acceptance at the off-site disposal facility. In contrast, OU5 will to dispose of material containing hazardous waste constituents in an on-property engineered waste disposal facility, and OU3, in its RI/FS Report, is considering on-site disposal for its material.

IMPLEMENTATION OF THE CAMU AT THE FEMP

The boundaries of the CAMU are designated in the OU5 ROD to coincide with the 1,050 acres of land within the FEMP property boundaries, such that remediation waste from the entire site (CERCLA's definition of "site") can be managed within the CAMU. The CAMU is also designated to include the on-property disposal facility, provided those wastes meet site-specific waste acceptance criteria that are protective of human health and the environment. A map of the Fernald site (Fig. 1) shows the area of excavation, which was determined by the lateral extent of uranium that exceeds the site-specific, risk-based, cleanup level. The cleanup level is based on an 1 x 10-5 incremental lifetime cancer risk for an undeveloped park, as described in the OU5 Proposed Plan and ROD. The Production Area on Fig. 1 represents the anticipated areas containing potential hazardous wastes. The on-property disposal facility will also function as part of the CAMU. Existing structures to be closed during remediation, and TUs under the CAMU Rule, as needed for on-property disposal, will be designated in the appropriate remedial action work plans.

Fig. 1

The seven criteria described above will be met through the selected remedies described in the OU5 and OU3 RODs, respectively. Each criterion above is referenced below in parentheses where each one is addressed. The on-property disposal facility will serve as a reliable method of containment, which will be designed to be effective for 1,000 years [40 CFR Part 192] (Criteria 1 and 4). In addition, the on-property disposal facility will minimize the land area for wastes that remain on-property (Criterion 7). The use of concentration-based WAC for on-property disposal of LLMW constituents will ensure protection to the sole-source aquifer beneath the site, which are determined through site-specific extensive remedial investigations, risk-based calculations and modelling (Criterion 2). Therefore, on-property disposal will be more cost-effective than shipping large volumes of LLMW off-site for disposal (Criterion 1). In addition, the statutory preference for treatment

(Criterion 6) will be met through treatment of those excavated volumes that are statistically indicated to be characteristically hazardous.
The CAMU will expedite remediation by minimizing segregation, analytical testing, and handling time that otherwise would be needed to meet the specific LDR treatment requirements for individual hazardous wastes (Criterion 1).
The use of existing interim status HWMUs and TUs under the "CAMU Rule" during remediation will also expedite remediation because the need for construction of new storage or treatment facilities will be minimized. In addition, the use of these existing facilities and TUs will not cause any additional impact on the environment where soil and groundwater contamination already exist. Application of MTRs, intended to prevent contamination of soil and groundwater, would not be logical for existing facilities/units since the very situation which they are intended to prevent already exists at the site. If TUs are used for the Fernald site, initial analysis within the OU5 Feasibility Study (FS) indicates they might be needed for the duration for soil remediation.
Remediation costs will be considerably reduced by the application of the "CAMU Rule" at the FEMP. OU5 contains an estimated minimum volume of 28,000 cubic yards of soil containing RCRA-regulated constituents. Most of this soil contains constituents that may be from listed wastes, but which are not anticipated to exceed the WAC for the on-property engineered disposal facility. Only small volumes of soil may be statistically representative of characteristically hazardous waste.
OU3 (the former production facilities) activities currently involve decontamination and dismantlement of the structures under a Record of Decision for Interim Remedial Action (IROD); up to 10% of the material removed under the IROD can be disposed at an off-site location. It is anticipated that this will typically consist of radioactively-contaminated building materials which will be handled as low-level waste (LLW). Other wastes are currently being managed in accordance with approved removal actions. Final disposition of the material removed in building dismantling will be addressed in a combined Remedial Investigation/ Feasibility Study (RI/FS) Report and Proposed Plan currently in preparation, leading to a final ROD. The OU3 RI/FS is analyzing three alternatives: 1) no further action (indefinite storage); 2) disposal in an on-site engineered disposal facility; and 3) off-site disposal.
Although some of the resultant OU3 remediation waste material will likely be classified as mixed waste, the implementation of the "CAMU Rule" will impact the level of treatment that will be required for disposal in an on-site disposal facility. The constituents in the material are not expected to exceed the WAC for the on-site disposal facility. If they do, however, the remediation waste material will either have to be treated to meet the on-site WAC, or be treated in accordance with LDR requirements and be disposed of off-site at a permitted/licensed mixed waste disposal facility at a significantly greater cost.
IS THE CAMU FOR YOU?
EPA's intent in promulgating the CAMU Rule was to allow sensible cleanup solutions for existing contamination problems while attaining the statutory standard to protect human health and the environment. Therefore, any site in the process of developing a cleanup strategy for existing contamination should consider using the CAMU Rule as a tool for implementing a potentially more cost-effective remedy. Attributes of a

site that might influence a decision to designate a CAMU include the presence of contaminants at a site that would be regulated under RCRA and/or CERCLA, and where corrective action or remedial action is indicated. In addition, the use of a CAMU would be most appropriate for sites that plan to treat waste on-site so that staging areas, treatment units and existing facilities could be designated for remediation purposes, and especially if the remediation waste may be managed permanently in an on-property disposal facility.

SUMMARY

The use of the CAMU at Fernald is approved by the regulatory agencies via OEPA concurrence and EPA's signature of the OU5 ROD (January 1996). A similar approach is anticipated for OU3's ROD (projected for EPA signature in late 1996). The CAMU Rule is the most appropriate method for compliance during remediation of soil and debris at the FEMP because the substantive requirements under Subtitle C will be met for providing long-term, cost-effective, practical and protective remediation. In this respect, application of the CAMU should be considered at other sites undergoing environmental restoration, regardless of whether it is being conducted as a CERCLA response action (removal action or remedial action) or RCRA corrective action.

Session 56 -- ROBOTICS AND REMOTE TECHNOLOGY

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HOUDINI: RECONFIGURABLE IN-TANK ROBOT

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ABSTRACT

Across the DOE complex, hundreds of above and below ground storage tanks contain large amounts of high-level waste. The Fernald Environmental Management Project, a site within the DOE complex, contains four, above-ground, domed, concrete waste tanks. Some structural deterioration has been noted in the walls and domes of the tanks, resulting in concerns about leakage of their contents. Waste retrieval for final remediation must begin in 1997.

RedZone Robotics, Inc. and Carnegie Mellon University (CMU) are developing a tethered mobile robot, Houdini, to work inside waste storage tanks in support of the Department of Energy's Environmental Restoration and Waste Management (EM) Program. The development of Houdini is funded by the DOE's Office of Science and Technology through the Morgantown Energy Technology Center. Houdini will first be deployed at Fernald in support of the waste retrieval for final remediation of the K-65 waste silos. Fernald personnel are active members of the development team.

Houdini is a tethered, hydraulically powered, track driven, teleoperated, work machine with an expandable frame chassis that allows it to fit through confined entries as small as 0.57 meters (22.5 in.) in diameter.

The Houdini system performs heel removal, waste retrieval, waste mobilization, waste size reduction, and other tank waste retrieval and decommissioning tasks. It is equipped with an articulated plowing blade and a manipulator with exchangeable end effector tooling. Tooling consists of a gripper, scoop, shear, vacuum suction hose grip, and spray-down nozzle. The system can locomote over and through a variety of waste forms, and can operate fully submerged. An operations trailer houses Houdini's primary operator interface console. Joysticks, switches, and remote-viewing monitors allow the operator to control the system from a remote location. A second, portable control station provides the capability for local operations, on-site debugging, system checkout, and emergency recovery. A navigation system developed at Oak Ridge National Labs (ORNL) provides the operator with feedback on the vehicle's position and orientation inside the tank.

Robotic systems are needed to work inside waste storage tanks in support of the DOE's EM activities. As a technology for supporting the DOE's EM program and in comparison or collaboration with other competing technologies, Houdini provides many benefits. The Houdini system is designed to eliminate or reduce potential public and operational health risks associated with work on DOE tanks. The system provides fully-contained remote operation, reducing the risk of spreading contamination outside of the tanks. Because of Houdini's similarity to bulldozers and backhoes from the construction industry, it provides simple, intuitive, and efficient waste handling techniques. Houdini's transportation, installation, deployment, and removal operations are simple due to its compact size. Houdini's simplicity and operational capability lead to cost efficiency with respect to development, operation, and maintenance. The Houdini concept was developed specifically in response to the needs at Fernald, but is useful at a number of other DOE sites where tank remediation projects are planned. Upon successful completion of the development program, in late 1996, RedZone will make the results of this program commercially available.

INTRODUCTION/NEEDS

RedZone Robotics, Inc. and Carnegie Mellon University (CMU) are developing a tethered mobile robot, Houdini, to work inside waste storage tanks in support of the Department of Energy's Environmental Restoration and Waste Management (EM) Program.a This project is funded by the DOE's Environmental Management Office of Technology Development through the Morgantown Energy Technology Center (METC). Our goal is to develop technology that is useful for in-tank operations throughout the DOE's EM program. The first application of the Houdini system is to support the waste retrieval action planned for the final remediation of the Fernald site's waste silos. RedZone and CMU have discussed potential applications for the system with personnel from several other DOE sites, and have found that the system would be widely useful in the DOE complex for tasks both inside and outside of waste storage tanks. We are tailoring the first implementation of the Houdini system to the specific needs of the Fernald silo remediation. The Fernald application-specific design constraints are primarily interface issues and should not interfere with the utility of the system at other sites.

In addition, DOE personnel at the Oak Ridge National Laboratories (ORNL) have expressed a strong interest in the Houdini system. They have a target application scheduled for mid-1996. This program represents a unique opportunity to develop a new technology that has immediate

application in two CERCLA cleanup actions; the proposed applications at Fernald and ORNL support Federal Facility compliance agreements.

OBJECTIVES/PROBLEMS

The primary application for the Houdini system is to support the final remediation of Silos 1, 2, & 3 at the Fernald Site. Houdini will perform essential missions in support of this final remediation effort that will retrieve waste from the tanks and vitrify it for long term storage.

The CRU4 area at the Fernald site contains 4 above-ground, concrete waste silos. All four domed waste silos are 24.4 meters (80 feet) in diameter, 11 meters (36 feet) high at the center of the dome, with 8.3 meter (27 foot) high vertical walls. Four 0.51 meter (20 inch) diameter manways are evenly distributed around each tank dome at 15 feet from the side walls, at a slant of 17 degrees from horizontal. A fifth 20 inch manway near the center of the dome will be eliminated when a 6 foot opening is created to support the remediation activity.

Some structural deterioration has been noted in the walls and domes of the silos, resulting in load restrictions. No detectable load is allowed on the 6.1 meter (20 foot) diameter center section of the dome. On other dome areas, the maximum live load is limited to 700 pounds including personnel and gear.

Waste material in Silos 1 and 2 is described as K-65 material and has the consistency of toothpaste. The waste is covered with a 0.3 meter (12 inch) thick layer of Bentonite clay to reduce radon emissions from the waste. Material in Silo 3 is a light, dry metal oxide powder similar in consistency to talcum powder. The Silo 3 waste may be compacted near the bottom of the silo. In addition, each silo contains pipes, wrenches, sample bottles, gloves, and other debris that has fallen into the tanks over the years.

The fourth silo is identical to the other three, but was never used for waste storage. It will be used as an uncontaminated mock-up facility to fully test all procedures prior to remediation of other silos. Silo 4 may be partially filled with a surrogate waste material to support these tests.

To enable the waste retrieval operations in light of the dome load restrictions, a superstructure has been constructed over Silo 4. An equipment room situated over the center of the dome is supported by the superstructure. Seals will be installed between the equipment room and the silo dome and a six foot diameter opening will be made in the silo dome directly under the equipment room. Doors in the equipment room floor will provide access to the tank. Controlled entry points, rails, winches, equipment carts, and other mechanisms in the equipment room will support the deployment and retrieval of equipment into the silo.

The primary retrieval method for Silo 1 and 2 waste will be hydraulic removal. A sluicing pump will be lowered into the tank from the equipment room. Water will be added to the waste material and the liquefied waste will be pumped out of the tank. In Silo 3, pneumatic conveyance will be used to retrieve the waste material. The methods will remove the bulk of the waste materials from the tanks, leaving only debris and a waste heel to be removed by other means.

APPROACH/SOLUTION

The Houdini system will be used for heel and debris removal from the tanks, during and after the bulk material removal by sluicing and pneumatic conveyance. In Silo 4, Houdini's capabilities will be fully tested in an uncontaminated environment. The silo will be partially

filled with surrogate waste material and debris that approximates the waste properties in Silos 1 and 2. During bulk material removal, Houdini will be deployed to remove debris material that interferes with the sluicing operation. Using a shearing tool and gripper, Houdini would be deployed to gather debris, size reduce the debris as necessary, and load the debris into a tram bucket for retrieval from the tank. After the bulk waste removal action is complete, Houdini will be deployed to perform waste heel removal and debris collection. To perform the heel removal, a small sluicing pump will be lowered into the tank. Houdini will use its gripper to deploy a water spray nozzle. Water spray will be used to mobilize the heel material and wash it toward the sluicing pump for removal. The Houdini plow blade will be equipped with squeegees on the sides and bottom and will be used to push slurried waste material toward the sluicing pump. The edge of the plow blade can be used to push waste material away from the side wall of the tank to clean the edges of the tank floor. Houdini can also be used to wash down or apply a spray coating to the tank side walls for decontamination.

In Silos 1 and 2, Houdini will be used to support waste retrieval for final remediation in the same modes as described above for the Silo 4 functional test. The radioactive waste in Silos 1 and 2 will be primarily retrieved by sluicing, after which, the waste will be vitrified for long term storage.

In Silo 3, Houdini will support waste retrieval for final remediation in conjunction with the bulk material removal by pneumatic conveyance. Specifically, Houdini will perform debris collection, size reduction, and removal during and after bulk pneumatic conveyance, and perform heel removal by deploying a pneumatic vacuum hose. In addition, Houdini could be used to plow waste material to a central point for pneumatic retrieval. Houdini could also deploy tools to assist in removing compacted waste material from the tank floor.

PROJECT DESCRIPTION/TECHNOLOGY

System Overview

The Houdini system consists of five main components and their subsystems; the vehicle, deployment system, PDCU, control consoles, and tooling.

Vehicle

The vehicle is a hydraulically-powered, track-driven, folding frame machine similar to a small bulldozer. The vehicle can fold to fit through a 0.57 meter (22.5 inch) diameter opening for deployment, and is equipped with a plow blade and a manipulator arm. The plow blade also folds for deployment and can be height-adjusted for plowing various materials at various rates. The manipulator is a Schilling Titan class hydraulic dexterous manipulator, which can deploy a variety of tooling for performing work inside a tank. The vehicle tether is attached to the rear of the folding frame assembly. The tether termination will support the full weight of the vehicle and tooling to enable deployment and retrieval. Two camera and light assemblies provide visual feedback for remote operation. One camera and light unit is mounted on the forearm of the manipulator. The camera is aimed by orienting the manipulator. The second camera unit includes a pan and tilt unit and is mounted on a mast at the manipulator shoulder. A microphone will provide audio feedback to the operator. Navigation system sensors will be installed on the vehicle as part of the navigation system interface.

Deployment System

The deployment system is designed to interface with the superstructure and equipment room above the Fernald waste silos. It is an important part of the Houdini system. First, it provides a convenient way to remotely manage and store the 150 feet of tether that is the lifeline of the Houdini vehicle. Second, it provides the lifting force that is needed to lower and raise the vehicle into and from the tank. Third, it provides a "docking area" where the vehicle can be secured during storage or transport, and lastly, it provides an area where spare parts or tools can be stored when not in use.

Tether Reel

The tether reel is a "spool" looking device that is 48 inches in diameter and 30 inches wide. A flange at each end contains the tether on the drum. Payout of the tether is controlled by a mechanical level-wind system that ensure that the tether does not cross over on itself and tangle. The tether reel is driven by a hydraulic motor with a "power-off" brake in case of hydraulic power loss. The hydraulic motor is sized to allow a pull force of 2000 lb tangent to the reel surface. A payout sensor is used to monitor the length of tether that has been reeled out. The sensor shall also indicate ends of travel (i.e., when the tether is completely in or out). A means of manual retrieval is necessary in case of hydraulic power failure. This is accomplished with an external, battery powered, hydraulic supply and manually operated valve.

Tether

The tether is used to lower and raise the vehicle in the tank and provides control signals and electric and hydraulic power to the vehicle. The tether will be a custom fabrication that includes:

- Strain-relief termination
- Hydraulic supply and return lines
- Shielded-twisted pairs for control and feedback signals
- Mini-coax lines for camera signals
- Shielded-twisted pairs for navigation system signals Conductors for power to onboard valving
- Kevlar braid for structural support (to carry the weight of the vehicle)
- Abrasion resistant coating

Power Distribution and Control Unit

An environmentally sealed and temperature controlled power distribution and control unit (PDCU) will be installed on the superstructure, outside the equipment room. The PDCU includes the electric transformers and distribution/conditioning equipment, the control system, and tether and control system interface connectors. A separate enclosure will house the electrically-powered hydraulic power supply.

Control Console

The operator console provides the operator interface to the Houdini system. The console includes joysticks, switches, a master manipulator, and video monitors for controlling system functions and monitoring system operation.

Navigation System Interface

An interface will be provided to the Position and Orientation Tracking System (POTS) which has been developed at ORNL. ORNL has agreed to make POTS available to this program as Government-Furnished Equipment. POTS will provide accurate feedback on the vehicle's position and orientation inside a tank to enable more efficient and robust controls.

Control Trailer

A control trailer will house the operator control console. The trailer will be a simple mobile, industrial trailer that provides heated and air conditioned real estate for the operator. Tie downs will be provided for securing the trailer against high wind. The trailer will be structurally capable of being moved with the control console inside. The trailer will require 110 volt site power.

Suitcase Controller

A hard-wired suitcase controller will be available to perform system checkout, local operations and provide for emergency operations in the case of console/control computer or telemetry failures between the control center and the deployment system. Switches, buttons, and a single remote viewing monitor will provide for simple operations from the suitcase controller.

Tooling

Specialized tooling will be provided to enable the use of the Houdini system in support of the Fernald waste retrieval plans. The bottom and sides of the plow will be equipped with squeegee blades to provide efficient mobilization of the waste slurry on the floor of the tank. A gripper, shear, and scoop will be provided for deployment from the manipulator. For sluicing operations, Fernald will provide a small pump for heel removal operations; RedZone will equip the pump with a water line, hose reel, and spray nozzle that can be deployed by the Houdini manipulator to spray-wash waste material toward the sluicing pump. For vacuum retrieval operations, Fernald has agreed to provide a hose grip that will attach to the manipulator and enable the deployment of a pneumatic vacuum hose.

APPLICATION/BENEFITS

As a technology for supporting the DOE's EM program and in comparison or collaboration with other competing technologies, Houdini provides many benefits. The Houdini system is designed to eliminate or reduce potential public and operational health risks associated with work on DOE tanks. The system provides fully-contained remote operation, reducing the risk of spreading contamination outside of the tanks. Because of Houdini's similarity to bulldozers and backhoes from the construction industry, it provides simple, intuitive, and efficient waste handling techniques. Houdini's transportation, installation, deployment, and removal operations are simple due to its compact size. Houdini's simplicity and operational capability lead to cost efficiency with respect to development, operation, and maintenance.

Evaluating the merits of the Houdini system for these applications requires comparing it to competing technologies. In comparison to mobile robot systems that are currently available, Houdini's folding frame technology provides a substantially larger work platform which can fit through existing tank openings. As a larger platform, Houdini is more powerful, more efficient, and more capable than other, smaller mobile systems.

Several non-robotic retrieval methods are being considered for use in DOE tanks. These technologies, such as sluicing, pumping, and pneumatic conveyance, are appropriate or preferred technologies for some of the tanks in the complex. As it will at Fernald, Houdini could assist in the application of these retrieval and conveyance methods. In addition, the current design could be applied for use in Oak Ridge's north and south tank farms.

Depending on specific work tasks and application sites, Houdini can be deployed to either complement or replace a long-reach manipulator (LRM) system. Used in conjunction with LRMs, Houdini provides additional or enhanced capabilities inside a tank. In tasks where Houdini is useful instead of LRMs, Houdini will be simpler and less expensive to deploy, operate, retrieve, and decontaminate than LRMs.

FUTURE ACTIVITIES/APPLICATIONS

In addition to the primary application at Fernald, other applications for Houdini have been identified in support of tank waste retrieval operations in the DOE and private sector. Also, several tasks outside of tanks have been identified for which Houdini would be useful.

Oak Ridge North and South Tank Farms

The north and south tank farms at Oak Ridge National Laboratory have a total of 16 domed, cylindrical, single-shelled, underground storage tanks made of Gunite (similar to concrete), ranging in diameter from 20 to 50 feet and equipped with 24 inch diameter manway penetrations. These tanks were used to store laboratory waste and are expected to contain a wide variety of materials, with estimated radiation levels of 1 to 100 R. During 1983-1984 the tanks were emptied through a sluicing method, leaving a heel of up to several feet thick at the bottom. The heel waste must be removed to prevent the migration of waste material out of the tanks. ORNL is under a Federal Facility Compliance Agreement to complete a CERCLA treatability study on the Gunite and associated tanks. The baseline plan for this study includes the evaluation of both vehicle- and arm-based retrieval systems. The current plan at ORNL is to evaluate the Houdini vehicle during this study, and if successful, Houdini might be selected for the final remediation retrieval action.

Other DOE Tank Applications

The Houdini system is useful in a variety of other DOE tank waste retrieval operations. Houdini could be deployed in a tank prior to the major removal action to collect additional information about the waste content and tank interior.

In support of other in-tank work systems, such as long reach manipulators, Houdini could be used to deploy cameras, lights, and sensor systems. The mobile deployment of such monitoring equipment will provide viewing and data gathering flexibility that cannot be achieved by mounting such equipment on fixed masts or on a long reach manipulator.

The long reach manipulator (LRM) systems that are being developed for tank waste retrieval will require a variety of tools to accomplish their tasks. The Houdini crawler could serve as a mobile tool carrier for the LRM, carrying several tools and making them available at the most appropriate location inside the tank.

In support of final tank decontamination and decommissioning, Houdini could deploy tools to scarify internal tank surfaces.

Commercial Tank Applications

Periodic cleaning and inspection of storage tanks in petro-chemical industries are becoming common maintenance procedures. It is likely that these procedures will be required by law in the US in the next few years. We have been in contact with several service providers in the petro-chemical industry, who have expressed an interest in the Houdini system in support of these operations.

Non Tank Applications

Alternate uses currently envisioned for this system include indoor as well as outdoor tasks. In support of buried-waste excavation programs,

Houdini could perform fine excavation and monitoring to assist a larger remote excavator, perform fine excavation to isolate and extract specific objects, and assist removing a drum in one piece. In support of decontamination and dismantling programs, Houdini could be used as a small platform to gain access through tight areas for selective equipment removal and could lend assistance to larger worksystems as a tool-carrier platform, size-reduction system, or waste packaging system. In support of surveillance and monitoring operations, it could perform such functions as monitoring drum storage areas and decommissioned processing areas requiring access to tight corridors. Removal of the tether is possible through the use of a gas-engine or batteries and the interface of a radio telemetry system.

ACKNOWLEDGMENTS

This project is funded through the U.S. Department of Energy's Morgantown Energy Technology Center, with RedZone Robotics, Inc. as the prime contractor, Carnegie Mellon University as a sub-contractor, and Fernald as the primary user of the equipment. For further information, please contact:

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ABSTRACT

This paper describes the design, manufacture, integration, testing, training, and commissioning of 2 Remote Controlled Material Handling System (RCMHS) for the retrieval of approximately 5,600 drums containing solid Thorium waste at DOE's Fernald site. The work was performed by Spar Environmental Systems in partnership with RSI Research under a contract from Fernald Environmental Remediation and Management Company (FERMCO). The RCMHS design and development program presented unique challenges to produce two remotely controlled vehicles capable of inspecting, acquiring, retrieving, and disposing of drum containers containing solid thorium waste. A very aggressive schedule had to be matched with meeting stringent safety requirements and developing systems capable of handling drums ranging in size from 35 to 110 gallons and weighing up to 1300 lb each. The system also had to lift and transport a fully loaded Thorium Overpacking Container (TOC), which is a 6 drum holding pallet and a lid, weighing up to 8,000 lbs when full.

The design concepts were developed at the proposal stage and were adhered to with minimal changes on a fast track program to mitigate risk. The program required that detailed design be completed, materials and parts be manufactured, software written, and the system integrated and tested within 12 weeks from issue of the contract. The RCMHS system was designed around a standard Nissan "Box Car" fork lift truck which was heavily customized for remote control. The vehicle includes a unique Drum and Plywood Handler (DPH) which is a specialized 4 degree of freedom attachment designed to manipulate either drums or plywood sheets, see Fig. 1. In addition the RCMHS is capable of remotely picking up loose material with special purpose clean-up tools.

The two systems have been delivered and commissioned at the Fernald site and are ready for operations inside the drum storage warehouse, Fernald Site Building 69. With 12 site operators fully trained and the system is now being put into service.

Fig. 1

RCMHS GENERAL DESCRIPTION

The Remote Controlled Material Handling System (RCMHS) is a highly modified 8000 lb commercial fork lift truck which provides the capability of handling radioactive thorium waste stored in a warehouse. The drums are stacked tightly together, three tiers high with plywood sheets separating the tiers. To complete the remediation project the RCMHS undertakes the following 5 functional tasks:

1) Normal drum retrieval, through grasping the drums by their chimes;
2) Corroded drum retrieval through grasping and supporting the bottom of the drum or rotating of the drum so that the lid becomes the new base;
3) Plywood separator retrieval, by lifting and clamping the plywood sheets and placing them in a stack;

4) Debris clean up, achieved by the DPH picking up the spillage container and the debris sweeper;
5) Moving Thorium Overpacking Containers and removing / replacing the lid to these containers, achieved with standard forks and sideshifter.

REMOTE CONTROL BASE UNIT

The Remote Control Base Unit (RCBU) consists of the Base Fork Truck, the On Board Remote Controllers (3), the Visual Monitoring System, Hydraulic control modules, hoses, and cables, see Fig. 2 for general overview layout. The Base Fork truck is an 8000 lb capacity, 2 wheel drive, liquid propane gas powered, "Box Car" Fork Lift Truck (Box Car models are approximately 8 inches shorter which gives a tighter turning radius). A three stage mast provides up to 151 inches of lift capability with 6925 lbs at a 24 inch offset from the face of the forks. With 8000 lbs at 24 inches, the mast can be raised to a height of 41 inches in free lift mode; where the carriage rises on the central cylinder but the mast stages remain collapsed. The mast can be tilted from 5 degrees forward and 5 degrees backwards.

Fig. 2

Permanently mounted to the RCBU lift carriage is a side shifter which provides +/- 4 inches of lateral movement and functions as a platform for the mounting of the DPH or Forks. The centre of gravity of the cargo was offset by an additional 3 inches from the front axle of the lift truck as a result of the thickness of the sideshifter being added to the carriage. This added to the weight of the side shifter, approximately 400 lbs, meant that the counter weight at the back of the vehicle had to be supplemented with an additional 600 lbs to ensure the handling characteristics with full load were maintained. The On Board Remote Controller (OBRC) provides local vehicle control for the RCBU hydraulic actuators and camera/ Pan/Tilt Unit (PTU) functions. It also receives and transmits data to the Control console, and provides system health monitoring. The RCBU includes the camera/PTU assemblies and their support frame. The electrical system of the lift truck was modified to include an additional battery for the OBRC to ensure that the voltage levels were maintained during start up. The alternator was replaced with a larger capacity unit to supply the needs for the electronics, lights, camera assemblies, and hydraulic valves. The gear pump that was supplied with the lift truck was replaced with a pressure compensated hydraulic piston pump to reduce the heating effect on the hydraulic fluid.

ON BOARD REMOTE CONTROLLERS (OBRC)

The OBRC controls the RCBU and drum /plywood handler functions through proportional valves, solenoid valves and, pressure and flow regulators. In case of an emergency, the parking brake is automatically activated by the Controllers if the health monitoring system is triggered by a telemetry fault, loss of hydraulic pressure, or electrical power disruption. The fail safe capability of the parking brake is achieved through spring activation with the spring rate being set to provide a an acceptable stopping rate. The OBRC substitutes for the normally human controlled functions on the fork truck such as braking, throttle, gear shifting, parking brake release, mast lift and tilt, and steering. The manually controlled valves were removed and replaced with electrically actuated proportional control valves. The parking brake has a manual release, to allow towing on the RCBU in case of a complete system failure. The main brake is integrated into a single control which is proportionally released as the joystick on the control panel is moved

from the centre position. The brake will not release unless a motion enable trigger (dead man switch on the joystick) is held closed. When mast or DPH functions are desired the RCBU is switched to handling mode. This causes the engine RPMs increase, providing additional hydraulic power for lifting or handling drums. To ensure that the resultant extra engine power is not transferred to the drive train, thus creating an unsafe condition, forward and reverse motion is disabled while in handling mode.

Due consideration of the radiation environment in which the OBRC and RCBU are operating has been made in selecting the electronics and CCD camera systems. In general, commercially available electronic components and CCD devices are capable of sustaining total radiation dosages of the order of 5 kilorad without sustaining damage or degradation in performance. This is roughly an order of magnitude higher than the total dosage expected throughout the operational life of the RCBU, thus avoiding the need for radiation hardened components. The major hazard in the Building 69 is airborne thorium particulate, which presents no hazard to the RCBU components.

Three controllers are utilized on the RCBU. Controller 1 provides the base vehicle functions such as brakes, throttle, gear shifting, mast lift/lower, mast tilt, and sideshift. The second controller operates the Drum and Plywood Handler's solenoid valves. The third controller drives the camera functions (zoom and focus) and pan and tilt motions. The Third unit also handles the communications between the OBRC and the Remote Operations Console controller. The complete Electrical Block diagram for the RCMHS system is shown in Fig. 3.

Fig. 3

DRUM AND PLYWOOD HANDLER (DPH)

The DPH is designed to handle normal drums, corroded drums that could rupture, 4 by 8 foot plywood sheets used to separate the tiers of drums, and with the help of remotely installed tools pickup loose material from the floor. The DPH attachment was based on a "Cascade" 3500 lb rotator/clamp that has been modified to better complete the tasks required of it. The DPH is capable of +/- 168 degrees of rotation. This allows a suspect drum to be carried on its side then placed in its lid on the Thorium Overpacking Container. Hydraulic travel stops limit the rotation so that the internal cable cassette is not over stressed.

The drums are held by two grippers which are mounted on a set of arms. Each arm is driven by a hydraulic cylinder and their motion is synchronized with a hydraulic flow divider. The clamp arms are capable of gripping drums from 20 to 30 inch in diameter,which corresponds to drums of 35 to 110 gallon capacity. Travel stops on the cylinders limit the minimum to maximum gripper openings from 18 to 35 inches respectively. Gripper pressure is maintained on a drum even if the RCBU is shut down by pilot operated check valves. The grippers have 3 pressure settings for closing, low for empty drums (300 psi), medium for loaded 55 gallon drums (650 psi), and high for overpacking drums (800 psi). The load settings can be adjusted to any pressure using the relief valves.

The barrels are held on either side with two large grippers. Each gripper has fifty-five rubber nipples which are pressed onto the barrels to provide both grip and support of the drum chimes. The grippers slide along linear bearings on the support arms to retract the drums over the support plate. The gripper Extend/Retract drive utilizes 1 hydraulic motor on each arm which moves the grippers through a chain/ sprocket

drive. The two grippers are synchronized using a pressure compensating flow valve which keeps their rate within 4 percent of each other. If the gripper plates do get out of synchronization they will equalize at the end of travel hard stops.

The Grippers present a narrow profile when approaching drums, thus allowing the drums to be removed even when they are placed chime to chime. The drums in the stacks are tight to one another, so to remove them, the RCBU approaches the stack at a 60 degree angle to the face. The corner drum is removed first thus producing a wider profile on the adjacent drum. This procedure is repeated along the face of the stack until all the drums are removed. The second row is then removed in a similar manner. When all 8 drums are removed from a plywood separator, the sheet can be taken away to expose the tier below. Limited pushing of the drums can be attempted with the grippers, however, if the drums are heavily corroded this action increases the risk of rupture.

The drums are supported after they are retracted into the DPH with a support plate. Two dual acting hydraulic cylinders provide the support plate with up to 8 inches of vertical travel. The Lift plate is hinged to prevent damage to itself or the tops of drums if the mast is tilted forward during operations.

For tasks of lifting the plywood separator sheets, the support plate is inserted under the edge of the plywood and the RCBU is driven forward so that the plate slides under the sheet. The support plate is raised until it contacts 4 rubber nipples on the under side of the grippers. The rubber securely holds the plywood so that it can be manoeuvred by the RCBU.

The DPH can be removed from the side shifter with quick mechanical, electrical, and Hydraulic connections. The forks can then be installed on the carriage to allow the RCMHS to handle the TOC. The DPH to fork exchange task or the reverse operation can be performed single handedly in less than 5 minutes.

SWEEPER PLATE AND SPILLAGE CONTAINER

The sweeper plate and the spillage container are designed to be remotely installed onto the DPH by driving the support plate into a pocket at the back of the Spillage container. The Sweeper is clamped with the grippers, then lifted out of a tool rack. The sweeper plate can be retracted towards the spillage container using the gripper extend /retract drive. The sweeper preforms two functions, the first is to pull debris into the spillage container and the second is to hold the spillage container closed during transportation to the disposal container. The scoop on the front of the spillage attachment is hinged and will dump automatically due to the force of gravity, if the sweeper is extended away from it.

REMOTE OPERATING STATION (ROC)

The Remote Operating Station is installed into a trailer adjacent to Building 69. The ROC is an operator work station providing two control consoles, the main console is used for RCBU control functions and the other provides control of the cameras and PTUs. In addition, two color televisions provide the operators with the camera views from the RCBU. The operator console features a joystick for RCBU forward, reverse, speed and steering control. A deadman trigger is incorporated in the joystick and if released it will apply the brakes and cut the throttle. The command console features 2 key locks to enable the RCMHS and 10 status warning lights, Fig. 3 shows the warning sensors. Individual switches give the operator control of all the discrete functions in the RCMHS. An

emergency stop button on the console provides the operator with a method of shutting down the system if needed.

A separate console with 2 joysticks is provided for the camera pan and tilt, and zoom and focus controls. This console also houses the supervisor emergency shutdown, for potential hazardous conditions unobserved by the operator.

Hardware safety features are backed up by software interlocks that automatically stop the RCBU if protocols, handshakes, and system parameter set points are violated. The RCBU is stopped by an internal E-Stop which functions similar to the RCMHS Halt.

VISUAL MONITORING SYSTEMS

The VMS is a critical subsystem for the successful operation of the RCMHS under remote teleoperation. The ability to remotely navigate the RCBU within the confines of the building and to safely manoeuvre the RCBU and operate the attachments to inspect, grapple, secure, transport and dispose of drums, separators and loose material, are totally dependent on the visual system. Wide angle lenses give the cameras approximately a 90 degree field and give the operator the ability to see both the vehicle as a reference and work area. To increase the operators efficiency, additional camera views are provided at the work site to give an overall perspective to the operations. Reversing and manoeuvering in confined spaces with grappled payloads is a challenge. The VMS provides the operator with reverse and forward viewing by panning the cameras.

The minimum focus distance required for handling the drums is approximately 5 ft, which is compatible with the minimum focus of the 10:1 zoom ratio lens selected. Each camera has its own quartz halogen flood light which is co-linearly mounted on the Pan and Tilt Unit to provide direct illumination of the work site. The light from the floods is cast over an area which closely matches the cameras widest field.

Since all operations require the operator to discern the position of the RCBU and its tools from the surroundings, visual cues on the mast, DPH, and tools are critical to successful completion of the tasks. The mast height, tilt, and side shift can be seen directly when the left side camera is tilted down. The camera is able to view flat black markers against a bright yellow or orange background. The height of the carriage on the mast is directly shown in one foot increments. For the DPH grippers the end of travel positions for extend/ retract are given with a black stripe.

CLPA MOUNTING ON THE RCBU

The Cameras, Lights and Pan/tilt Assemblies (CLPA) are positioned on the top of the centre stage of the mast. The cameras are positioned approximately 2.5 m above ground level with the carriage operating in the free lift range. When the DPH is raised for third tier drum operations the cameras rise with the centre stage thus maintaining a perspective view. With the two cameras mounted on the same plane, the operator is provided with a perspective to sense depth. The cameras also give a clear sense of the position of the RCBU centre line which is a critical step in alignment for drum handling.

TELEMETRY AND VIDEO COMMUNICATIONS

A hard wired telemetry, audio, and video system provides a total of 500 feet of communications length between the ROC and the RCBU. For operations within Building 69, a 9/32 inch steel aircraft cable is strung the length of the warehouse, approximately 250 feet. The telemetry and video cables are then hung on the aircraft cable with quick connection

hangers spaced every 12 feet for a total length of 240 feet. The remaining cable is used to provide the run between the trailer and Building 69.

To provide the vehicle with lateral movement capability within the building, a cable pulley system is mounted on the right side of the RCBU. This allows up to 30 feet of movement to either side of the aircraft cable. The pulley system also provides some tension to the cable for retrieval behind the RCBU as it moves along the length of Building 69.

REDUNDANT EMERGENCY STOP

The RCMHS is equipped with a Redundant Emergency stop which ensures that even if the primary telemetry link does not respond to the command, a second path will safe the RCBU. The redundant emergency stop operates through a separate set of conductors and when the signal is tripped, a power relay is cut which shuts down the RCBU engine and vehicle controls. When the Emergency Stop is commanded the following occurs:

a) the engine is stopped,
b) the parking brake is activated, by cutting power to the solenoid,
c) the transmission is shifted to neutral,
d) all mast and attachment functions are stopped,
e) the electronics and the video system put in standby.

RCBU MANUAL OPERATIONS

A pendant can be manually attached to the RCBU which allows limited local operations of the base vehicle, mast and sideshifter. This pendant is used to assist the operator in the change out tasks of the DPH to the fork and vice versa or in case of a control system failure.

DESIGN AND MANUFACTURE

Due to the extremely short schedule for the development of the RCMHS it was critical that the design and manufacturing stages be integrated, so that the critical components were fully detailed and in the process of manufacture, while the less critical components were being designed. The design concepts, layouts and detailed drawings were all performed using Autocad.

The Drum and Plywood handler hydraulic hosing and tubing layout had to be undertaken right on the vehicle due to their complexity and short development schedule. The first vehicle became the prototype for the hydraulic arrangement and then this was duplicated onto the second vehicle. A detailed photographic record of the hydraulic layout was kept as a reference for future builds.

The DPH and scoop/sweeper had to be remotely stored and installed with no manual intervention, this meant that each tool required its own custom stand. The stands had to not only support the tools, they also had to provided alignment guides for the visual system. A combination of course visual cue and fine mechanical guides were used for the removal and replacement operations. The visual cues were developed during the initial testing phase of the RCMHS. This process of tailoring the RCMHS system to the actual tasks being performed, optimized the design and reduced the operators time to perform tasks by a factor of 2.

OPERATIONAL PROCEDURES AND TESTING

To reduce the risks that the RCMHS system would be unable to perform the tasks required of it, the operational procedures were developed in conjunction with the layout design and the functional requirements. The control panels functions were defined from the requirements generated in the operational procedures. This parallel design process meant that the

hardware design, system requirements, and the operational procedures were developed and updated co-incidently.

To reduce the risk that the RCMHS System may not function correctly while performing its tasks, a step by step operational analysis was under taken using the Autocad drawings. Each individual function (degree of freedom) for the RCBU was placed on a different layer on the drawing. By selecting individual or groups of layers, specific functions could be checked and procedures verified. As an example, all the mast and DPH layers could be selected and rotated with respect to the front axle of the truck to simulate the tilt function of the mast. This type of action was carried out in discrete steps for all the critical operations for the tasks of picking up drums or the plywood. The drum handling operations were repeated for drums on the top, middle and bottoms tiers in the stacks. The viability of the operating scenarios was validated by this analysis. This analysis of the operating procedures proved to be very useful in the training of the FERMCO site operators. The sketches produced, gave the operators a much clearer visualization of the tasks being undertaken then the written procedures did.

The RCMHS specification called for the vehicle to be able to position a drum within one inch of the desired position. The first time that this test was performed using the complete system, the drum was placed within a half inch. The operator also found that the this was an easy tolerance to meet on repeated tests.

For the drum removal task from the third tier, it was found that the removal procedure could be accomplished within 10 minutes of start. This included the time required to place the drum onto the TOC. Corroded drums were found in the test program to present the operator with no additional challenge as compared to fully intact drums. The large grip area combined with the support plate, resulted in low stresses on the drums so damage was minimized during handling.

Waste pickup using the scoop and sweeper was a task that required the DPH to be very close to the zero degree position. With the scoop level this tool was capable of picking up about 95% of the material on the floor. For the test, the material used was of a fine grain particulate. Since, the thorium waste is congealed, it should be even easier to sweep up than the tests demonstrated.

DELIVERY

The two complete RCMHS systems were delivered to the FERMCO site in September and October of 1995. At the site the vehicles were given a functional test to verify their performance following shipment from Spar. The RCMHS System has exceeded the performance requirements originally set out for it.

FURTHER APPLICATIONS

The RCMHS is a versatile remote controlled platform that can be outfitted with other attachments e.g. a manipulator, so that the system could undertake other types of operations in a hazardous environment. Modifications to the drum / plywood handler can be incorporated to allow grasping of much larger diameter, heavier, or odd shaped objects.

The cable telemetry/video system can be replaced with a Radio Frequency system that would allow the RCMHS to have increased freedom for operations. The base fork lift truck need not be limited to propane power. The option exists to have the RCMHS controllers and attachments placed on electric, diesel, or gasoline powered lift trucks or tow motors.

ACKNOWLEDGEMENTS
J.Shepard, Program Manager, RCMHS, Spar
H.J.Tucker, Manager Mechanical Engineering, Spar
F.Ito, Manager RCMHS Program, FERMCO

56-3
DRUM INSPECTION ROBOTS: APPLICATION DEVELOPMENT

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ABSTRACT
Throughout the Department of Energy (DOE), drums containing mixed and low level stored waste are inspected, as mandated by the Resource Conservation and Recovery Act (RCRA) and other regulations. The inspections are intended to prevent leaks by finding corrosion long before the drums are breached. The DOE office of Science and Technology (OST) has sponsored efforts towards the development of robotic drum inspectors. This emerging application for mobile and remote sensing has broad applicability for DOE and commercial waste storage areas. Three full scale robot prototypes have been under development, and another project has prototyped a novel technique to analyze robotically collected drum images. In general, the robots consist of a mobile, self-navigating base vehicle, outfitted with sensor packages so that rust and other corrosion cues can be automatically identified. They promise the potential to lower radiation dose and operator effort required, while improving diligence, consistency, and documentation.
The Stored Waste Autonomous Mobile Inspector, or SWAMI, has been supported by the Robotics Technology Development Program (RTDP) and built by the Savannah River Technology Center (SRTC), with assistance from Lawrence Livermore National Laboratory (LLNL). Other systems have been administered by the Morgantown Energy Technology Center (METC) as Program Research and Development Announcements (PRDA's) with OST support. These systems include the Autonomous Robotic Inspection Experimental System, or ARIES, built under contract to SCUREF (South Carolina Universities Research and Education Foundation) by a team from University of South Carolina, Clemson University, and Cybermotion, Inc. Additionally, the Intelligent Mobile Sensing System, or IMSS, was built at Lockheed Martin Space Systems in Denver. Finally, a stand-alone image analysis method entitled Automated Baseline Change Detection (ABCD) system is being developed under a NETC PRDA with Lockheed Martin Missiles and Space in Palo Alto, CA..
The Fernald Environmental Management Project (FEMP) has been an early testing site and has worked extensively with the development teams in an effort to more closely link potential customers to the researchers. This work has included development of site specific inspection procedures and standards, design of a test plan to evaluate the robot's performance, modification of facilities, and the operation and test of the prototypes. To date, only SWAMI has been tested at Fernald, though the IMSS and ARIES are scheduled for testing in 1996. The systems will ultimately be evaluated in a "bake-off", supported by the Mixed Waste Focus Area (MWFA) and organized by Fernald. The comparative test, to be held in about a year, will identify what systems may be ready for commercialization and

which parts of others might best be incorporated into a final commercial product. Early indications are that the application is more challenging than first anticipated, due to variations in lighting, environment, facility and drum layout. These circumstances strengthen the need for identification and integration of the best demonstrated available technologies, accomplished through mutual development and competition of alternate technical approaches.

INTRODUCTION

This paper describes the development of a new application for robotics, waste drum and facility inspection, and the effort to evaluate its utility in the field. The work has been conducted at the Department of Energy (DOE) Fernald Environmental Management Project (FEMP), by the Technology Programs (TP) department of FERMCO, Fernald's prime operating contractor. TP has been a technical facilitator, integrating the needs of end-users and efforts of robot developers. To describe this process, the baseline manual practice and generic technology are first outlined, followed by details of the systems that have been built and the on-going program to test them. Experience to date and plans for completing the program are then presented.

Inspection robots perform drum and facility inspections by visually assessing drum condition and other aspects of the facility and inventory. They identify containers by their bar codes, capture and store their images, and evaluate them for signs of damage through machine vision techniques. Corrosion cues identified can include rust spots, streaks, and blisters. Dents, bulges and tilted drums can also be recognized if special purpose sensor suites are included. Other facility inspections may also be performed, including checking the floor for elevated radiation levels or puddles, measuring ambient gamma radiation, or monitoring environmental factors such as temperature, humidity, and lighting. Inventory checking is another possible application, if a materials database is maintained. By automating the initial inspection of the containers, human inspectors only have to physically verify the small subset of containers reported as 'suspect', or possibly damaged, by the robot. The robot must be conservative in its judgements so that damaged containers are never passed as acceptable, even if this means that some containers that are actually satisfactory are added to its suspect container report. Since the primary purpose of the inspections is leak prevention, and not leak identification, suspect drums are expected to be identified before they result in material releases.

Drum inspection robots have the potential to reduce worker exposure to radiological or chemical hazards present in the waste. Inspection quality will also improve through the use of these devices. Consistent performance and diligence result from the robots unhurried, methodical inspections. For containers stacked four pallets high in a facility, drums at the top and bottom of the stack receive the same treatment as those at eye level. The devices eliminate the need for inspectors to stoop down to scrutinize lower level drums, or climb ladders to properly inspect topmost containers. Demands on human inspectors could then be reduced even as inventory increases. While people will always be required in RCRA inspections, robotics can lower the amount of labor required for the activity. The machines provide timestamped, unalterable documentation of inspection activities and drum condition by archiving images and other findings. This improvement in documentation will be a major attraction to RCRA regulators.

TECHNOLOGY SPONSORS AND DEVELOPERS

Assessment of the potential benefits of the application and the maturity of component technologies led the DOE Office of Science and Technology (OST) to support several efforts to develop drum inspection robots. Each system has similar functions, though they offer different technical approaches. The result is that teams from national laboratories, universities and the commercial sector have been able to participate in an exhaustive search for viable solutions. One such robot is the Stored Waste Autonomous Mobile Inspector, or SWAMI, built by Savannah River Technology Center (SRTC), and funded by the DOE Robotics Technology Development Program (RTDP). Image analysis and inspection reporting were developed by Lawrence Livermore National Laboratory (LLNL). Fernald developed SWAMI's archiving and database access routines.

The Morgantown Energy Technology Center (METC) issued Program Research and Development Announcements (PRDA's) for two other full systems. The Intelligent Mobile Sensing System (IMSS) was built at Denver, Colorado, by Lockheed Martin Aerospace. A prototype demonstration was held in April 1995. The Autonomous Robotic Inspection Experimental System (ARIES) robot has been built by University of South Carolina (USC), Clemson University, and Cybermotion under a contract with SCUREF (South Carolina Universities Research and Education Foundation). It was demonstrated in November 1995. METC has also more recently issued a PRDA with Lockheed Martin Missiles and Space for the Automated Baseline Change Detection (ABCD) system, a novel approach to image analysis that compares subsequent inspection results to identify corrosion cues. The DOE Mixed Waste Focus Area (MWFA) recently launched a follow-on initiative to test systems and integrate the best elements into a final, commercial ready system. This represents a renewed and more comprehensive approach towards application development in this field.

APPLICATION DEVELOPMENT PROGRAM

It has been found that though drum deployment in facilities varies significantly across the DOE complex, the need to inspect drums is very common and a growing challenge. At Fernald, the facilities are not highly contaminated, waste inventory has been plentiful and ambient radiation levels are low. At some other sites, the waste emits significant gamma radiation and thus renders the inspection task more hazardous. This has made the Fernald facilities an appropriate initial test area. The goal of the application development and testing program has been to discover the optimum level automation and functionality, in accordance with user desires, current technical capabilities, and common sense.

The full impact of a new technology is difficult for anybody to predict, including waste facility operators. End-users must learn technology limitations, and developers should appreciate the full range of environmental variability in the field. For the drums inspection robotics program, this started with a dialog between the two groups. Specifications matching technology supply to demand could then be generated. Contacts were made with regulators, a set of tests was prepared, performance criteria for application acceptability were outlined, and facility modifications were determined. As projects progressed it became clear that the amount of variability in the field was greater than had previously been envisioned. Teams then had to redesign to meet performance requirements, leading to modifications of testing program scope and schedule. Priorities for Fernald's waste facilities have since changed and its inventory is expected to be

eliminated in a short time. Other sites have expressed a strong interest, leading possibly to different application requirements. Thus, the testing program has required almost as many iterations at the technical approaches to the application, demanding flexibility and adaptability. Identification of user needs was a first task for all the development teams, and they toured several DOE sites to investigate the need and interest in drum inspection robots. In addition to Fernald, several other sites have expressed interest, including the Idaho National Engineering Laboratory (INEL) and Los Alamos National Laboratory (LANL). At the latter site, previously buried drums of transuranic mixed waste are being excavated and will greatly increase inventory, as no disposal has yet been envisioned. Additionally, Hanford had plans to use automated inspections in a new waste processing facility, and at one point was scheduled to demonstrate the IMSS in their RCRA storage areas. Based on the potential applicability on site and the willingness of FEMP Waste Programs Management (WPM) to give SWAMI a chance to show its potential, a demonstration at Fernald was planned early in the RTDP program.
Ultimately a single demonstration site at Fernald was selected. TS-4 is a translucent, 90 by 450 foot Tension Support (TS) building with a 12,000 drum inventory in regularly spaced aisles. Fernald information has also been shared with the ARIES and IMSS teams. The ARIES project team has since expressed an interest in tests at Fernald, and the IMSS team followed suit after plans at Hanford fell through.
Early activities were focussed on producing guidelines for the development of the ultimate SWAMI system while an initial testbed, SWAMI I , was being developed at SRTC. Toward that end,a site use requirements document was written that outlined the technical requirements for use at Fernald. This included a request for four-high stack inspection of multiple drum sizes, an 'aisle abort' feature, and full accountability of all drums in the facility. It was noted that the robots must achieve a level of robustness in operation and inspection reliability so as to give credence to its potential for daily usage. The document also surveyed the drivers and practice of inspection at Fernald.
A Work Plan was written to inform the local DOE field office and RCRA regulators in Ohio about the intended demonstration and proposed acceptability criteria. The goal was input from the regulators on what tests would constitute a sufficiently strong case that robots were at least as good or better than human inspectors in certain aspects of inspection. A set of tests to be used in the actual demonstration was described. Success criteria were tuned to the ultimate customer's needs and represented achievable though technically challenging benchmarks. The response from the Ohio EPA was that they would have no comment on acceptance criteria until the technology was more mature. However, they did suggest that the accuracy of current inspections should also be assessed.
A Test Plan was then developed, detailing procedures to systematically evaluate SWAMI for suitability of use in the field. Many of these tests were focussed on production-quality machines that are ready to be operated by site personnel. The original test plan subsequently was modified to match changes in technical scope and schedule. The revision focussed on checking a list of 'baseline' and 'plus' performance goals. The attributes separated achievable near term goals for core functions from desirable but nonessential 'plus' features. SWAMI was then brought to Fernald and tested. A demonstration showcasing SWAMI was held in

December 1995. In the last year, the IMSS and ARIES have also had major demonstrations, in mockup drum storage areas. The MWFA is now supporting an effort to consolidate the work that has been performed to date in this research area and identify the best individual solutions to the overall application. The centerpiece, dubbed the "bake-off", is the comparative test of each robot, in the same facility and in short succession. This will not necessarily be held at Fernald. In advance of the bake-off, a new outreach effort is being made to develop a user's group to supplement early Fernald application guidance to the developers. The users will also be asked to help develop the a new test plan, in conjunction with input from the development teams. The bake-off will occur in early 1997. Before the bake-off, IMSS and ARIES will be given the opportunity to bring up their sensor suites for data collection and then test their full systems at Fernald, as SWAMI has done, over the next six months. The bake-off site will be selected based upon the highest level of commitment to end-use, from the user's group.

CURRENT INVENTORY AND INSPECTION PRACTICE

Clearly an important element of the new application is the environment in which the machine will operate and the baseline practice that it is supplanting. Those issues are considered in this section. At many DOE sites, mixed waste has accumulated in part because of the difficulty in disposing of it. At Fernald, operations resulted in the accumulation of over 100,000 drums of mixed and low level waste. Legacy waste from previous operations is likely to have been originally stored in drums. Newly generated waste is now often placed in B-25 containers and larger boxes because of their better packing efficiency. All stored waste is subject to inspection requirements. The requirement for inspection of Mixed Waste inventory and facilities is found in the Resource Conservation and Recovery Act (RCRA). RCRA is enforced at the FEMP by the Ohio EPA. Low Level Waste (LLW) also is inspected regularly. A consent decree with the State of Ohio drives the activity at Fernald. The growth of inventory and inspection demands lead to the program to develop drum inspection robots.

The machines must operate in storage facilities that range from outdoor concrete pads, to former processing buildings, or unheated temporary storage structures. 55-gallon drums are most prevalent, though other sizes are also used, including 85 and 110-gallon overpack containers. They are stored on pallets stacked up to four high, though three-high is typical. Aisle widths of 36" are most common and encouraged by the Occupational Safety and Health Administration (OSHA), though requirements vary amongst facilities and can be as low as 26 inches.

The inspections are intended to prevent leaking by early identification of corroding drums (1). Presently, they are performed visually on a daily basis and on a formal weekly schedule to meet RCRA demands. Drums are always positioned so that the side seam and locking screw for the ring that holds the top down are visible because corrosion has been found to start most frequently along the side or bottom seam. Only the visible portions of the drum are inspected. This has been found to be sufficient and reasonable by facility operators and regulators. Suspect drums are categorized according to corrosion severity, with the levels defined in Standard Operating Procedures (SOP's). The most likely cause of container degradation is rust. Dents in the containers can also potentially breach the container or, more likely, act to raise the internal stress of the container in the dented area, thereby making corrosion more likely.

Blisters usually start inside the drum, and very small defects can result in leaks. As a rule, containment ability is not affected by general external corrosion if only paint and/or minor metal flaking is occurring. Certain drums are much more likely to leak because of corrosive material within them, and these should be checked more diligently. Freeze cycles and high humidity also accelerate container degradation.

GENERIC TECHNOLOGY AND APPLICATION

The three robots are very similar in the function their machines perform. However, each has a distinct technical approach. Indeed, there are some differences in functionality, as well. The attributes that they all share in common include the main subcomponents: Base vehicle, Inspection sensor packages, Off-board operator controls, Data analysis and presentation, and wireless Ethernet communications (between the host and the robot). Individual components and techniques needed to produce these systems have all been field proven, but the integration of the subsystems into a single machine has proven to be challenging. In order to emphasize the approach towards fair and equal consideration of each system in the bake-off, common elements in all three robots are described here before their individual implementations are introduced. Some of the technical issues faced in application are also detailed.

Self-navigating mobile robots used as the base vehicle for these systems have been under development for over ten years and are commercially available, though the requirements for specific applications often cannot be readily met by off-the-shelf equipment. Several commercial systems have been field-proven to not harm people, the environment or themselves. The basic function of the vehicle in this application is to position sensor packages near each drum in the facility so that it can be inspected, and to house the sensors and actuators that comprise the mission package. For this application, additional battery power for the sensors, a reduced size to navigate in aisles and access all drums, and tolerance to temperature extremes are required.

The machines autonomously travel throughout the facility using an internal map of the environment that originates from the standard CAD files. Navigation techniques including dead reckoning, active or passive landmarks, and local features are used to constantly improve the robot's estimate of its position on that map. Obstacle identification capabilities give the robot the ability to find and then navigate around objects and are accomplished by a redundant set of tactile sensors (bumpers) and non-contact laser or ultrasound ranging systems. Most of the vehicles include autocharging systems so that the robot returns to the base station to recharge when necessary. On-board computers are provided as part of the base robot and the level at which external computers interface with them vary amongst manufacturers. In some cases modifying or externally controlling the machine can be hampered by the manufacturer's restriction of source code access.

Variations in drum position, aisle location, lighting, and temperature are amongst some of the environment-driven challenges for this application. Solutions are available but can complicate system design. In real facilities, aisles are frequently created, rearranged and relocated. They are not necessarily all the same length, or width. Whole drum rows are dismantled to access drums at the far end, resulting in changes in drum and pallet location. Multiple size drums are often stored in a single row, and the arrangement of the drums on a pallet varies from size differences and random placement error. Temperature extremes must be

tolerated, since most storage areas are unheated. Some water and ice is also possible. Untended operation is desirable so that the robot can operate when there is less activity in the warehouse. For some machines, night operation is preferred because it reduces the variation and intensity of background light.

Inspection packages on the robots consist of color cameras, strobe light illumination and barcode readers at a minimum. Dents and other topographic features can be inspected by using structured lighting or other methods. With structured lighting, a light emitter such as a laser and a camera that captures the reflected light. The surface can be then be reconstructed using geometric models of the light source, receiver, and ideal drum surface, when the distance from the drum to the robot is known. Multiple sensor packages are used on all of the drum inspection robots to increase throughput since data collection is time consuming. Multiple drum images may be taken to capture corrosion features on the edges of the visible drum surface. The number of images needed is also dependent on the drum size, which can vary in a facility.

Drum center sensors are included in order to position the robot in front of a drum stack. Additional actuators are often used to give the proper standoff for data capture, reposition the cameras for multiple images per drum, and cover more than one level per sensor pod. Supplemental on-board computers are used to control the added equipment and provide an interface to the base vehicle and host workstation. Drums are not necessarily stacked evenly on each pallet, and some systems include fine positioning capability to adjust for offsets either across or along the length of the aisle. Designs must consider the need to inspect all the drums at the far end of dead-ending or dog-legged aisles.

A great deal of data is gathered during robotic inspections. Currently, the robots typically process three or four-high stacks of drums in 60-90 seconds. Multiple images for each drum must be uploaded to the host workstation and commands from the base computer must be received. Wireless Ethernet has been the solution selected by all development teams. The transceivers do not require special frequencies and they support the most common protocols. However, transmission within the stacks of metal drums may be somewhat unreliable. It can usually be improved by moving into direct line of sight with the base antenna. The amount of data transmitted depends on whether images are compressed, if analysis occurs on board, and whether images of non-suspect drums are also retained.

Lighting has a strong effect on the success of image analysis used in the drum inspection application. Images can get overexposed in bright light, sodium lights artificially increase the amount of red in the image, and strobes can create blind spots. Lasers used for dent detection, drum center finding or other purposes are susceptible to wash-out and specular reflections on glossy black drums. Unfortunately, glossy black is a very popular drum color at Fernald. Higher powered lasers can be used but they require special safety precautions that could impede other facility operations. Streaks should not be confused with the handwriting or stencils often found on drums. Red paint on labels or color coding on drum ribs should not be identified as rust. Straps securing top level drums should not be confused with dents.

Special purpose hardware is needed for image analysis. Because of the large amount of data and the sophisticated machine vision techniques used, processing the drum images can be time consuming. Inspection goals

were developed for the SWAMI project and represent a reasonable match between technical capabilities and application requirements. Through additional interviews with expert inspectors and input from the new MWFA drum inspection robotics user's group, these values will be revised prior to the bake-off. The current standards are for rust spots greater than 1/4 inch diameter, vertical discoloration streaks larger than 1/4 inch wide and 6 inches long, and dents larger than 1 inch depth, 1.5 inches wide and 2 inches long. A specification for blisters was not identified at that time. However, blisters as small as 1/8 inch in height and diameter have been found to result in leaks. They are also challenging to identify since they may have both discoloration and/or relief cues that must be successfully detected. For all defect features, false positives are somewhat undesirable but false negatives are much less acceptable.
Base stations for the machines consist of a charging area and one or more host workstations. The robot is operated through graphic-based controls that provide for facility set-up, task inception, monitoring and control, results presentation and archiving of inspection records. To check inventory location records, as well as for mission planning and reports uploading, interfaces to site-wide material databases are often provided. At Fernald, an ORACLE database called the Site Waste Information Forecasting and Tracking System (SWIFTS) is used. Several other sites store their waste inventory records in a similar format. Printers, additional mass storage, and high quality color monitors are also supplied. Some of the development teams have envisioned the development of features to reduce the occurrence of false positives. This would allow insignificant artifact or corrosion features to be ignored on subsequent inspections. For rust spots, if the size increases over a threshold between inspections, the feature would again be highlighted.

SYSTEMS UNDER DEVELOPMENT

Fig. 1

SWAMI

A first SWAMI prototype was demonstrated in a mockup area at Savannah River Site in November 1993 (2). Figure 1 shows SWAMI II as it was demonstrated in December 1995 at Fernald's TS-4. It is a modified HelpMate robot from Transitions Research Corporation. It has automated battery charger docking and sophisticated obstacle avoidance. It also has many enhancements including local aisle following, a system from the University of Michigan for backing out of dead-end aisles (3), a floor radiation survey system, image capture and processing, and wireless Ethernet. An off-board computer is used as the primary interface to the robot, and a separate VME card cage houses electronics dedicated to image processing. The rad survey system uses scintillation counters instead of gas proportional detectors found in previous SRTC robots. It was tested offline but not as part of the integrated system. The on-board supervisory controller consists of three VME based microcomputers that utilize the GENISAS control software from the RTDP to dispatch tasks between a supervisory control system and various subsystems.

Two sensor pods are positioned by a vertical mast on the robot high enough to cover four-high stacks of 85 gallon drums. The pods are offset by the height of two drums so that the first and third drums are inspected before the second and fourth. Each has cameras and strobes for image capture, a barcode reader, and lasers with black and white cameras for dent detection. Two image cameras cover the required field of view while three sets of two lasers and a camera are used for dent detection.

Drum centers in a stack are individually found by a special purpose laser system which then adjusts the position of the sensor pod to achieve optimal height, reach, and lateral offset to the inspected surface by individual linear motions. The lasers are all rated Class 2, which is essentially eyesafe. The pods are attached behind the robot, inspecting only one side of the aisle per pass. They fold so that the robot can spin at aisle dead-ends and then inspect the other side on the return path. If end-of aisle clearance is less that 8 feet, the robot backs out of the aisle and then re-enters backwards so that the other row can be inspected. The data is compressed on-board and transmitted to the base station.

SWAMI off-board equipment includes base station and vision processing computers, printers, and a charging station. The main operation interface for daily system users is called the SWAMI Operator Interface (SOI). Drums to be inspected are selected from a map of the warehouse updated with the latest information from the site's SWIFTS database. Simply by selecting one or more row halves, a mission profile is automatically generated and downloaded to the robot. It also highlights suspect drums that have been found and identifies drums that are out of place as compared to the database.

Images are uploaded and then analyzed on the vision processing computer as they are collected and then uploaded from SWAMI over one of two wireless Ethernet channels. This process starts in the field and continues after the robot has returned to recharge, because of the long time required per picture. The method of rust identification includes color recognition and thresholding, noise reduction and region growing. Streaks are also found based upon their orientation and shape. Structured light data is also uploaded and then analyzed for dents and blisters. The inspection findings are presented through a separate window that prioritizes the drums according to corrosion severity. By selecting a drum with the mouse, a representation of the drum is shown to the operator, with the suspicious areas boxed by the program. Utilities are provided to print out results for physical verification in the field. The images are ultimately stored in a data archive indexed to the SWIFTS database.

Fig. 2

ARIES

The ARIES robot, shown in Fig. 2, was first demonstrated as an integrated system November 1995 in a cold test area mocked-up to match Fernald specifications. Hot testing at Fernald early in the summer of 1996 is planned after additional enhancements are completed. Drums of 55-, 85-, and 110-gallon capacity can be inspected in aisle stacks up to four pallets high. ARIES is the first robot to demonstrate inspection of multiple drum sizes, and the only one to inspect 110 gallon drums (4). It has an autodocking station and is capable of backing and turning around in a 36 inch aisle. Off-board control workstations provide mission planning and monitoring, while wireless Ethernet provides links to on-board computers used for real time mission control and analysis. A power management system, supplemental dexterity package, and a radiation hardening study have also been included.

The machine base is the Cybermotion K3A, redesigned by the vendor to accommodate 36-inch aisles. It navigates using dead reckoning (measuring motion distance and heading changes) and position updates to passive landmarks. The landmarks appear as tennis ball can-sized cylinders

covered with retroreflective tape. The reduced size vehicle still has the synchro-drive system that permits six wheels on three horizontal leg units to each be individually steered and/or driven. This improves accuracy of motion recording by maintaining constant wheel contact with the floor. A Camera Positioning System (CPS) is installed on the robot top and moves four instrumentation packages, one for each level. Its stowed height is 10 feet and at maximum extension it is 16 feet tall. The top three levels are positioned using an extending mast in which one linear motion controls the height of the mast and another controls the separation between pods. The lowest level pod is dropped behind the robot using a mechanical linkage. Each image acquisition subsystem consists of a camera with a strobe lamp above and below it. Both strobes are fired in sequence so that reflections can be subtracted from the image. A laser structured light source projecting five dots against drums is used to identify the drum size, drum center location, and tilting. All the lasers used are eyesafe.

Color processing, using specialized algorithms, incorporates supplemental multi-strobe lighting and differential strobe based structured lighting. The design goal was to detect rust larger than 1/2 by 1/2 inch. With wide angle lenses, this leads to a requirement for two images per 55 gallon drum. The images are stored in a Hue-Saturation-Intensity color representation from which a range of values can be attributable to rust. Dent and streak detection are not currently supported though a solution to blister identification is included. This is done by modeling the blister as a conglomeration of small bubbles protruding from the surface, and then performing a frequency analysis of the image intensity. The images are matched and reconnected for clarity in presenting results to the operator. This is achieved by blanking out all regions judged to not be on the surface of the drum, using the projected laser spots in the drum image to guide the cropping process (5). A single drum is currently processed in six seconds, and the time required to inspect a single four high column and move on to the next is 1 minute. ARIES present throughput is about 2,500 drums per 24 hours.

Robot operation is controlled from a Unix workstation in a portable software environment that has been designed to be scalable to smaller systems for commercialization. However, path and facility simulations that have been developed work best on Silicon Graphics systems. A series of menus guide users or site managers through the facility setup, mission profile, dispatch and monitoring, all activated from the main program. The facility map is derived from an AutoCAD file. A path assembler is used to generate the mission script with location attributes and procedures along a series of linear path segments.

Fig. 3

IMSS

The IMSS was demonstrated at the Lockheed Martin Denver facility in April 1995. Figure 3 shows the machine as it appeared during the demonstration. Hot testing at Fernald in the spring of 1996 is also planned. Unlike the other two robot development teams, Lockheed Martin built a custom designed robotic base (6). The welded frame, steel skinned vehicle is quite narrow and can thus enter aisles as small as 30 inches wide. It has special Mecanum wheels that allow it to move or rotate in any direction including sideways. Obstacle sensors consist of ultrasound range sensors and miniature limit switches set behind the sheet metal so that they trip upon contact. They are hard-wired directly into the power distribution

system. Aisles are expected to remain fairly constant in location and size, for the current system. The waste facility is described as a main corridor and a series of evenly lined pallet stacks that define the aisles between rows. The robot uses its side ultrasound sensors to align itself with aisle ends and to count aisle entrances. 4-5 hours battery life is the maximum obtainable over the full temperature range specified, using a sophisticated set of sealed Nickel-Cadmium batteries that eliminate the explosion hazard from hydrogen off-gas during recharge. At room temperatures, charge capacity is even greater.

Three fixed sensor arrays are mounted on a vertical mast to inspect up to three stacks of drums concurrently. Extensions to four high stacks are also possible. Each sensor suite consists of two ultrasound sensors used to determine drum centers, a barcode reader, two color cameras and diffused halogen strobe for rust inspection, and a class 3A laser and B/W camera for dent and tilt detection. Two vertical parallel axes of motion move the sensor suites left or right so that the optimal standoff distance and angle is maintained. The mast can swing to the front or rear of the vehicle to inspect drums at the end of an aisle. Four camera shots are required on 55 gallon drums in order to meet detectability goals. Tilt axes on the sensor suites allowed full coverage of the drum from its center.

Images are stored in uncompressed format, but only when defects are found. Multiple sensor arrays are used to improve inspection time. Streaks and blisters were not addressed in the first prototype. Dents on the flat cylindrical surface of the drum, as well as tilting and bulging, were detected using the structured lighting system. Some difficulty was reported with glossy black drums in fluorescent lighting, due to the reflectivity of the surface.

The operator interface presupposes that there might be multiple IMSS robots operating simultaneously. Aisles to be inspected are chosen by double clicking on the spreadsheet entry for that row. Rows are assumed to be all the same length presently. Post-mission, a newly compiled defects database is presented with each feature as a separate spreadsheet entry. The workstation communicates with the robot through a wireless Ethernet. They included the concept of a "baseline" mission, in which a defect database is generated. This database can then be consulted so as to reduce false positives by noting already present defects that are not severe. Now the development team will be integrating the ABCD image analysis system with the IMSS which uses a similar type of baseline inspection method.

Automated Baseline Change Detection

The Automated Baseline Change Detection (ABCD) project seeks to develop enhanced analysis capability for autonomous inspection, that could be useful on any of the robotic vehicles described above. The effort got under way in 1994 and will result in a prototype system ready for demonstration in Fiscal Year 1996. The system compares a current inspection image with an archived baseline image. Any change is identified. If further interpretive analysis verifies that the change is benign, then no action is required. If the change is not benign or is not recognized, then human operators are notified. The key to this process is the use of a commercially available system to identify a target label affixed to all drums, compute sensor distance from the drum, and then rapidly and precisely reposition the sensor. Image transformations are

used to match the pose of the newly acquired image to that of the archived baseline.

OPERATIONAL TESTING AND QUALIFICATION

The most important evaluation to be made, and the one that best summarizes robot's effectiveness as an inspector, is the side-by-side comparison of the robot's inspection reports with those generated using existing methods. Thus, during field trials, the waste storage facilities are to be inspected by both the robotic and human inspectors. Accountability for all drums in the warehouse, whether checked by robots or people, is required to guarantee that every drum has been inspected. Throughput and reliability are also very important. All these tests require the machine to be evaluated over as long a time as possible so as to be able to compare machine results against manually generated weekly RCRA reports.

Typically, end-users want to minimize facility changes and developers want to permit facility changes that reduce variability. For SWAMI tests, a moderate set of facility modifications was agreed upon, including the extension of aisle ends and the enlargement of barcode labels. The first test plan, released July 1995, described the full evaluation of SWAMI, its supporting equipment, and the documentation, from the perspective of a DOE site end-user. Its intent was to evaluate a machine nearing production-ready status, outlining ten test methods to evaluate the system from a subcomponent and system level. It called for a Pre-Start Audit addressing basic safety concerns to be completed before other tests are run. A series of offline, or one-time tests were then described, including Safety and Diagnostics, Operator Interfaces, Vehicle Locomotion and Power, Navigation, Barcode Reading and Inventory Checking. During Operator Interface tests, the usefulness and clarity of the controls were to be evaluated as the procedures were simultaneously documented.

It became apparent that intermediate data collection exercises would be necessary before production-quality testing. Other development teams also requested access to the facilities for development purposes. Image data from the warehouses up to that point consisted of video footage and photographs of the drums. Two new types of tests were identified: data collection using individual teams's hand-carried sensor suites, and developmental testing of the robot in the facility, allowing for technical iterations and improvements in the field. The tests acknowledge the value of actual data to the developers. For instance, the design and optics of the camera system, as well as the positioning accuracy of the sensor mast, have a great effect on inspection performance. For the SWAMI project, sensors were brought to Fernald to get better system training data, and this proved very useful in developing algorithms at the laboratory.

In the final SWAMI test plan, radiation detection system tests were removed, others consolidated, and procedures were outlined in greater detail. The final test plan describes two methods to study robot performance over a substantial period of time. The Duration test method compares robotically and manually collected RCRA reports and other key performance attributes, such as the incidence of false positives and false negatives, throughput, and labor requirements with automation. An identification exercise was also included, to compare human and robotic attentiveness. Colored dots were to be randomly placed on visible drum surfaces and then both inspection methods would be compared for completeness of coverage. This test requires additional routines in the

robot's image analysis modules, to separately tabulate the count of dots. The other robot feature requested to support testing is a reporting function to record and timestamp selected internal machine variables such as position, heading, or recent actions.

The Inspection Data Capture and Analysis test method focusses on determining inspection accuracy for each corrosion feature type (rust, streaks, dents, blisters), detectability as a function of feature size and location, and sensitivity to aisle placement or drum color. This is accomplished through the addition of a test aisle in the facility. The test aisle consists of a number of empty drums, some of which, called 'ringers', have intentionally placed defects whose description and location have been documented. These ringers are stored, swapped and moved amongst the population of test aisle drums. Four 'Feature Standards' were also included, one for each corrosion feature type. Graduated sizes of dents, rust, etcetera were created on each, from below to above pre-agreed detection limits. The blister standard was made by initiating corrosion from the inside, using a strong acid. Four 'Color Standards' were also created, with random corrosion features placed on red, orange, white, and flat black drums. They were to be used to investigate inspection accuracy as a function of drum color.

SWAMI was received at Fernald in September 1995 and was released in late December, following its demonstration. A technical team from SRTC and LLNL completed system integration and operated the robot during the tests. Access requirements for workers were met with 24 hours of classroom safety training. Baseline functions met during the tests included navigation to straight aisles and through the facility, drum center finding, rotation at aisle ends, image acquisition, barcode reading, and night operation. Data upload rates were determined to be 1,000 drum records in 13.6 hours, and other user interface control and reporting goals were achieved. SWAMI can accommodate drum stacking error with its two linear motions per sensor pod.

Image analysis and geometric (dent) inspection could not be demonstrated as part of an integrated SWAMI system but good results were achieved using images collected during previous data collection exercises. Streak detection, imaging of dents, and rust detection were shown. Blister detection was considered a 'plus' item and has not been completed. Because the inspection software could not be integrated with the robot in time, the two key duration tests could not be conducted. However, SWAMI is expected to participate in the bake-off, at which time it will have an opportunity to be evaluated.

The IMSS and ARIES have both been demonstrated in mockup facilities within the last year and will complete their next developmental phase by the end of 1996.They have not yet been tested at Fernald, and they will have that opportunity this coming year. The step following that is the MWFA bake-off, a side-by-side comparison of the drum inspection robots. A new test plan, set of facility modifications, and success criteria will be developed for the bake-off with fresh input from the end-users, principle investigators, and project managers. The goal is to consolidate the best components from all development efforts into a single robust and practical container inspection robot. A new User's Group is being formed and will be a key contributor to the project. The group will provide guidance as to what functionality should be targeted for demonstration by technologists and a new and revised set of requirements for use at a typical DOE site. They will also be asked to contribute to the

development of both the test plan and the criteria used to judge success. The bake-off site will be selected from this group based upon the level of interest and the availability and suitability of facilities for the demonstration. Several sites have already expressed an interest in hosting the bake-off and working these machines into their long-term waste facility operations plans.

CONCLUSIONS

So far, each robot has presented unique capabilities and some shortcomings. Though developmental testing is not yet complete and projects are not at the exact same developmental phase, some comparison is possible based on early demonstrations. The IMSS currently inspects 3-high stacks and features dent detection. It is also the narrowest vehicle, and thus can be used at a greater range of facilities. Inspection results are expected to improve with the addition of the ABCD system, which will be tested on the IMSS during the bake-off. ARIES has done an excellent job in vision system development and is the only system currently inspecting 110-gallon drums and identifying blisters, in stacks up to 4-high. They have not yet addressed streaks and dents, however. SWAMI is very tolerant of real-world aisle, pallet and drum position variations and has multiple redundant systems for obstacle detection. It is also reporting the best results for inspection of glossy black drums and identification of dents on drum rims. SWAMI can only inspect 55 gallon drums currently, however, and presently its laser based inspection and drum centering do not work in bright light conditions.

The need for improvements in inspection accuracy, completeness and efficiency is still strong as is evidenced by interest from various sites. However, the application has proven to be more challenging than originally anticipated. The fact that individual subcomponents have been proven in the field does not imply that a machine full of them will work reliably and robustly in the first design iteration. Because of the diversity of the teams that have independently pursued this problem, solutions to most of the challenges in the application have been demonstrated by at least one team. However, none of them have yet progressed to pilot-scale level performance. With the bake-off site undetermined at this time, an appropriate final testing location will now be sought so that the system is first optimized for an end-use site. Flexibility and accommodation have been required to foster this new application, and will remain important. The path towards integration of system components and technical maturity will not be easy, but the value and potential for drum inspection robotics remains strong.

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DEVELOPMENT AND INTEGRATION OF THE LIGHT DUTY UTILITY ARM SYSTEM

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ABSTRACT

The Light Duty Utility Arm and Deployment System (LDUA) is being developed by Spar Aerospace Limited under contract to the Westinghouse Hanford company. This robotic system will be used for the characterization and removal of nuclear waste stored in underground storage tanks (UST) at many of the U.S. Department of Energy (DOE) nuclear weapons facilities. It will also be used to assess the structural integrity of the tank walls.

The design and development aspects of the LDUA system are described. An overview of the integration of the various subsystems into a system is provided as well as the methodology used to verify system performance. Lessons learned and problems encountered are identified.

INTRODUCTION

The Light Duty Utility Arm System (LDUAS) will be used to perform surveillance, inspection and characterization activities in support of the remediation of underground storage tanks containing hazardous, radioactive wastes. It may also be used to perform light utility tasks and remediation.

The Light Duty Utility Arm System (LDUAS) is illustrated in Fig. 1. It is a truck-mounted mobile system which is erected at a tank site. A telescopic mast inserts the seven-degree-of-freedom 13.5 ft-long manipulator through the tank's 12 inch diameter riser a distance of up to 47 ft. into the UST. Before entering the tank, one of many different end effectors is attached to the end of the arm. The arm is controlled from an operations trailer which can be located a distance of up to 950 ft. away. The system features automatic collision avoidance and a graphical user interface permits off line path planning and training as well as on line graphical visualization to augment CCTV views.

Fig. 1

The design and development of the LDUAS involved a number of challenges due to the operational requirements, dimensional constraints and harsh environment. Some of these requirements are listed below:

To deploy into the tank through a 12 in. Schedule 40 riser and to provide a reach envelope 47 feet deep x 13.5 feet horizontal radius (see Fig. 1).

To include dexterity and kinematic redundancy to reach around obstacles obstructing the reach envelope.

To provide good teleoperator/telerobotic operating characteristics, e.g., good resolved motion, minimal oscillation, smooth motion.

To supply a large number of utilities (power, signal, video, water, air) to a variety of user-supplied end effectors mounted on the tip of the arm.All components including utility lines tobe packaged within a sealed and purged enclosure to provide contamination protection, tolerance of aggressive decontamination processes and for explosion-proofing.

All in-tank components and materials to be radiation-resistant; all external components and materials to be corrosion resistant (to both acidic and alkaline environments).

All components to be accessible for inspection and maintenance.

Some of these requirements are potentially contradictory, e.g., the small access riser and the large reach envelope, the fully enclosed packaging and the accessibility. This paper discusses some design challenges arising from these requirements and how they were met. The integration and testing of the system is also described with emphasis on the main features of the test program. An important factor in the successful resolution of many issues was the cooperative and interactive approach adopted by Spar and its customer in the early stage of the project. This enabled problems to be solved quickly and efficiently by a combination of relaxing specification requirements and concept modifications.

DESIGN CAPABILITIES AND CHALLENGES

In the paragraphs below some special capabilities and the associated design challenges are discussed with specific reference to:

Vertical Positioning Mast
Manipulator Elbow Joints
Packaging and Enclosure
Purge System
Operator Interface

Vertical Positioning Mast

The Vertical Positioning Mast (VPM) is required to deploy through a riser which is nominally a 12-in-dia. Schedule 40 pipe x 10 feet long but in practice after deformations, intrusions and tolerances the available I.D. is approximately 11.0 in. yielding a specified maximum O.D. for our equipment of 10.5 in. Since the stowed length of the VPM with manipulator is limited to 35 feet, the vertical reach of 47 feet, involving a deployment stroke of 60 ft., can only be achieved by a telescopic mast (see Fig. 2). However, the annular space for a telescopic mast is extremely limited and this is compounded by the need for rolling element bearings to achieve smooth motion and the use of CRES 316 as the only available corrosion resistant tube material limits the contact stresses of rollers on the tube.

Fig. 2

The design solution is shown in Fig. 2. The use of large-diameter convex rollers with Verpel tires mounted to and penetrating the inner tube reduce the contact stresses on the outer tube. Neither of the contact surfaces (rollers or interior of outer tube) is exposed to the contaminated in-tank environment. Because of the high ratio of stroke to stowed length, a multi-tube configuration would normally be preferred but the annular space precludes the use of more than a total of two tubes. Fortunately the customer was able to extend the stowed length requirement

from 30 feet maximum to 35 feet maximum and this made a critical difference, allowing a reasonable tube overlap for a two-tube design.Nevertheless, the bearing stresses in the outer tube were high compared to the strength of the CRES 316 and careful optimization of the design was required.

This rather involved evolution of the concept has paid dividends because the result is a robust, free-moving, fully-sealed mast with a smooth and regular external profile which is easy to decontaminate.

Manipulator Elbow Joints

The LDUA is a seven-jointed manipulator and was illustrated in Fig. 1. Its two adjacent elbow joints give it excellent dexterity in terms of its ability to articulate the end effector and reach around obstructions, as illustrated in Fig. 3. Another advantage of the configuration is the short link length compared with the overall reach which enables the manipulator to deploy into a constrained workspace, e.g., within 6 feet headroom or less depending on the end effector size.

Fig. 3

The elbow joints are a special design which combines:

High torque capacity (up to 38000 in.lb.)

Large angular travel (100 deg.)

Small diameter (8.62 in.)

Free internal volume for routing the large bundle of end effector utility lines.

The use of linear actuators (hydraulic cylinders) housed inside the main structural boom provided the necessary torque capacity, while an articulating linkage maintained a high torque through the full range of travel. The novelty and high performance of this joint required an early development effort to mitigate risk. Firstly, a theoretical optimization of the linage geometry and the pivot loads was conducted. Secondly, a Proof of Principle (POP) Elbow Joint was constructed and tested. As a result of the testing the primary link and its mounting were reinforced and a suitable joint servo control strategy was developed.

The benefits of this careful development approach were realized when the real elbow joints were found to be stiff, backlash-free and readily-controllable.

Packaging and Enclosure

The LDUA packaging must accommodate the following requirement:

Strength and stiffness require closed structural sections.

Enclosure of all components for contamination protection and explosion proofing means that they must be mounted inside the structure.

Internal routing of the large utility harness which comprises power, signal, video, hydraulic, water and air lines to support a variety of end effectors as well as to operate the arm itself.

Access to all components is required for maintainability.

The outside diameter is limited to 9.0 in.

These requirements are rather contradictory. The solution is illustrated in Fig. 4 and includes the following important features:

Fig. 4

Fully-enclosed design with large, removable covers.Covers are structural elements strongly connected to arm sections topreserve section properties.

High-strength corrosion-resistant material (CRES 17-4 PH) and fasteners accommodate concentrated loads at the structural connections.

Very careful internal packaging of components, verified by full-scale mock-up.

Purge System

The purge system renders the LDUAS in-tank components explosion-proof and prevents the ingress of particulate or liquid contamination into the interior of the LDUA and VPM. It achieves this by maintaining a positive pressure within the LDUA body and VPM tubes relative to the tank atmosphere so that no explosive gases can migrate inside and come into contact with electrical devices or hot components. The purge system is required to conform to ANSI/NFPA 496 for Class l, Division l, Group B components.

The purge system schematic is shown in Fig. 5. The purge air or gas supply is connected to the purge air coupling on the VPM housing. Inside the VPM the supply pressure and flow are regulated and routed through the VPM and LDUA utility harnesses to the TIP. When an end effector is attached to the LDUA the purge air or gas flows through the TIP hose coupling into the end effector. The return flow is via the interior of the LDUA, through the VPM inner and outer tubes and into the VPM housing. The VPM housing will be maintained at a pressure which is positive relative to the tank atmosphere (to ensure positive pressure in the LDUA and VPM tubes) and negative relative to the ambient atmosphere (to ensure containment in the event of leaks). This is accomplished by permitting the purge air to exhaust to the TRIC since the TRIC pressure is controlled by the buyer's ventilation system to be positive relative to tank and negative relative to ambient.

Fig. 5

The LDUA and the VPM tubes are thus explosion-proof both because they are maintained at a positive differential pressure relative to the tank atmosphere and because there is a significant purging flow through them. The purge system ensures this by monitoring the internal pressure at the top of the VPM inner tube and the VPM housing relative to the tank. The differential pressure at each location is monitored by two independent sensors connected to independent control functions in the control subsystem. One sensor is connected to a hard-wired switching circuit which cuts off all power to the LDUA if the differential pressure falls below a certain threshold. The second sensor is monitored by the system software which performs the same function and provides an error signal to the operator. This arrangement ensures that no single or common-mode failures can prevent the de-energizing of the LDUA if it is not maintained at a positive differential pressure relative to the tank.

Operator Interface

Operation is normally from a Remote Control Trailer located outside the Tank Farm fence. The Spar equipment includes two workstations, which together with joysticks and video monitorsprovide full control of, and feedback from the system. The operator interface has been designed to be User Friendly so that operators can learn and become comfortable with the system as quickly as possible.

The main control console uses a Sun Workstation with intuitive graphical point and click screens to guide the operator through the start-up, deployment and shutdown phases. Telerobotic control of the LDUA is via two 3-degree-of-freedom joysticks (one rotational; one translational) which allow either single joint or resolved motion commands. Visual feedback is provided from the LDUA shoulder camera, one or more in-tank "overview" cameras, and various video or 3-D laser end effectors.

A second (Silicon Graphics) workstation running Spar's Supervisory Graphics Control Software (SGCS) and Deneb's IGRIP simulation package provides a 3-D kinematic model of the LDUA arm and VPM which can be moved around inside a "world model" of the tank interior derived from the customers' 3-D mapping of the tank. Complete inspection missions can be simulated and recorded, potential collisions between the LDUA and tank obstacles can be detected and alternate paths chosen to avoid collisions, and optimize the inspection task. The planned mission can then be implemented and the simulation observed in real time to supplement the visual information provided to the operator.

INTEGRATION AND TEST

The integration and test program provides a logical and structured flow from assembly and subsystem through to final integrated system testing. The main steps in this process are:

- Manipulator Performance Test
- LDUA System Acceptance Test
- LDUA Inspection System Test and Evaluation at the Hanford Cold Test Facility

Manipulator Performance Test

After verifying that the manipulator has achieved its basic mechanical functionality, the manipulator and controller are connected and subjected to a series of tests designed to characterize and tune the manipulator performance. The test arrangement is illustrated in Fig. 6

Fig. 6

The first step is to carry out joint servo loop tuning to optimize individual joint performance. In the second step, the arm is operated in its resolved motion modes to verify and optimize the arm level control (ALC). This involves adjusting parameters such as washout spheres, transition and position hold spheres input command filter characteristics, joint position limits, self-collision avoidance, elbow and torque limit avoidance and LDUA deflection compensation.

These tests also demonstrate that the arm can meet its specified reach envelope, velocity and payload requirements together with its budgeted repeatability and resolution.

Considerable care is required during these tests, particularly in the early stages when full controllability is not established, because loss of control or operator error can result in collision with the floor or the test fixture.

System Acceptance Test

The system-level testing requires a special facility because the total height of the fully-deployed system is nearly 100 ft. (See Fig. 1). The selected facility is a partially-completed and mothballed thermal generating electrical power station currently used primarily for storage. The main turbine-generator hall provides an indoor location where the system can be craned on to the upper level and the VPM-LDUA deployed down to ground level as shown in Fig. 7. The building itself is unheated but a heated office space is available on the mezzanine level where the remote control station (console subsystem) is set up.

Fig. 7

A sequence of tests is conducted which represent most of the operational situations of the system in use, including the following:

- Ability to deploy/retract through a tank riser.
- Resolved motion including the VPM and its deflection compensation.

Payload capacity, velocity, and repeatability and resolution at various locations within the reach envelope.

Smooth motion and minimal oscillations.

Purge system functions.

Failsafe braking, health monitoring, interlocks and other safety features.

LDUA Inspection System Test and Evaluation at WHC

After delivery to WHC, the LDUA System will be installed in their indoor Cold Test Facility which provides adequate headroom and allows the VPM to be fully deployed into a vault containing simulated waste and a partial tank mock-up. Here the Spar LDUA system will be integrated with various customer supplied end effectors and a supervisory control and data acquisition system to verify the operation of the complete tank inspection and characterization system.

This test set up will be used for ongoing sensor and controls development, to develop operational procedures, and to provide operational and maintenance training.

CONCLUSIONS

A number of design challenges and development problems have been experienced in the LDUA program. These were successfully resolved by a combination of innovative engineering, careful design and development testing conducted on breadboard, proof-of-principle and mock-up hardware. The patience and flexibility of the customer was also a key factor in obtaining technically acceptable, cost-effective solutions. The final verification of the solutions was obtained during subsystem and system testing with a test plan designed to be rigorous yet cost-effective.

The first LDUA system is expected to be delivered to the Hanford site in February, 1996. After delivery to Hanford, the system will undergo testing at the site and end effectors will be integrated in support of a deployment in an underground storage tank at Hanford in September, 1996. The versatility and dexterity of the LDUA are expected to assist DOE in obtaining vital waste characterization data regarding the chemical composition of thewaste in these tanks.

Additional LDUA system will be delivered to the Hanford, Idaho and Oak Ridge sites during 1996 and are expected to be used to characterize underground storage tanks at these sites in 1997.

ACKNOWLEDGEMENTS

The authors should like to acknowledge the support of the following organizations and individuals.

Department of Energy (DOE), Office of Technology Development, David Geiser, Tank Focus Area

Department of Energy (DOE) Tank Focus Group, James Lee, Characterization and Retrieval Program Manager

Westinghouse Hanford Company, Betty A. Carteret, Surveillance and Retrieval Technology, Applied Systems Development

Oak Ridge National Laboratory, Dr. Barry Burks, Program Manager, Environmental Restoration & Waste Management Applications

Idaho Falls National Engineering Laboratory, Cal Christensen, UST Technology Department, Integrated Product Team Lead

Spar Aerospace Limited, 9445 Airport Road, Brampton, Ontario, Canada L6S 4J3.

HIGH LEVEL CONTROL AND DATA ACQUISITION SYSTEM FOR ROBOTIC CHARACTERIZATION OF HAZARDOUS WASTE IN UNDERGROUND STORAGE TANKS

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ABSTRACT

The U.S. Department of Energy (DOE) has significant amounts of radioactive waste stored in underground single-shell storage tanks. Many of these storage tanks have reached their design life and are deteriorating structurally. The wastes are chemically and radiologically hazardous with radiation levels ranging from slightly above background to hundreds of rads/hr. The consistency of the waste ranges from pumpable liquids and slurries to thick sludges and large crystalline masses. A priority mission of the DOE is to retrieve the waste from its present storage and process it for final disposal. The Light Duty Utility Arm (LDUA), a robotic system being developed by the DOE's Office of Science and Technology, will be used to gather information about the waste and the tanks in which it is stored to better plan and execute the clean-up mission. The LDUA will become operational at the Hanford Site in 1996, making it the first robotic system to be deployed in a radioactive underground storage tank.

Sandia National Laboratories is developing both a Supervisory Control System (SCS) and a Supervisory Data Acquisition System (SDAS) for use on the LDUA. The SCS is based on graphical programming and model-based control which allow previewing of programmed robot motion by the operator before actual robot motion is performed; this advanced capability is needed for robot safety in the hazardous environment. The SDAS is an integral part of the robotic system; it collects data from several end-effectors and controls both video storage and end-effector operations. SDAS uses a client/server architecture which allows multiple programs to receive data in real time including an operator interface, an expert system, and a data logger. This paper discusses the design and architecture of both the SCS and SDAS systems.

INTRODUCTION

The high-level radioactive wastes created as by-products of plutonium production and nuclear power generation have been stored in underground storage tanks at various U.S. Department of Energy (DOE) sites for more than 40 years. For example, at DOE's Hanford site approximately 140,000 m3 (37 Mgal) of waste material now resides in 149 single-shell tanks which range in size from 208 m3 (55,000 gal) to 3,785 m3 (1 Mgal) in capacity (1). These tanks consist of a carbon steel liner surrounded by a reinforced concrete shell. Many of these tanks have exceeded their design life and some are leaking contaminants into the surrounding environment. Characterization and remediation of these waste tanks is a high priority for the DOE's environmental restoration program. Radiation levels are too high for human workers to enter the tanks; therefore, remote means of characterization and removal must be considered. In support of this

mission, the DOE's Office of Science and Technology is sponsoring the development of the Light Duty Utility Arm (LDUA) system. The LDUA is an integrated robotic deployment system which will perform inspection, surveillance, waste analysis, and small-scale retrieval tasks in underground storage tanks (2). Current characterization methods used at tank sites are limited to positioning sensors and instruments directly below access risers. The advanced technologies being developed under the LDUA program will allow critical characterization and inspection throughout the tank, not just below the access risers.

The Light Duty Utility Arm is a seven degree-of-freedom articulated robotic arm on the end of a telescoping vertical positioning mast. The arm is designed to deploy a variety of tools (end effectors) into the underground tanks. Its redundant degrees of freedom give the arm the ability to maneuver around obstacles in the tank. The robotic arm and subsystem controller are being designed and developed by Spar Aerospace Ltd. Detailed information on the LDUA design can be found in Ref. 3.

Supervisory control and data acquisition for the LDUA will be implemented in stages. The initial deployment of the LDUA will be video and photographic surveys of the tank and waste (4). The Supervisory Data Acquisition System (SDAS), developed by Sandia National Laboratories, will control and gather video from both overview and end effector cameras. Control of the LDUA will be performed using joysticks in a teleoperated mode. The Supervisory Control System (SCS) described in this paper will be implemented on the LDUA in later missions. The SCS will improve the operations efficiency and safety of the LDUA system by adding graphical previewing and collision checking, automated path planning, and point-and-click motion to the control system. Further advances include task-level automatic sequencing and automatic trajectory planning for collision free motion in the tank.

MODEL BASED CONTROL & GRAPHICAL PROGRAMMING

A world model is a three-dimensional computer model of the robot and its operating environment. In an underground storage tank, the world model is developed with computer drawings of known aspects of the tank as well as with sensor-developed models of the unknown aspects of the tank (i.e. the waste surface.) The purpose of the world model is to allow the operator to visualize the position and motion of the robot relative to other objects in its environment. Motions are planned and previewed within the world model to verify their correctness and freedom from collisions before they are executed. This planning and safety checking in a preview mode is termed Graphical Programming.

Graphical Programming uses three-dimensional animated graphics models as intuitive operator interfaces for the programming and control of complex robotic systems (5). In remote deployment of robotic systems, such as the LDUA, the operator cannot directly interact with or view the robot motions. In a purely teleoperated control mode, the operator must guide the robot motions with input from cameras at discrete locations in the tank. The development of collision-free paths is completely dependent on the ability of the operator and the viewing angles of the in-tank cameras. In an underground tank environment, collisions (especially collisions with the tank wall) can have catastrophic effects. Graphical programming and model-based control provide a means for ensuring collision-free motion as well as reducing operator error and fatigue.

In a graphical programming system, the graphic representation of the robot, its environment, and controls allow an operator who is not an

expert robot programmer to easily interact with the robot and command safe robot motions. The user interface has pop-up menus, displays, and the ability for "point-and-click" motion commands which intuitively guide the operator through robot control. Using these graphic operator inputs, the robot's supervisory control software:

Translates commanded tasks into graphical robot motions;

Simulates and analyzes robot motion to check for safety;

Commands the robot to execute motions that have been determined to be safe;

Monitors the robot's motion to verify task compliance;

Updates the graphics model as tasks are performed by the robot.

Simulation and monitoring are key functions of the graphical programming supervisor software. Robot tasks are simulated before they are performed and the simulation system's safety validation functions determine whether the tasks can be performed safely (5). For example, motion previewing of this type is used to avoid collisions with tank walls and other obstacles within the tank. While the motion or task is being performed, the supervisory software slaves the simulation system to the robot's actual motion and monitors to verify that the move was performed as simulated. The supervisory software can also interrupt robot motion that excessively deviates from previewed paths or results in entry into hazardous regions.

Model-based control of robots as described above is dependent on a model of the environment in which the robot is functioning. On the LDUA system, creation of the world model will begin with the tank structure and will be based on best existing knowledge of the tank (e.g., design drawings and external measurements) (6). A topographical mapping system will be used to map contours of the waste surface and any unrecorded objects in the tank. Sandia has developed world model building capabilities which will rapidly create models from these mapping subsystems. As the world model is reasonably, but not completely accurate, a minimum approach distance will be established for the world model based on an analysis of all sources of error (6). In a more ideal graphical programming system, the robot would be equipped with physical sensors to compensate for any errors in the world model. For example, a pipe cutting end effector would be instrumented with proximity sensors to accurately dock the cutter at the correct location (7) or the entire arm would be instrumented with proximately sensors to sense and avoid near collisions (8). These sensors could then be used to update and increase the accuracy of the world model. Ideally, in the future the LDUA will be instrumented with these types of sensors, thus increasing both the safety and the accuracy of the robotic system.

FLEXIBLE DATA ACQUISITION FOR ROBOTIC SYSTEMS

A data acquisition system for use in robotic characterization of underground storage tanks must be capable and robust, as well as easily adaptable to changing requirements. Fundamentally, the system should provide a single point from which data from various end effectors can be acquired, processed, displayed, and stored. An additional goal of the design of such a system is that it should provide an open architecture that easily accommodates new end effectors and adapts to new missions with little disruption to the existing system. Ideally, the system design should allow for further development like the addition of neural networks and artificial intelligence which could provide data analysis concurrent with collection to give the operator guidance concerning tank characterization.

LDUA CONTROL AND DATA ACQUISITION DESIGN

The LDUA system is designed to be run by two operators using workstations located in a remote trailer. One operator uses the graphical interface on the Supervisory Control System (SCS) to operate the arm using path previewing and collision avoidance. The other operator uses the Supervisory Data Acquisition System (SDAS) to operate end effectors and gather data. The two systems are linked by a robot position client and server which provide current arm positions from the subsystem controller to the SDAS. Both the LDUA control and data acquisition systems are designed with an open architecture using the client/server model to achieve maximum flexibility. The designs of both systems are detailed below.

Supervisory Control System

The LDUA control architecture is based on the Generic Intelligent System Controller (GISC) approach developed at SNL (9). The GISC-based design of the SCS is illustrated in Fig. 1. This system is composed of four basic software components: a graphical supervisor (Sancho), a commercial robot simulation package (IGRIPa), a device driver (LDUA Robot Server), and communication interfaces. Both the simulation package and the supervisor are processes which run on a Silicon Graphics, Inc. workstation. The device driver software executes on a separate CPU which is attached to a Versa Module Eurocard (VME) bus.Communication interfaces exist between the supervisor and the simulation package, between the supervisor and the device driver, and between the simulation package and the device driver.

The supervisory program, Sancho, is written in the C programming language. Users interact with Sancho through a graphical user interface (GUI) based on the XForms library for X-windows (10). Sancho drives the simulation package using IGRIP's Command Line Interface (CLI) commands (11). Nettools, a Deneb Robotic's interface based on TCP/IP communications, is used to pass CLI commands and values returned by CLI commands between the supervisor and the simulation package. Sancho commands and interrogates the LDUA Robot Server device driver through a generic set of commands based on the Robot Independent Programming Environment/ Robot Independent Programming Language (RIPE/RIPL) programming paradigm (12). The generic RIPL command set includes commands such as: SetSpeed, SetToolLength, MoveTo, LoadPath, and MoveAlongPath. In addition to issuing commands to the device driver, the supervisor can also respond to asynchronous events that are generated by the LDUA Robot Server. Examples of such subsystem events include the completion of a commanded move and an externally-generated emergency stop.

Communications between the supervisor process and the device driver is accomplished through the GENeralized Interface for Supervisor and Subsystems (GENISAS) (9) communications library. GENISAS provides a convenient way to link commands, queries, and data exchanges to the appropriate robot control functions and data transfer functions within the device driver. GENISAS also provides the mechanism by which asynchronous events (E-stops, move complete, etc.) from the device driver are associated with the appropriate functions within Sancho.

As shown in Fig. 1, two separate instances of the LDUA Robot Server are spawned on the VME CPU. These two servers are referred to as the Real Robot Server and the Simulated Robot Server, respectively. As their names suggest, one server controls and queries the state of the actual robot arm whereas the other server is used for generating simulated motion for the IGRIP simulation package. GENISAS connections exist between Sancho

and the Real Robot Server and between IGRIP and the Simulated Robot Server.

Fig. 1

Both of the robot servers communicate with the LDUA Subsystem Control software developed by Spar Aerospace Ltd, the manufacturer of the LDUA. This communication takes place through a set of library functions referred to as the GISC interface. The Subsystem Controller is responsible for servo control of the robot arm. In addition, it can be placed in state known as "simulation" mode. In simulation mode, the Subsystem Controller accepts motion commands as usual, but, instead of moving the arm, it simulates the move by generating a sequence of joint positions. The Simulated Robot Server only interacts with the Subsystem Controller when simulation mode is enabled, and, likewise, the Real Robot Server interacts with the Subsystem Controller when simulation mode is disabled and motion is enabled.

When an operator issues a motion command to the SCS, the motion is first previewed through IGRIP. During the preview phase, IGRIP uses Deneb's shared library functions to command the Simulated Robot Server which, in turn, issues commands to the Subsystem Controller. Collision checking is done inside of IGRIP. If a collision is detected, the command is terminated. Following the preview phase, the SCS then queries the operator to determine if the motion should be executed by the real robot arm. If the motion is to be repeated by the actual robot, the Sancho application sends commands to the Real Robot Server to initiate, control, and monitor the move. Sancho displays the monitored motion in the IGRIP simulation to provide real time feedback to the operator as the move is executed.

The menu-driven user interface to Sancho overlayed on the IGRIP display is shown in Fig. 2. The interface allows the user to command motion to a selected tag point or along a predetermined path. Scripts for more complex tasks, such as tool-exchange operations, can be executed and modified through the Sancho interface. A tool for generating paths and motions for scanning operations is also available. The user can easily change the view in the IGRIP workcell as well as enter and leave IGRIP user mode.

Fig. 2

Supervisory Data Acquisition System

The Supervisory Data Acquisition System (SDAS) for the Light Duty Utility Arm (LDUA) is a flexible data acquisition and control system. SDAS operates independently of the Supervisory Control System (SCS) and provides services that do not affect the robot controller. SDAS is a general system that allows quick changes in the operator interface and input/ouput (I/O) configuration.

SDAS is composed of two parts: a real-time server and an operator interface. The server portion handles the real-time aspects of data acquisition and control. The operator interface provides a graphical interface for the operator to view data and control non-robotic LDUA subsystems. Both aspects of the SDAS are discussed in more detail below.

The real-time server runs under the VxWorksb operating system on a VME bus architecture, supporting an extensible set of input and output (I/O) boards. The server is a collection of records in a database with each record holding a single real-time value. Records obtain values from hardware inputs (i.e., analog to digital converters (ADC), digital to analog converters (DAC), and digital output devices), operator inputs, or

from calculations. Records are "processed" whenever a new value is received. Whenever a record value is updated, the current time and robot position are stored with the record.

The server schedules records to be processed at regular intervals, in response to events, or simply due to normal flow within the database. When a record is processed, it may optionally schedule additional records for processing. For example, an ADC record may read a voltage in counts, this record would then cause a calculation record to process which converts the ADC counts into engineering units. Two types of records in the database control this flow of additional processing. The first type unconditionally schedules a list of records for processing each time it is processed. The second conditionally schedules a single record for processing based on an input index. The configuration of the server database is specified via a text file each time the server is started. This allows considerable flexibility when changing the system and/or adding additional capability.

The integration and addition of various input and output devices are handled with corresponding device drivers within the data acquisition system. Device driver objects are independent of the SDAS program and are liked and loaded at runtime. The text database configuration file discussed above allows records to be associated with device drivers at runtime. This allows system administrators to add devices to an existing SDAS installation without modifying the SDAS code. The current SDAS contains drivers for support of the standard Industrial Pack mezzanine, Opto-22, and RS-232 and RS-485 serial standards. However, the system is not limited to these I/O devices; new drivers can easily be incorporated into the system.

In the current data acquisition system, client applications to the server include the operator interface and an auxiliary interface for viewing data. Again, the system is not limited; future client applications may include historical log files, commercial databases, or expert systems. These client applications connect to the server over ethernet using TCP/IP socket connections. After establishing a connection, client applications open records for reading and writing on an individual basis. Whenever a record is processed, all clients who have opened the processed record receive the new data along with the time and position stamp. The client/server architecture allows several clients to connect to the server at any given time and have access to the same records. At the same time, a record may be locked by a client, thus allowing single point of control.

The SDAS operator interface is a graphical user interface based on National Instrument's LabVIEWc software. A copy of the highest level SDAS operator interface is shown in Fig. 3. The interface consists of a collection of LabVIEW Virtual Instruments (VIs) which control and monitor LDUA subsystems. A LabVIEW VI is composed of a front panel and an associated wiring diagram. The front panel contains buttons, graphs, charts, menus, and other user interface graphics. The wiring diagram controls the front panel using a visual programming paradigm. The operator interacts only with the front panel. In the example shown, selecting any of the displayed buttons brings up the control and data acquisition front panel for that particular subsystem. SDAS provides a library of sub-VIs which send data to and receive data from the SDAS server. The interactive building of the front panel and visual

programming of the wiring diagram allow the operator interface to be easily maintained and updated to reflect changes in the system.
Fig. 3
For the first deployment of the LDUA system at Hanford, SDAS controls the pan and tilt camera systems and video system. This is accomplished for the camera systems by using a single Relay 16 Industrial Pack by Systran Corporation and an RS-232 serial connection and for the video system (composed of a video switcher, two VCRs, two video titlers, and a video printer) through multiple serial connections. In addition to the video and camera systems, future deployments of the LDUA will include the collection of data from various sensor end effectors including an SNL developed "minilab" multi-sensor end effector (13).

FUTURE PLANS AND DEVELOPMENTS

The first deployment of the LDUA in a radioactive waste tank will be at the Hanford site in 1996. The Supervisory Data Acquisition System described above will be used to control and store video images from various end effector and overview cameras (4). The Supervisory Control System described in this paper will be implemented on later deployments of the LDUA arm. At that time, the graphical commands, path previewing, and collision checking capabilities within the SCS will be used to operate the LDUA in a telerobotic mode versus pure teleoperation. Automated path planning algorithms will also be implemented to allow the operator to easily perform a complete survey of a surface with video, camera, and/or sensor end effectors.

Future deployments of the LDUA will use end effectors that must operate in contact with or in close proximity to the surface of the waste or tank structures (6). To support this mode of operation, the SCS will be extended to read position and force feedback from sensors on the LDUA end effectors and modify the motion of the arm for surface following, compliant motion, and docking-type operations (7). The ability of the arm and end effectors to be controlled from task-level automatic sequencing will be added to the SCS in later deployments. Further developments include advanced trajectory planning algorithms that calculate collision-free paths in congested work spaces.

SUMMARY

This paper describes the development of both a Supervisory Control System (SCS) and a Supervisory Data Acquisition System (SDAS) that were designed specifically for use on robotic systems in waste storage tanks. The two systems will be implemented on the Light Duty Utility Arm (LDUA) system, an integrated robotic deployment system which will perform inspection, surveillance,waste analysis, and small-scale retrieval tasks in underground storage tanks. SDAS will be implemented on the LDUA in its first deployment at the Hanford site in 1996. SCS will be installed and used to control the arm in later deployments.

ACKNOWLEDGMENTS

The Light Duty Utility Arm System is being developed under the sponsorship of the DOE Office of Science and Technology's Tanks Focus Area (EM-50). The LDUA is being implemented by a consortium of DOE sites, national laboratories, and commercial partners. These include: Westinghouse Hanford Company, Sandia National Laboratories, Oak Ridge National Laboratories, Pacific Northwest Laboratory, the Savannah River Site, the Idaho National Engineering Laboratory, and Spar Aerospace, Ltd. The work performed at Sandia National Laboratories was supported by the

United States Department of Energy under contract number DE-AC04-76DP00789.

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Session 57 -- PANEL - WASTE MANAGEMENT CHARGEBACK SYSTEMS IN THE DOE COMPLEX

Co-chairs: R. Piscitella, INEL/LITCO;
James Van Vliet, INEL

57-1

CHARGEBACK SYSTEMS IN THE DOE COMPLEX

The workshop conducted during this session addressed Chargeback Systems in the Department of Energy (DOE) Complex. The purpose of this workshop was to present and discuss successes and problems that are currently being encountered with implementation of chargeback systems for waste

management activities in the DOE Complex. The workshop was introduced by the two chairmen of the session: J. A. Van Vliet, Director of Waste Operations for Lockheed Idaho Technologies Company (LITCO) at the Idaho National Engineering Laboratory (INEL) and R. R. Piscitella, Low-Level Waste Type Manager also with LITCO at the INEL. As part of the introduction, current reviews by the National Academy of Science, the Environmental Management Advisory Board, and other reviewing organizations that suggest chargeback as a funding mechanism were discussed. A definition of chargeback was also presented as follows:
A charging system where the waste generator is either partially or totally billed for waste management activities that are directly related to the waste that the generator produces.
Following this introduction, short presentations were made. Questions from the audience were answered during the presentations. Abstracts of these presentations are given below in the order presented.
Idaho National Engineering Laboratory, Waste Operations Chargeback System -- R. R. Piscitella, Telephone Number: (208) 526-1137
Chargeback systems are being considered for implementation on the INEL for two primary reasons: To give the waste generator a strong economic incentive to reduce the amount and/or hazard of the waste generated, and to give INEL treatment, storage and disposal facilities a strong incentive to be competitive with comparable private business and other DOE laboratories. The current waste management system on the INEL does not support either of the objectives listed above.
The proposed system would continue to directly fund Waste Management treatment, storage and disposal (TSD) facilities for activities such as legacy waste and post-closure activities. However, waste generators would pay for waste management activities that directly relate to TSD activities for the wastes they produce. A chargeback pilot programthat addresses only low-level waste (LLW) processing and disposal is being completed in FY-96. This pilot program will modify waste tracking and accounting systems to allow "simulated" bills to be sent to LLW generators. The pilot program will be evaluated at the end of this fiscal year and, based on the results, future chargeback activities will be determined.
Sandia National Laboratory, Operational P2 Chargeback and New Chargeback System Development -- S. A Schrader, Telephone Number: (505) 848-0381
To support its Waste Minimization/Pollution Prevention Program, Sandia National Laboratories adds a surcharge to each unit of hazardous waste that is managed by the EM-30 Waste Management Program. While providing funding for Waste Minimization/Pollution Prevention activities, this also provides a financial incentive for organizations to reduce waste generation. It is not a full cost recovery system, but exemplifies a method of recovering costs from specific customers. The system relies on waste collection data to determine charges to customer organizations. Cost allocation is managed using a "Service Center" mechanism provided by Sandia's Corporate Financial Information System.
This year, Sandia's Sample Management Office (SMO) is implementing a full cost recovery system using the Service Center mechanism. Estimated budgets are used to predetermine cost allocations, which are adjusted each quarter to assure equitable distribution of actual costs among SMO customer organizations.
USDOE Albuquerque, Reorganized (Re-Engineered) Waste Management -- J. Orban, Telephone Number: (505) 845-4421

Results of an independent technical review of Lawrence Livermore National Laboratory, Los Alamos National Laboratory and Sandia National Laboratory waste minimization and management programs pointed out the fact that program drivers are fundamentally different when compared to the private sector. The only focus of a commercial waste minimization and management program is compliant, cost-effective management of the minimum waste. In contrast, DOE and the Laboratories treat waste minimization and managementas the commercial equivalent of a "profit center" rather that a support service. The primary program focus is on money management, not waste cost management. There is no Laboratory equivalent to the commercial profit "bottom Line" incentive for controlling waste streams, volumes, and program costs.

A DOE working group was established to take the result of this technical review and define alternatives for re-engineering DOE Waste Management. DOE Laboratory activities were divided into a Commercial Portion and Investment/Legacy Portion. The Commercial Portion would be a full cost recovery service center and would cover activities associated with treatment, storage and disposal of currently-generated waste. The Investment/Legacy portion would cover legacy storage, legacy workoffs, etc., and would be individually estimated as separate projects. Current implementation plans call for restructuring selected DOE sites into Commercial and Investment/Legacy in a pilot program in FY-97 with "real" billing occurring in FY-98. Other DOE sites would follow with a one- year delay.

Pacific Northwest National Laboratory, New Chargeback System Piloting and Implementation -- K. L. Peterson, Telephone Number: (509) 372-4540

The DOE-EM funded Pacific Northwest National Laboratory Inventory and Waste Management Chargeback design project was discussed. This project is coordinated through the PNNL Waste Management Division and is currently looking at a lab-wide chargeback system that would include: 1) Waste Management Chargeback, 2) Pollution Prevention Investment, 3) Inventory Control, and 4) Effluent Monitoring Costs. Chargeback costs would be divided into fixed costs that would be funded through the Activity Data Sheet (ADS) process and variable costs that would be funded by the waste generator. Implementation issues include: generator planning for FY-97, year-end closeout carryover, accrual of costs, and notification of generators.

Current schedule for implementation includes development of the system this fiscal year, with a waste generator chargeback system expected to be implemented in FY 1997.

Chem-Nuclear Systems, Inc., Chargeback in the Private Sector,
G.J. Antonucci, Telephone Number: (803) 758-1807

Chem-Nuclear Systems, Inc., (CNSI) operates the commercial LLW disposal facility at Barnwell, SC. CNSI conducted a study of LLW disposal at the INEL from the perspective of operating like a private business. Steps that would be followed in the establishment and operations of "running INEL LLW disposal like a private business" were discussed. These steps included: 1) develop and agree upon the real costs for LLW disposal operations, 2) make generators aware of the chargeback system and get their buy-in to participate, 3) develop waste forecasts, 4) establish rates and surcharges that apply to forecast waste streams, 5) treat, store, and dispose of waste at established rates, and 6) if disposed rates exceed forecasts by greater than 10%, either the generator stores the waste or a 30% increase in fee is applied.

Session 58 -- WASTE CHARACTERIZATION AND SPECIFICATION TO MEET REGULATORY REQUIREMENTS AND WASTE ACCEPTANCE CRITERIA
Co-chairs: Michael B. Hughes, WSRC
58-1

JUST-IN-TIME CHARACTERIZATION AND CERTIFICATION OF DOE-GENERATED WASTES

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ABSTRACT

The goal of just-in-time characterization and certification, which is based on the just-in-time manufacturing process, is to streamline the certification process by eliminating redundant layers of oversight and establishing pro-active waste management controls. Just-in-time characterization and certification would rely on a waste management system in which wastes are characterized at the point of generation, precertified as they are generated (i.e., without iterative inspections and tests subsequent to generation and storage), and certified at the point of shipment, ideally the loading dock of the building from which the wastes are generated. Waste storage would be limited to accumulating containers for treatment, if necessary, or for cost-efficient offsite shipment only.

Just-in-time characterization and certification would be accomplished by establishing 1) a comprehensive waste management quality assurance (QA) program consisting of infrastructure controls and on-going verification measures that provide the overall confidence in data validity and waste package integrity to allow certification upon demand and 2) three essential functions of characterization at the point of generation, "precertification" (including necessary inspections, data validation, nonconformance control, and document compilation and review), and final waste package certification (i.e., a document package review ensuring that all applicable regulatory and disposal facility requirements are satisfied).

Just-in-time characterization and certification is most applicable to wastes generated by decontamination and decommissioning activities. However, implementation of the process can be phased, using other waste categories (e.g., environmental restoration waste, rejected waste, legacy waste, newly generated waste) to introduce the concept at the facility and for use in employee training.

INTRODUCTION

Transportation and disposal of wastes generated by Department of Energy (DOE) activities, including weapons production and decontamination and decommissioning (D&D) of facilities,require that wastes be certified as complying with various regulations and requirements. These certification requirements are typically summarized by disposal sites under specific waste acceptance criteria (WAC). Although a large volume of wastes has been generated by past activities and are presently in storage awaiting disposal, a significant volume of wastes, particularly from D&D projects,

has not yet been generated. To prepare wastes for disposal in an efficient manner, it is suggested that a program of just-in-time characterization and certification be adopted.

The concept of just-in-time characterization and certification is based on established just-in-time manufacturing practices, in which goods are produced as needed to fill customers' orders; parts and product inventories, and the storage needs for these inventories, are minimized to reduce costs. The linchpin of just-in-time manufacturing is a well-developed infrastructure that allows quick and dependable response to customer needs. The key to successful just-in-time characterization and certification is confidence derived from an effective characterization and waste management system, and the resultant ability to certify wastes as they are processed (i.e., without iterative inspections, tests, and re-evaluations subsequent to generation and storage).

The following description of the just-in-time characterization and certification concept is based on an evaluation of the waste certification program at the Rocky Flats Environmental Technology Site (RFETS) and experience at other DOE facilities, including the Los Alamos National Laboratory (LANL).

REGULATORY DRIVERS AND THE NEED FOR JUST-IN-TIME CHARACTERIZATION AND CERTIFICATION

Requirements for waste certification are derived from DOE Order 5820.2A, 10 CFR Part 71, 40 CFR Part 191, and 49 CFR Part 172 (1,2,3,4). These specifications require waste generators to assure waste treatment, storage, and disposal facilities that the regulatory requirements for the waste are met. This assurance includes the implementation of the controls and measures necessary to ensure compliance with the disposal facility WAC. Certification requirements are also present in the WAC for various disposal facilities, as follows:

- Envirocare of Utah, Material Acceptance Process Manual (5)
- Hanford Site, Hanford Site Solid Waste Acceptance Criteria (6)
- Nevada Test Site, Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements (7)
- Scientific Ecology Group (SEG), Waste Acceptance Criteria (8)
- Waste Isolation Pilot Plant, Waste Acceptance Criteria for the Waste Isolation Pilot Plant (9)

The specific certification requirements of DOE, U.S. Environmental Protection Agency (EPA), U.S.Department of Transportation (DOT), and different treatment, storage, and disposal facilities can result in multiple layers of oversight (Fig. 1). In turn, these multiple layers of oversight may increase the amount of bureaucracy, time and resources (i.e., personnel and budget) needed to get a waste shipment offsite, which reduces efficiency and increases costs. The goal of just-in-time characterization and certification is to streamline the waste characterization and certification processes. This can be accomplished by eliminating redundant levels of oversight and establishing pro-active controls rather than "final inspection quality" to achieve WAC compliance. The degree to which extended storage of waste can be reduced or eliminated is also fundamental to streamlining the waste disposal process.

Fig. 1

JUST-IN-TIME CONCEPT

The basic concepts inherent to the just-in-time operational management philosophy have direct application at DOE facilities. The goals of just-

in-time manufacturing are to reduce costs, increase efficiency, and increase the company's responsiveness to its customers. Eliminating overhead in the form of parts inventories, product inventories, and storage requirements associated with these inventories, reduces costs; in addition, process bottlenecks can be identified and corrected. Manufacturing products at the rate of demand improves the efficiency of the process. Because the products are manufactured at the same rate as demand, the time required for responding to shifting customer demands can be reduced.

These concepts can be applied to waste characterization and certification at DOE facilities. Rather than storing waste until it is ready to be shipped and incurring costs (e.g., inspections, utilities, building floor space) associated with this storage, the just-in-time process would ship wastes immediately after generation. In addition to reducing or eliminating storage costs, this process would enable DOE facilities to meet changing WAC as the wastes are generated, not as the packages are pulled from storage, which results in higher costs and delays in shipping because of the rework of the packages to meet the new acceptance criteria.

JUST-IN-TIME CHARACTERIZATION AND CERTIFICATION

Just-in-time characterization and certification at DOE facilities would allow wastes to be shipped from the generation location and eliminate the need for long-term storage. Wastes would be stored only for the purpose of accumulating enough waste packages to make shipment economically viable. Both characterization and certification would be required at or near the point of generation.

For certification on demand (e.g., as needed) to be a viable option, the entire process must be supported by a well-planned, comprehensive, and rigorous waste management QA program. Such a program provides the procedural infrastructure necessary to achieve confidence that all aspects of the waste management system are implemented in an effective manner. This procedural infrastructure includes the basic elements of the existing QA standards (e.g., NQA-1, DOE 5700.6c, ASQC-E4); therefore, the infrastructure provides for operations control through personnel qualification, procedural control of activities, nonconformance control, and planning and provides assurance of program effectiveness through systematic verification activities (i.e., inspection, surveillance, audit, and corrective action).

With such a program in place, demonstration of program effectiveness for external oversight or regulatory agencies becomes a relatively simple exercise. Also, confidence in the adequacy of waste characterization, such as characterization data and package integrity, is sufficient to allow certification based on existing data and without the intense scrutiny of individual packages commonly associated with programs plagued by suspect data, continuous external inspection, and audit findings. With such a program in place, a facility can concentrate on strengthening and streamlining the important functions of waste characterization, data validation, and documentation review that allow just-in-time certification. Such a program is depicted in Fig. 2.

Fig. 2

For characterization to support the just-in-time concept, the waste generated by an environmental restoration (ER) or D&D project must be characterized (i.e., sampled and analyzed as necessary, with complete analytical results) before the wastes are placed into containers. This

requires the characterization personnel to participate during the planning stages of a project. As the responsible manager and engineers are planning the process for performing the project, the characterization personnel would be reviewing the planning documents, designing the sampling plan, and performing initial sampling of materials that would be generated as wastes during the project. Initial sampling that is conducted during the planning stages of the project could also be applicable to the validation of process knowledge for future ER and D&D activities. The characterization plan should prescribe measures to be taken when anomalous results or unknown or unanticipated waste items are encountered. During the actual performance of the project, waste characterization personnel must be available, on an on-call basis, to resolve such issues so that they are not pushed aside to become uncharacterized backlog wastes that may cast aspersions on the validity of overall characterization data.

Wastes generated by the project would be certified at the point of shipment, similar to what currently occurs at many facilities. However, the wastes would be precertified at the point of generation. Precertification would entail verification of characterization and data validation completeness, waste package inspection, verification that characterization data is traceable to waste packages, and compilation of the documentation necessary to support certification. Precertification would be performed before containers are sealed and while the containers remain the responsibility of the generator, which allows any problems to be corrected while the waste is still easily accessible and while the generator can be held accountable for, and learn from, such problems. Final certification would primarily be a review to verify that the waste package meets disposal facility and transportation requirements, as opposed to the detailed scrutiny of paperwork common to facilities that cannot rely on the viability of the overall waste management system to produce a quality product.

PERSONNEL REQUIREMENTS

The adoption of just-in-time characterization and certification may require the realignment and refocus of waste management organizations. In addition to waste generation, just-in-time characterization and certification would require emphasis on three functions: characterization at the point of generation, waste precertification, and waste shipment certification.

Two requirements must be met for successful implementation of just-in-time characterization and certification. First, waste characterization and certification activities must be autonomous in terms of organizational structure and funding. The responsible organizations must carry the authority to resolve problems identified during the waste generation, packaging, and certification process.

Second, all personnel responsible for the characterization and certification process must be qualified in their disciplines and must be knowledgeable of the overall project. A fundamental precept of the International Atomic Energy Agency (IAEA) is applicable: personnel auditing, inspecting, or approving work must be at least as qualified as the personnel performing the work being overseen. A root cause for many problems in existing waste management organizations is deficiencies in personnel training and qualification. Such problems must be eliminated for just-in-time characterization and certification (or any waste management system) to function efficiently.

Waste Characterization

To provide for just-in-time characterization, the personnel responsible for this characterization must be available to the personnel generating and precertifying the waste. These characterization personnel must also be involved in the planning stages of the project. Waste characterization personnel would prepare characterization plans and procedures, implement and oversee sampling and analysis, ensure proper data validation, and prepare final characterization data packages, including regulatory justifications. The personnel would attest to the proper characterization of the waste being placed into the container and be available to provide guidance when sampling unknown or suspicious wastes as they are encountered by the personnel performing the waste generating activity. Guidance on the proper packaging of individual or unique wastes that are being generated would also be available from waste characterization personnel.

Waste Precertification

Under just-in-time certification, a detailed review of the container documentation would need to be performed during the process of waste generation. Although the waste precertification personnel's duties would be similar to that of a waste inspector performing in-process inspection of a waste package, the level of inspection detail is greater. This detailed review would be performed by waste precertification. The waste precertification function would also include verification that waste characterization data are complete and satisfactory, and compilation of all documentation required for the final certification.

Waste precertification would control access to the waste packages; ensure the proper setup of the container for the waste stream being generated (e.g., number and type of liners, presence of a carbon filter); ensure that generators placing waste into the package have current qualifications, and ensure that the wastes placed in the package are of the same, or compatible, waste types. In position and responsibility, the waste precertifier would be analogous to the package certifier described in NVO-325, Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements (7).

Waste Shipment Certification

When D&D and ER activities are undertaken at DOE facilities, the volume of wastes generated will increase dramatically over current generation rates. Consequently, the certification function will become important to managing waste and meeting overall project milestones. It is probable that certification staffing levels will have to increase significantly at many facilities.

In the just-in-time certification concept, final certification becomes a matter of ensuring that waste packages meet disposal facility and transportation requirements rather than a detailed scrutiny of all documentation on a package-by-package basis. The waste management QA program would control the accuracy and completeness of the documentation. The precertification function would control oversight and review. Thus, the waste shipment certification official would be freed from the checking function and can devote efforts to resolving larger issues. The waste shipment certification official would also have better communication with disposal facility officials and regulators. Consequently, personnel in this capacity will need to be of a senior level and must be knowledgeable of all disciplines involved in the waste generation, treatment, and disposal process. If the facility wishes to be

successful in processing the volumes of waste anticipated from the D&D and cleanup of DOE facilities, it will have to recognize the importance of this function, and staff and support it accordingly.

IMPLEMENTATION

Implementation of the just-in-time concept into a manufacturing operation typically involves five steps. Similarly, these just-in-time concepts can be adapted to waste management, characterization, and certification, as follows:

Cleaning up the Operations. Factors that prevent the certification and shipment of waste (e.g., procedural roadblocks, duplicative or competing activities) can be identified and eliminated or reduced by performing an evaluation of the existing process for waste generation and certification and a root cause analysis of problems diagnosed during this process.

Establishing a Comprehensive Waste Management QA Program. If the facility does not presently have an effective QA program, which is essential to the viability of the just-in-time concept, establishing such a program is essential, but is not a trivial undertaking. The process can be briefly described as evaluating the facility to determine the activities that are important to safety and waste quality, and then ensuring that all activities are proceduralized to ensure that they are performed adequately and that auditable documentation to demonstrate satisfactory performance is provided. Development of such a program must be the responsibility of senior, experienced personnel with the authority to institute necessary changes.

Effecting a Company-wide Culture Change. Because DOE facilities are in a period of transition from weapons manufacturing to ER and D&D, a facility-wide culture change might be necessary, particularly if a comprehensive waste management QA program is not currently in place. This would be an ideal opportunity to introduce the concept of just-in-time waste characterization and certification.

It is worth noting that cross-cutting cultural changes have been successfully accomplished in many troubled industries, both commercial and nuclear, over the last few decades. This process has been successful only where management was fully committed and where the responsible individuals were given full authority to make necessary changes. However, in those facilities that resisted such change, the process has in many cases been less than fruitful.

Preparing Logistics for Just-in-time. Procedures and processes for the certification of waste should be reviewed so that unnecessary or redundant requirements are eliminated. It may also be necessary to upgrade existing facilities that are used to ship the large amounts of waste expected to result from ER and D&D activities. Areas requiring logistical upgrades should be identified after a more detailed analysis of the facilities has been done.

Implementing the Program. Another aspect of preparing for just-in-time characterization and certification implementation is training personnel to the revised procedures and processes. Characterization, certification, and their associated quality assurance requirements should be pushed as far back in the waste generation process as possible, ideally to the point of generation.

Reviewing and Monitoring the System after Implementation. Specific measurable performance objectives should be established for each department or program that will be contributing to just-in-time characterization and certification. Ongoing verification of performance

by an autonomous organization will be necessary to establish the validity of the overall system and to demonstrate system viability to external oversight personnel.

It is recommended that any just-in-time certification process first be established as a pilot program. The intent of this program would be to introduce and develop the concept of just-in-time certification and to provide training for the required personnel (i.e., waste shipment certifiers, waste precertifiers, and waste characterizers). This training would be accomplished by rotating the personnel through the pilot program and then placing them into other waste generating buildings, at which time just-in-time characterization and certification would be implemented. By having such a centralized, hands-on training process, consistency in training could be achieved, personnel motivation enhanced, and lines of communication established. To ensure the certifiability of wastes generated during this pilot program, it is recommended that the waste packages in-process be inspected and certified using the existing certification system on a temporary basis, in parallel to the certification being performed under the just-in-time system.

To further the transition from the current waste generation and certification process to the just-in-time process, and to simplify it, present waste packaging procedures should be reviewed to identify those requirements absolutely necessary for the certification of wastes to the disposal facility WAC. Requirements that are not absolutely required should be eliminated. The procedures themselves should be simplified, combined, and rewritten so that one procedure would provide sufficient information for the proper packaging of waste.

As these procedures are revised, and as the just-in-time process is developed, it is also recommended that the responsibilities of the organizations involved in waste characterization and certification be reviewed and, as necessary, combined so that only those organizations absolutely necessary to the certification process are involved. Responsibilities of the retained organizations, and the lines of communication among those organizations, should also be defined as completely as possible to remove any ambiguity that may confuse or slow the process.

Finally, it should be recognized that no procedure or program should ever be implemented without clear specification of how existing waste will be viewed and addressed under the change and regulator and disposal facility endorsement, where required or appropriate.

SUMMARY

Just-in-time manufacturing and supply concepts work for many commercial enterprises. These concepts work where business programs and procedures are so well organized and reliable that the company can respond at a moments notice to changing customer needs. At DOE facilities, the customer is not only DOE (and, consequently, the public), but also project commitments and schedules. Just-in-time characterization and certification programs can help meet the accelerated disposal schedules anticipated for future D&D and ER activities. Our customers deserve the well organized and reliable waste management program that makes just-in-time concepts possible now.

REFERENCES

1. DOE Order 5820.2A, Radioactive Waste Management, U.S. Department of Energy.

2. CODE OF FEDERAL REGULATIONS, Title 10, Part 71, "Packaging and Transportation of Radioactive Material."
3. CODE OF FEDERAL REGULATIONS, Title 40, Part 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Wastes."
4. CODE OF FEDERAL REGULATIONS, Title 49, Part 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements."
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58-2

DEVELOPMENT OF A LOW-LEVEL RADIOACTIVE WASTE ACCEPTANCE PROGRAM AT THE NEVADA TEST SITE

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ABSTRACT

The Nevada Test Site (NTS) accepts low-level radioactive waste from generators across the United States. The program utilizes multi-organizational concepts in the contracting of scientists and specialists. Personnel are involved with all facets of the program including the development of the Waste Acceptance Criteria, review of generator documentation, and assessment of generator programs. The development of the NTS low-level radioactive waste acceptance program has been through an evolution of regulatory drivers, policy, and expertise, resulting in a safe and effective waste management and disposal strategy.

HISTORY/OVERVIEW

The Nevada Test Site (NTS) has historically been the nuclear weapons test facility for the U.S. Department of Energy (DOE) and its previous agencies such as the U. S. Atomic Energy Commission. The NTS is located approximately 65 miles northwest of Las Vegas, Nevada, and encompasses 1,350 square miles. The NTS also serves as a major disposal facility for low-level radioactive waste (LLW) generated by DOE and other non-DOE installations throughout the United States.

Once the test ban treaty prohibiting the testing of atmospheric nuclear weapons was ratified, awareness for the need of clean-up and remediation of the atmospheric test debris and areas of testing at the NTS began to arise. As the process began, the research for disposal methods of low-level radioactive waste were initiated. With concerns for the safe and final disposal of LLW came the realization that criteria which address

geology, hydrology, and the climatic concerns of the disposal area must be established. At the same time, it was realized that consideration must also be given to safety, shipment methods, identification of waste content, Federal and state regulations, and radiological parameters. As preparations began to select and utilize an existing crater from a previous underground test, a plan was developed to address the subject criteria for the disposal of the LLW. Considered in this plan was the arid climate, deep aquifer,alluvial soil, and isolation from the public of the selected disposal site. This plan evolved into the "Operational Radioactive Defense Waste Management Plan for the Nevada Test Site," NVO-185.

In 1978, as waste disposal activities began, the development of criteria paralleled the disposal operations. Lessons were learned and the activity evolved. Researchers began to better understand the ideal attributes of the area and continued to develop techniques for safe and permanent disposal options for DOE's overburdening amounts of LLW. In January 1985, NVO-185 was revised for the fourth time to implement the technically evolving criteria for both on-site and off-site generators.

With the onset of increased Federal regulations such as the Resource Conservation and Recovery Act (RCRA) and the inclusive Land Disposal Restrictions (LDR), the DOE re-evaluated its program to manage the growing volumes of wastes being generated and stored at its various sites across the country. In September of 1988, DOE developed and implemented DOE Order 5820.2A, "Radioactive Waste Management." This order provided requirements and guidance to the operations and field offices for the management and disposal of their wastes.

Since the NTS was familiar with the regulatory issues and in anticipation of the new DOE Order being issued, the NTS began to utilize its NVO-185 Plan to develop waste acceptance criteria (WAC) that would incorporate additional lessons learned, the new and evolving RCRA requirements, and the anticipated DOE Order 5820.2A. In October 1988, NVO-325, "Nevada Test Site Defense Waste Acceptance Criteria, Certification, and Transfer Requirements," was formally issued, replacing NVO-185. The NTS was now disposing of waste in well-designed subsurface disposal trenches as well as in bulk crater sites. Many strides had been made in the utilization of disposal packaging, disposal techniques, waste characterization, and shipment mechanisms. This information was included in the new WAC.

As off-site DOE generators were attracted to the potential for shipping their LLW to the NTS and began to submit applications as required by the new criteria, the DOE Nevada Operations Office (DOE/NV) realized that professionals needed to be assigned to review these applications against the WAC and verify through assessment of the generator's site that the LLW met the NTS WAC.

Through the appointment of a DOE/NV Program Manager and the acquisition of additional needed technical expertise from the disposal site Management and Operating Contractor (M&O) and other contractors, DOE/NV developed a team of specialists that performed the needed functions. As the generator numbers and volumes increased, the NTS Program's team of specialists and scientists gained knowledge and experience. With the lessons learned and ever-evolving Federal and state regulations (including a very concerned state of Nevada), it was realized that the WAC needed to be revised to address these issues and to assure that the NTS, which is currently not permitted to dispose of LLW with RCRA regulated hazardous components (low-level mixed waste [LLMW]) was

complying with all applicable waste regulations. In June 1992, NVO-325, Revision 1, was issued.

This document describes the Acceptance Program responsible for developing, maintaining, and implementing the WAC. In addition it summarizes the current WAC being used by DOE/NV for the disposal of LLW at the NTS Area 3 and Area 5 Radioactive Waste Management Sites (RWMSs). Revision 2 to NVO-325 is currently being developed and is anticipated to be released in late 1996.

NTS WASTE DISPOSAL VOLUMES-CURRENT STATUS

In the first seventeen years of operation, the NTS Areas 3 and 5 RWMSs accepted and disposed approximately 17 million cubic feet of LLW. The NTS safely disposed of 752,644 cubic feet of LLW in Fiscal Year 1994 and 884,614 cubic feet in Fiscal Year 1995.

The NTS disposed of 85 percent of the DOE complex off-site LLW in 1993 and 1994. NTS disposal forecasts estimate receiving 1.1 million cubic feet of LLW in 1996 and 1 million cubic feet in 1997.

The NTS is the nation's largest volume LLW disposal site (commercial or DOE) and has fifteen approved on-site and off-site generating facilities. These facilities are listed in Table I. Currently, 24 LLW generators have received waste designation for potential use of the NTS disposal facilities. The NTS has two on-site LLW generators (contractors).

Table I

RADIOACTIVE WASTE ACCEPTANCE PROGRAM (RWAP)

The NTS disposal program, through the guidance of a single DOE Program Manager and the involvement of a team of M&O contractor personnel, continued to evolve. The synergy from active team participation in the program's development was quickly realized by DOE and the need for additional team expertise was recognized. As the program and generators evolved, a second Program Manager was assigned. These Program Managers developed the team approach and began to utilize expertise from the M&O and other contractor specialists. A unique team approach was implemented which allows active input and ideas, and encourages participation.

The RWAP program has developed a mission statement, instructions that provide guidelines for its operations, and uses on the job training with discipline specific training for its members. These DOE/NV instructions further detail the process followed in the RWAP program. These instructions are controlled and issued to all RWAP support organizations. Training is required in the instructions, as well as other areas, as applicable, to each team member's discipline. Development of the instructions was performed by the RWAP team and is an example of this unique team approach. Quality performance and a customer oriented approach are emphasized.

The RWAP team of active DOE and varied contractor personnel is encouraged to work and motivate as a single team unit and not as individual personnel or organizations. This unique approach has been very successful and currently allows multi-contractor personnel direct input into all operations and activities.

NVO-325, Revision 1, provides DOE/NV with the necessary waste management plans and policy guidelines regarding waste acceptance and transfer. The RWAP team applies these plans and guidelines through basic RWAP program elements.

The Program. As discussed above, the program utilizes a multi-contractor organization team of specialists under the direction of four DOE Program Managers who operate to finite, documented approaches and guidelines. The

team consists of sub-teams that perform audits and surveillances, site visits to assist generators, and technical issues such as criteria revision and problem solving. Often, waste generator organizations are invited to participate on the sub-teams, including audits of waste generator programs other than their own.

The Generator Approval Process. The process begins with a waste designation by DOE Headquarters. The generator then contacts DOE/NV WMD and requests to apply. If accepted, a current copy of NVO-325 is issued to the generator. In concert with the development of the application, the generator also develops and implements a waste certification program and process which includes the development of characterization and certification plans and procedures. If the generator is having difficulty with plan development or other portions of the program, RWAP personnel(the actual DOE/NV auditors) may be sent to assist the generator in understanding the program requirements. RWAP personnel will not set up the program but rather provide guidance and conduct a "mock" audit. These mock audits can be beneficial as they allow a generator to understand what the audit team will be looking for and identify deficiencies in the program without the need for formal corrective action and documentation. The generator then characterizes their waste and completes their waste application. Once the generator has completed the application to ship waste to the NTS, it is submitted to DOE/NV RWAP for review and comment. Upon completion of the application review and comment processes, DOE/NV schedules and conducts a programmatic and performance-based audit. Based on any deficiencies identified in the audit, the generator completes the corrective actions, and a follow-up surveillance is conducted to verify their accuracy and completion. Upon satisfactory closing of the corrective actions, the Manager of DOE/NV approves the generator for the waste streams reviewed, based on a recommendation from the responsible RWAP Program Manager.

Generator Facility Audits The RWAP audit teams evaluate all aspects of the LLW generator's waste management program including areas such as characterization, procurement, training, document control, traceability, and certification. The RWAP audit methods include checklists, document reviews, personnel interviews (large emphasis on field personnel), and visual evaluations of field activities such as the packaging of the waste. Typical teams consist of three areas. These areas are quality assurance, characterization, and traceability. Subject matter experts are sometimes used as needed. The quality assurance team utilizes one or two auditors and the Lead Auditor is always one of these individuals. Characterization and traceability can be comprised of one to four individuals each. The teams work independently of each other during the daily audit activities and then meet in a team meeting at the end of each day to compare conclusions and discuss discrepancies.

Generators are given as much guidance as possible; however, the human element of all programs result in mistakes, errors, and misunderstanding. Therefore, follow-up evaluations (surveillances) are conducted to verify that all deficiencies resulting from the audit are corrected, or that corrective action is being adequately implemented, before waste is approved for shipment to the NTS.

In the earlier years of the program, only one team of core audit personnel existed. Because of the high demand for audits and surveillances and the rigorous travel required (in some cases up to seven weeks in a row), as well as an increase in the volume of waste streams

and number of generating facilities, concerns for the team's ability to maintain performance at a acceptable technical level (individual burn-out) became a serious concern. In 1992, the development of two "core" teams, with additional matrix personnel that specialize in the various disciplines, was initiated. The matrix personnel are utilized as dictated by work load and schedules. By March 1995 two additional DOE Program Managers had been added for a total of four and the two RWAP audit and surveillance teams traveled to generator facilities approximately every other week of the calendar year. This doubled the amount of facilities that could be given RWAP attention at the same time and reduced the concern for "burn-out." The 1996 schedule has been developed to schedule audit trips for every third week of the calendar year based on available personnel and expertise.

NVO-325, NEVADA TEST SITE DEFENSE WASTE ACCEPTANCE CRITERIA, CERTIFICATION, AND TRANSFER REQUIREMENTS

NVO-325 provides waste acceptance guidelines for on-site and off-site generators for items such as characterization, certification, acquisition of analytical laboratories, and quality assurance activities for the RWAP. The criteria have evolved into comprehensive quantitative and qualitative guidelines for DOE/NV to provide oversight of waste generator compliance to the waste acceptance requirements through an application review and audit processes.

Although most DOE operations for the treatment, storage, and disposal of LLW are performed by contractors; official contact for waste designations, applications to ship waste, and final disposition of approvals and issues is between DOE/NV and each facility's DOE oversight office. All on- and off-site LLW generators are subject to the NVO-325 approval process. NVO-325 does not currently apply to the storage of on-site LLW, mixed waste, or transuranic (TRU) waste; all other criteria for disposal do apply.

THE WASTE ACCEPTANCE CRITERIA (WAC) - NVO-325

NVO-325 addresses the WAC and requires the LLW generator's application provide sufficient detail, as requested in Sections 3.0 through 8.0. NVO-325 does provide a formatting example to stipulate the requirement for the presentation of the LLW generating site's detailed data reflecting conformance of their waste to the WAC in their application. The following is a summary of the pertinent sections from NVO -325 (Rev.1):

Introduction (1.0) This section is offered to the generator to discuss the RWAP and WAC Purpose, Scope, Policy, implementation plan, and responsibilities.

Application Approval Process (2.0) Section 2.0 explains the defense designation requirements from DOE Headquarters, requirements for the generator's application format, the generator approval process (which includes the application review and comment), audit and surveillance process, and the annual program review and approval.

Generator Information and Application Requirements (3.0 - 9.0) Beginning with the following sections of NVO-325, the generator must address the required criteria in their application. By following the guidance in these sections, the generator should be able to develop an application which can be easily evaluated by the RWAP reviewers. These sections contain the minimum criteria for:

- application signatures and approvals,
- radioactive and hazardous waste characterization,

waste stream information which includes specific criteria for waste types, prohibited materials, parameters, shipping, packaging, and radiological requirements;

waste certification program development,

exemption requests,

direction on procedures and documentation, and waste transfer requirements.

Quality Assurance Requirements for Waste Certification Programs (NVO-325, Appendix C) The waste generator is required to define a management system to assure the quality of the desired results for waste certification activities. Appendix C provides specific interpretation of certain NQA-1 requirements for application to LLW certification programs. This Appendix follows the 18 criteria in ANSI/ASME, NQA-1 and adds a 19th. These quality criteria are identified as "Basic Requirements" and are listed in Table II.

Table II

CONCLUSION

The development of the NTS low-level WAC has been the result of practical experience, the evolution of regulatory drivers;political and scientific concerns from the State of Nevada and the local community; specific disposal site needs for operations, safety, and health; DOE Orders; characterization and identification of the wastes; disposal site performance assessment conclusions; environmental impact assessments; and a need for a quality process assuring safe transport and disposal of LLW. A disposal site's criteria are only as good as its enforcement program. Concerned members of the community and state where the site is located can only be assured that accepted criteria are adequate if there is documented evidence of the enforcement of the criteria. Through involvement in the development of the criteria and its enforcement, RWAP team members, under the direction of their respective DOE RWAP Program Managers provide this crucial step.

As the processes continue to evolve, so will the need for the NTS WAC to be revised. Such is the case with the development of NVO-325, Revision 2, to include all the elements necessary to continue to safely and effectively manage and dispose of the nation's radioactive low-level waste.

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ACKNOWLEDGEMENTS
The authors desire to recognize the following individuals who provided information for and/or review of the contents of this presentation:
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58-3

DOCUMENTING PROCESS KNOWLEDGE FOR LOW-LEVEL WASTE CHARACTERIZATION TO DEMONSTRATE COMPLIANCE WITH THE "NEVADA TEST SITE DEFENSE WASTE ACCEPTANCE CRITERIA, CERTIFICATION AND TRANSFER REQUIREMENTS," NVO-325, REVISION 1 (JUNE 1992)

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ABSTRACT
Process Knowledge can be a viable method for waste characterization when barriers such as personnel exposure (As Low As Reasonably Achievable [ALARA]), analytical laboratory limitations, waste matrices, and costs limit or exclude the use of sampling and analysis (1, 2). However, due to the lack of definitive regulatory requirements and specific definitions relative to process knowledge, waste generators have little guidance as to what level of documentation is needed to objectively demonstrate acceptable process knowledge.
Waste generators that ship waste to the Nevada Test Site (NTS) must demonstrate that each waste stream meets the criteria established in the "Nevada Test Site Defense Waste Acceptance Criteria, Certification and

Transfer Requirements," NVO-325, Revision 1 (June 1992). In order to demonstrate compliance with the criteria specific to waste characterization, generators must objectively demonstrate that methods and procedures have been developed and implemented that will ensure accurate determinations of the physical, chemical, and radiological properties of the waste (1, 3). Using a systematic approach for documenting process knowledge and incorporating pre- and postgeneration administrative controls can be effective in demonstrating adequate waste characterization.

INTRODUCTION

There are numerous activities involved in the generation and subsequent management of waste that have the potential to affect the physical, chemical, and radiological properties of the waste. The obvious activities that impact the properties of the waste are the type of materials used in the process, and the fate of those materials during the actual process that generates the waste. These are the areas that generators have historically concentrated on for process knowledge. However, to effectively demonstrate adequate characterization, especially the absence of hazardous wastes, other administrative controls and waste management activities must be utilized and documented. In most cases, by the time a waste is packaged, certified, and ready for disposal, numerous organizations have had some type of involvement related to the waste, either directly or indirectly. For the most part, documenting each organization's activities related to the waste lends itself to better demonstrating properties of the waste.

For the purposes of demonstrating adequate process knowledge to meet the waste characterization requirements encompassed in NVO-325, there exists a hierarchy of activities and information that must be documented.

Description of the characterization approach.

Description of the process generating the waste.

Identification of the procedures and/or other supporting documentation used in the process, materials used in the process, controls for segregation and handling of regulated materials, waste types (low-level, mixed, hazardous, etc.) to be generated, and methods used for determining constituent (chemical, radionuclide) concentrations in the waste.

Once the aforementioned aspects have been identified, a documented review of the information should be performed by personnel of appropriate expertise to ensure that proper waste determinations have been accomplished and adequate controls are in place to maintain the status of the waste characterization (1).

A good mechanism for developing a systematic approach for waste characterization via process knowledge is through the use of Data Quality Objectives (DQOs). By initially establishing DQOs, all of the parameters, inputs, decision rules, and documentation necessary to establish acceptable process knowledge can be identified (4). By documenting this approach, the reviewer is provided with an explanation of the data, decisions, and logic that went into the characterization of the waste.

A primary data collection source for process knowledge will be the actual person and/or organization that is responsible for the process that generates the waste. Depending on the specifics of the generator organization, and the regulatory requirements that must be met (Resource Conservation and Recovery Act [RCRA], Toxic Substances Control Act [TSCA], state regulations, Waste Acceptance Criteria, etc.), the characterization process should involve other organizational aspects such

as compliance, waste management, quality assurance, procurement, and training as necessary.

GENERATOR ACTIVITIES

For the purposes of this paper, the waste generator is a person or persons within the organizational group that is directly involved with the process that actually generates the waste. Because of its intimacy with the waste generating process, the waste generator can provide a wealth of information for establishing and documenting process knowledge. The generator can provide information relative to identifying and describing the actual process, procedures, logbooks, records, or other documentation used throughout the process. Depending upon the level of control involved in the process, the generator may also be able to provide detailed information about the identification of the actual waste components, any hazardous and/or other prohibited materials used in the process, associated concentrations, controls for segregation and handling of these materials, isotopes present and associated quantities, and the identification of all waste types that may be generated (low-level, hazardous, mixed waste). Other documentation that may not be directly related to the process, but that may provide good additional information,are things such as spill reports, occurrence reports, chemical inventories, abatement documentation, material tracking information, and purchase orders. Once the necessary information and documentation regarding the waste generation process has been obtained and the basis for waste characterization has been established, the information should then undergo a documented review by personnel of various expertise as appropriate (1).

ENVIRONMENTAL REVIEW

A documented review of information obtained from the waste generation activities can serve several purposes. First, it provides a mechanism for data verification; secondly, it strengthens the defensibility of the process knowledge by involving specific expertise in the waste characterization process. The unique aspects of individual processes and associated waste streams will dictate what type of expertise should be involved in reviewing and verifying the information supplied by the waste generator. For the purposes of demonstrating the absence of hazardous waste, personnel with expertise in RCRA, TSCA, the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA), etc., should be involved in the review (1). Along the same line of reasoning, personnel with experience in health physics should be involved in evaluating and verifying the radiological aspects of the waste stream such as radionuclide identification and activity determinations. Environmental reviews should not be limited to documentation reviews. In addition to reviewing operating procedures, logbooks, etc., personnel should also evaluate the waste generating process(es), examine facilities where the waste is generated, and evaluate the adequacy of controls that are in place for waste segregation, packaging, and handling (1). The outcome of this review should be the verification of the waste types that will be generated, determination of the adequacy of waste handling/segregation controls, and the acceptable identification of waste components for each waste type. Through the environmental review, the determination will be made that either adequate information and documentation exists to meet the data needs defined by the DQO process, or that areas with insufficient data will be identified and can be addressed as necessary (4). The review process can also be used in the

evaluation of waste minimization practices that are currently in use and can help determine whether or not any improvements can be implemented. A documented environmental review that includes review of documents, evaluation of controls, verification of types of waste being generated, and identification of acceptable and unacceptable waste components then becomes a major component of the final package for demonstrating adequate process knowledge.

FINAL PRODUCT

The final product with regard to using process knowledge for waste characterization must be able to demonstrate through objective evidence (documentation) that the waste has been adequately characterized to ensure proper treatment, storage, and disposal (1). Using this approach, the final product can demonstrate acceptable process knowledge through a documented description of the waste generating process; a documented characterization approach that identifies data inputs, parameters, and justification for decisions; the identification of data used in the characterization determination; a documented review of the data used; identification of acceptable and unacceptable waste components for each waste stream; and the identification and evaluation of controls for waste handling (packaging, segregation, storage).

Often, there is a substantial amount of documentation that is used in the waste characterization process. Assimilating all of this information into one area may not be feasible, so it is important that some type of traceability is established between all of the associated documentation such that the data can be presented in an auditable fashion. The presentation of the characterization data can be as simple as a documentation package that includes a summary of the characterization approach, description of the waste stream and waste generating process, and the identification (reference list) of the documentation that substantiates characterization of the waste.

Under NVO-325, generators are required to maintain characterization information in waste stream-specific files. The basic format for these types of files includes an executive summary that describes the specific characterization approach used and references the documentation that was used in the characterization process to demonstrate the characteristics of the waste. Examples of documentation that should be referenced are process procedures, operating procedures, waste packaging procedures, historical analytical data, laboratory logbooks, purchase orders, documented environmental reviews as described above, chemical inventory records, waste handling procedures, and post characterization controls for maintaining characterization status.

CONCLUSION

The substance of acceptable process knowledge is the ability to objectively demonstrate what the physical, chemical, and radiological properties of the waste are. Acceptable process knowledge goes beyond statements like "solvents are not in this waste stream because they are not used in the process." It is the procedure, purchase order, inspection reports, etc., that identify solvents as being absent from the process that objectively demonstrates the absence of solvents.

Defining a systematic approach (DQOs) for characterizing waste by process knowledge should be one of the first steps in the characterization process. The benefits of the this type of approach is that a great deal of information is documented at the beginning of the characterization process. Information identified during this definition stage (what

requirements have to met, what was initially known about the waste, what additional information needed to be obtained, decision rules, and decision errors) is, in itself, good process knowledge documentation . Once this systematic approach has been defined, the waste generators and generating process should be a primary source for data collection. All data defined as inputs to the characterization decision should then undergo a documented review by appropriate experts. This review will verify that the information adequately supports the waste determinations and demonstrates compliance with identified requirements. When the characterization process is complete, the information should be assembled and maintained in a manner that ensures document traceability, is readily retrievable and auditable, and can objectively demonstrate the properties of the waste and the adequacy of the waste determination.

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58-4

CHARACTERIZATION OF LOW-LEVEL MIXED WASTE FOR LAND DISPOSAL RESTRICTION COMPLIANCE CONSERVATIVE VS. REGULATORY APPROACH

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INTRODUCTION

The Resource Conservation and Recovery Act (RCRA) was passed into law in 1976. This was the initial start of a nationwide movement to restrict the land disposal of hazardous wastes. These regulations appear in Title 40 of the Code of Federal Regulations (CFR), Parts 264 and 265. In 1984, Congress passed the Hazardous and Solid Waste Amendments (HSWA). This mandated new land disposal limitations. HSWA required the U. S. Environmental Protection Agency (EPA) to issue new regulations referred to as the land disposal restrictions (LDRs).

The land disposal restrictions apply to waste management activities controlled by two environmental laws, RCRA and the Safe Drinking Water Act (SDWA). SDWA controls waste injected in deep wells. All other waste disposal activities are regulated by RCRA. The LDR program is designed to discourage placing untreated waste in or on the land when a better treatment or immobilization technology exists.

The Rocky Flats Environmental Technology Site (RFETS) requires the following steps to be performed to characterize a waste form for land disposal:

1) Preliminary investigation and identification
2) Sampling and analysis plan development
3) Development of sampling methodology
4) Administrative and technical preparations
5) Sampling of waste
6) Laboratory analysis
7) Data validation
8) Waste characterization report and wrap-up

We are focusing on the cost associated with the 8 tasks listed above for characterizing specific RFETS waste forms. Currently, Saltcrete, Item Description Code (IDC) 804 is being shipped for disposal at Envirocare of Utah, Inc. (Envirocare), having completed all of the above steps. Saltcrete is cemented chloride, nitrate, and sulfate salts generated from process water through an evaporation and subsequent cementation. Four other low-level (LL) and low-level mixed (LLM) waste forms have been proposed to be characterized for future disposal at either the Nevada Test Site (NTS) or Envirocare (commercial disposal site). They are as follows:

1) LLMW form Cemented Composite Chips, classified as IDC 327. This waste form represents approximately 95 cubic meters of LLMW. The cemented composite chips waste form is a solid matrix consisting of metal machining chips and Portland cement.
2) LLMW form Pondcrete classified as IDC 805. This waste form represents approximately 5,700 cubic meters of LLMW; of this we expect to qualify 2,000 cubic meters without treatment for offsite disposal. The Pondcrete is a cemented sludge. The sludge was collected from the solar evaporation ponds. It was generated by mixing Portland cement, water and pond sludge.
3) LLMW form Roaster Oxide classified as IDC 069. This waste form represents approximately 66 cubic meters of LLMW. Roaster Oxide is generated from stabilizing Uranium turnings. The uranium metal was roasted (converted into an oxide) to remove the pyrophoric nature of uranium metal.
4) LLMW form Blacktop, Concrete, Dirt and Soil classified as IDC 374. This waste form represents approximately 400 cubic meters of LLMW. This waste form was generated from various cleanup and construction activities around site.

CASE STUDY

The Saltcrete waste form will be used for the case study, with the results being applied to the other four waste forms.

Saltcrete carries a number of RCRA hazardous waste codes (F codes only). The codes that take precedence are D002 and F039. D002 is applied to certain waste streams at the point of generation; F039 is applied to the interceptor trench water that is included in the saltcrete process. D002 and F039 waste codes invoke the underlying hazardous constituent rule [40 CFR-268.40(e)], which requires all underlying hazardous constituents must meet the Universal Treatment Standards (UTS) found in 40 CFR-268.48, Table UTS, prior to land disposal. Invoking the underlining hazardous constituent rule requires the waste generator to know the concentration of over 219 constituents in the waste (many compound families are included as one constituent (e.g., polychlorinated biphenyls [PCBs] only count as one). However, the rule only requires that the waste generator

know the concentrations for constituents that may reasonably be expected to be present. The waste generator is not required to analyze for all of the compounds in the UTS list.a To date, RFETS has required all of the organic compounds found in the UTS list be determined.

To reevaluate the waste forms, two data bases were reviewed: Wastren Chemical Data (a study performed by a contractor to determine all the chemicals onsite); and the Plant Chemical Tracking Data. Analytical data from seven waste types were evaluated. All nine sources of information were summarized. The Wastren Chemical Data was considered to be out of date. The current plant chemical tracking data base contains all of the compounds (over 10,000 entries) in active use on site.b The analytical data from the 7 waste types reflects all of the chemicals that have been used and disposed of in the past. This includes not only saltcrete data, but also surface water plant discharge and interceptor trench water. This data was mapped into the UTS List of Compounds.

After evaluation of the listed data sources, 63 compounds were determined to be probable compounds for analysis. These are listed in Table I. Twenty-eight of these compounds were found in the listed waste streams and waste types in Table I, the rest were from the Plant Chemical Tracking Data Base. Eight compounds were identified from the analysis of the waste form that are not found in the Plant Chemical Tracking Data Base. These compounds were probably used on the site in the past: the phthalates (3) as plasticizer, the cresols (2) as wood preservative, and it has not been determined where the nitriles (2) and nitrophenol have been used in the past.

Evaluating the amounts of material present on site, 20 of the 63 compounds were eliminated from the concern list due to the small amount of materials present on site. For these 20, we assume a worst case contamination scenario; all of the material in any building being poured into the process line at the same time and ending up in the brine solution for processing. Calculations show the levels of these materials in the saltcrete will still meet LDRs. (The brine solution that is being processed at any given time represents approximately 3 million gallons of process water, making approximately 20,000 pounds of saltcrete.) Three other compounds can be removed by regulation:

Table I

carbon disulfide, cyclohexanone and methanol.c Of the 40 compounds remaining, all may be determined by requesting the "normal" SW-846; 8260 and 8270 waste analytes suite of compounds.

Saltcrete shows that by considering all the aspects of the waste form, it is possible to reduce the cost of characterizing the waste for land disposal.

CONCLUSIONS

From the case study, evaluation of the eight steps required to characterize waste for land disposal, we see Step 1 increasing in cost by using a strict regulatory approach. This is due to the additional time required to evaluate the waste form and processes that generate it. However, the other seven steps have decreased in cost. The overall savings range from 20% to 30%. This is a significant decrease in costs. This is shown in Table II for saltcrete analytical cost.

Table II

The Base Line cost for pondcrete for six of the steps is shown in Table III. For these six steps, a decrease in costs of over 30% would be realized by following the regulatory approach.

Table III
Applying this to the four waste forms in the study (Table IV), we see a potential savings of $679,200.
Table IV
Although all 8 steps are required to be completed (we have not removed any requirements), by changing the emphasis on each step, a large cost savings can be realized. The overall results show a projected savings of 24% for these 4 projects.
There are a number of factors that have not been considered. These include faster completion of projects, reduced staffing, improved worker morale, etc. These factors should lower the cost further.

58-5
A TECHNICAL AND REGULATORY FOUNDATION FOR MIXED-WASTE DELISTING

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ABSTRACT

The purpose of this paper is to propose a risk-based approach that can be used to identify mixed-waste streams that can be delisted from regulation under the Resource Conservation and Recovery Act (RCRA). The goal of this paper is to describe both the technical and regulatory foundations by which one can identify ostensibly mixed-waste streams for which the risk from the hazardous component of the waste stream is minimal, so that it can be treated as low-level radioactive waste. In addition, the approach described here may also be applicable to defining de minimis levels of purely hazardous or purely radioactive wastes.
First, an approach developed by the U.S. Environmental Protection Agency (EPA) for identifying de minimis levels of hazardous contamination in soils will be described. Second, it will be argued that an approach developed for unconfined soils can be used with great confidence at licensed low-level radioactive waste disposal facilities. Third, this approach will be argued to be legally equivalent to satisfying no-migration variance requirements under RCRA. Fourth, an approach will be proposed for establishing a formal procedure for carrying through analyses for delisting waste streams to be disposed at particular sites. Since the approach is founded on EPA-generated approaches and information, regulatory issues should be eased. Furthermore, the approach is consistent with an approach published by the U.S. Nuclear Regulatory Commission for defining acceptable levels of residual radioactive contamination. This means that the potential exists for developing a common framework for both minimally hazardous radioactive and minimally radioactive hazardous mixed waste streams.

INTRODUCTION

The purpose of this paper is to provide an initial assessment of an approach that can be used to identify de minimis levels of hazardous contamination in mixed waste streams. However, the approach described here is also applicable to waste that is only hazardous. The approaches and principles described in this report draw on numerous ongoing efforts in the area of mixed waste management. The National Council on Radiation Protection and Measurements (NCRP) has had a scientific committee (Scientific Committee 87-2) studying issues related to mixed waste risk

assessment for several years. The International Atomic Energy Agency (IAEA) has established an international program to study the issues associated with common risk assessments for the hazardous and radiological components of mixed waste. IAEA has been developing a first step toward developing a common risk assessment framework for assessing environmental impacts from different types of sources (1). This framework is intended to be a first small step toward addressing the mixed waste issue. The problems are recognized to be thorny, and nobody expects quick resolutions. Note that this is at the international level: the problems are not entirely rooted in regulatory differences between U.S. Government agencies. Nevertheless, it is generally recognized that a key component to resolving the mixed waste issue is to minimize the volumes of mixed waste by defining mixed waste to only include wastes in which there is truly both radioactive and chemical risk.

The goal of this paper is to identify ostensibly mixed waste streams, and demonstrate that the hazardous component of the waste stream is sufficiently minimal that it can be treated as low-level radioactive waste. The structure of this memo is as follows. First, an EPA-developed approach for identifying de minimis levels of hazardous contamination in soils will be described. Second, it will be shown how that approach can be used with confidence at low-level radioactive waste disposal facilities. Third, this approach will be shown to be equivalent to satisfying no-migration variance requirements under RCRA. Fourth, an approach is proposed that will establish a formal approach for carrying through analyses for delisting waste streams to be disposed at particular sites. Since the approach is founded on EPA-generated approaches and information, regulatory issues should be eased. Furthermore, the approach is consistent with an NRC-developed approach for defining acceptable levels of residual radioactive contamination. The means that the potential exists for developing a common framework for both minimally hazardous radioactive and minimally radioactive hazardous mixed waste streams.

DRAFT EPA SOIL SCREENING METHODOLOGY

In December 1994, EPA issued a draft approach for developing both generic and site specific Soil Screening Levels (SSLs) that are intended to be Administrative Improvements to the Superfund program (2). The idea behind the SSL approach was to define generically de minimis levels of hazardous contamination for Superfund sites. More importantly, the draft position provides a framework for developing site-specific SSLs if the generic SSL is found to be overly restrictive for a particular site.

A three-tiered approach is described in the SSL document. The approaches begin with simple, arguably conservative analyses and progress to less conservative,yet more complicated, analyses as shown in Fig. 1. At the first tier, soil concentrations are derived from screening models, argued to be generally conservative, that link the soil concentration to a health risk by exposure pathway models. If soil concentrations are less than this level, no further action of any kind need be taken. That is, they are effectively de minimis levels. At the second level, easily gathered site-specific information is used to modify parameters in the simple generic models to produce an alternative, less conservative, representation of the site. Concentration levels derived in this way are still considered to be de minimis levels, but represent site-specific de minimis. At the third level, a more detailed modeling approach is used, based on a more extensive collection of site-specific data. At this

stage, the level of effort needed for the analysis can be expected to approach the level of effort needed for low-level waste site performance assessments.

Fig. 1

It is interesting to note the dramatic similarity of this EPA approach with an independently developed approach from the NRC for the comparable soil-screening problem at decommissioning sites under NRC jurisdiction (3). The only significant differences between the two approaches are in the details of the models used for generic screening. It is also interesting to note the similarities between this evolutionary approach to site risk assessment and current low-level waste performance assessment guidance (see, e.g., Refs. 4, 5, and 6). This commonality suggests that there is the potential to develop common approaches for eliminating both minimally hazardous and minimally radioactive waste streams from the mixed-waste spectrum.

SOIL SCREENING LEVELS AND MIXED WASTE DISPOSAL

Let us now consider the significance of this approach for evaluating the risk of hazardous materials in soil relative to mixed-waste disposal. First, a variety of exposure pathways are considered in the SSL approach. Low-level waste assessments are usually dominated by the ground-water pathway, although ground water is not believed to be an important pathway at either the proposed Texas Compact site (7) or the Area 5 facility at the Nevada Test Site (8) because of low recharge, nor is it believed to be important at the Clive, Utah mixed-waste site because of high ground-water salinity (9). In any case, the SSL approach can be adapted for use in evaluating a low-level waste disposal facility, since it is a multiple pathway analysis. Nevertheless, for the majority of low-level waste facilities, a subset of the SSL methodology (the portion relating to potential groundwater contamination) will be adequate. Second, the SSLs relate to unconfined soils, not engineered disposal facilities. It seems safe to make a general statement that for the same level of contamination, the risks associated with engineered disposal facilities will be less than the risks associated with unconfined soil. Third, the provision exists in the methodology to make a more reasonable estimate of an acceptable soil concentration based on a more elaborate calculation. We can conclude from these arguments that the generic SSLs can be used to identify a de minimis level for the hazardous component of mixed waste. If the hazardous concentration of a waste stream is below the generic SSL, we need spend no more effort to delist the waste stream from RCRA, and can treat it solely as low-level radioactive waste. Furthermore, we can use the latter two stages of the SSL process to justify, on a site-specific basis, that higher levels of contamination can be considered de minimis. Again, the result should be that wastes identified in this way can be treated as strictly radioactive, and their hazardous component can be ignored from a risk basis. In its most extreme case, we might end up conducting calculations similar to the low-level waste performance assessments, which are rather detailed in scope, and using those calculations to justify a de minimis claim for relatively high concentrations of hazardous materials.

SOIL SCREENING LEVELS AND NO MIGRATION

RCRA contains the provision that "there will be `no migration' of hazardous constituents from the disposal unit or injection zone for as long as the waste remains hazardous." The demonstration that this is indeed the case for a particular disposal system is known as the "no-

migration variance." In 40 CFR 268 and 271, "EPA interprets this statutory language to require that petitioners demonstrate that hazardous waste constituents will not migrate from the land disposal unit in hazardous concentrations for as long as the wastes remain hazardous [emphasis added]." Davis et al. (10) noted that this means that "no migration" does not mean "zero release;" rather it says that releases are permissible as long as they do not exceed health-based standards. Davis et al. also noted that EPA incorporated this interpretation into its final rule making on underground injection wells under 40 CFR 148, and incorporated this interpretation into its reviews of "no migration" variances submitted under 40 CFR 268.6. This interpretation was upheld in Natural Resources Defense Council vs. EPA litigation (11). Draft EPA guidance for determining compliance with 40 CFR 268.6 also suggests that "model assumptions and input data should be conservative and tend toward overestimating rather than underestimating migration." (12). The SSL approach described above clearly follows this same philosophy.

There are precedents for conducting performance assessment analyses to demonstrate "no migration" under RCRA. Davis et al. (10) described the basis for a unified assessment approach for 40 CFR 191, DOE order 5820.2a, and RCRA no migration for the Greater Confinement Disposal (GCD) facility at the Nevada Test Site. The GCD facility contains high-activity low-level waste, transuranic waste, and hazardous components; consequently all three regulations needed to be addressed. They were able to show that each of these regulatory structures hold common themes and approaches, and that demonstration of compliance with one can be shown to be equivalent to compliance with the others. In the case of the GCD facility, they were able to show that demonstration of compliance with 40 CFR 191 would also show compliance with the other two regulations.

Anderson et al. (13) have described an approach being used to develop a no-migration variance for WIPP. The intent of this approach is to use the full probabilistic performance assessment model being developed to demonstrate compliance with 40 CFR 191 to establish (in a posterior sense) conservative parameter values. These are to be used in a deterministic analysis to attempt to demonstrate that under ordinary operating conditions, waste can be expected to be confined to the facility boundary.

Kincaid et al. (14) used their performance assessment to make a case that nitrate levels can be expected to be far below the MCL for nitrates as a result of disposal of liquid wastes in grout at the Hanford site. As part of the Tri-Party agreement, this disposal technology was abandoned in favor of vitrification of the waste. As a result, the approach proposed by Kincaid et al. has never undergone a full review. However, the approach is significant. The assessment is used to demonstrate that MCLs for groundwater are not exceeded now, nor can they be expected to be exceeded in the future. This is equivalent to establishing a risk limit for the chemical portion of the inventory.

A similar approach was used in the performance assessment of the Z-area Saltstone facility at the Savannah River Site. This performance assessment explicitly considered degradation of groundwater by the release of nitrates (15). The reason for including nitrates in the analysis was a special agreement with the state authorities, rather than any legal objective that had to be met. The Saltstone facility is used to dispose of a liquid waste high in 99Tc; the Tc is chemically fixed in

place by the nature of the Saltstone. Compliance with agreed standards for nitrates was the most stringent problem in this assessment.

Other performance assessments that are not considering chemical risks are (1) the Nebraska (Boyd County) commercial facility, (2) Oak Ridge (16), (3) the SRS E-area vaults (17), (4) the Hanford 218-W-5 disposal facility (18), and (5) the INEL radioactive waste management complex (19). It should be noted that there are no requirements to include chemical risks in these assessments, so this comment is not intended as a criticism of these assessments. There are serious issues related to analyzing the chemical risk from ordinary low-level waste streams. In particular, there are usually insufficient data available on the chemical inventory (developed from shipping manifests) to develop a satisfactory analysis.

PROPOSED APPROACH

The following tiered approach is proposed for substantiating a delisting of a waste stream using risk-based no-migration arguments. The first tier is to use the generic SSL concentrations for soil, and to apply them verbatim to mixed waste. Waste streams with concentrations below this level can be delisted and disposed of as purely radioactive waste with no further effort needed to justify the minimal importance of the hazardous component of the waste. The argument for doing this is that the engineered enhancements will lower the risk compared to the risk from contaminated soil. The SSL levels "generally correspond to a 10-6 risk level for carcinogens, and a hazard quotient (HQ) of 1 for noncarcinogens" (2). Consequently, the risk of carcinogenesis is lower by far than the permissible radiological risk from the same waste (Using current regulatory approaches, 15-25 mrem/yr is presumed to equate to about 5x10-4 risk of cancer fatalities. There is, of course, considerable uncertainty in the actual relationship between risk and dose.) The combination of improved packaging and extremely low risk levels makes a strong case that there is no appreciable risk from hazardous waste streams that meet the soil screening levels. The argument here is that "minimal risk" is equivalent to "no migration."

The second tier is to use the EPA models with some site-specific information to provide a less conservative estimate of acceptable levels of contamination in a disposal facility. At this tier, we are not taking credit for engineered barriers, containment, or immobilization. As a result, the results are still extremely conservative, and will therefore be easily defensible when presented to regulatory authorities. Nevertheless, it may still be possible to delist more waste streams in this way than were delisted in the first tier. In particular, if some site-specific information is introduced for the NTS Area 5 waste management facility, it is clear that less stringent concentrations can be shown to be acceptable, because of the extremely favorable conditions at that site.

The third tier is to conduct a more elaborate analysis using more site-specific information. At this tier, the analysis can include the influence of immobilization technologies, containers, vaults, covers, and other engineering enhancements of the facility. The basis for this analysis can be the existing performance assessments for the low-level waste sites at which the waste stream would be disposed. It is expected that since the performance assessments exist, at least in draft form, for all DOE low-level waste disposal facilities, the amount of effort that will need to be expended to conduct this analysis will be minimal. However, it should be understood that the difficulty of convincing the

regulator will be proportional to the complexity of the analysis. As more complexity is introduced, and as the analyst takes credit for more beneficial effects of the system, the regulator tends to scrutinize the analysis more. In the first two tiers, it should be very easy to make the case that the risk posed from the hazardous material is minimal.

We note in passing that Smith et al. (20) and Little et al. (21) have described a consistent framework for conducting health risk assessments of mixed waste using performance assessment methods. This approach, known as SACO, is under development in Europe for worldwide use. When completed, SACO may form a consistent basis for conducting risk assessments for hazardous materials, radioactive materials, or mixed materials.

SUMMARY

There is a common thread of an approach that exists in both CERCLA and RCRA for defining an acceptable risk level based on pathway and transport modeling. The approach for CERCLA has been made explicit in the draft Soil Screening Level (SSL) methodology. The approach for RCRA has not been formalized, but EPA's clear interpretation of "no migration" is that it refers to no migration above health-based limits. It does not mean "zero release." The SSL methodology is identical in concept both to the soil screening methodology developed independently by NRC for radionuclide contamination, and to the approaches used in performance assessments for low-level radioactive waste disposal facilities.

As a result, in this paper an approach is proposed for delisting the hazardous portion of mixed-waste streams that is identical in concept to the SSL approach. The tiers progress from more conservative to less conservative, from less complex to more complex, from generic to site specific, and from easily defensible to more complex to defend.

One flaw to this proposed approach is that the Soil Screening Limits are still only in draft form, and it is unclear when EPA will issue final guidance. However, while the specific concentration limits may change, and the details of the screening models may change, the general approach and philosophy is likely to remain the same. Consequently, the approach will be applicable to the level of detail that has been proposed here.

One of the most interesting concepts is the commonality of approaches between NRC's 10 CFR 20 Subpart I analyses to demonstrate the soil contaminations are minimally contaminated with radioactive materials, and the SSL approach, which does the same for hazardous materials. This raises the possibility for developing a common approach for both ends of the mixed-waste spectrum: both minimally radioactive hazardous waste and minimally hazardous radioactive waste.

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DIFFICULT BUT NOT IMPOSSIBLE: MONITORING HIGHLY ACTIVE WASTES RETRIEVED FROM INTERIM STORAGE FACILITIES

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ABSTRACT

Major national nuclear programs have resulted in a considerable number of storage facilities containing either highly active liquors or highly active solid wastes. The latter in many cases contain inhomogeneous mixtures of fuel, metal scrap and debris. In the UK major efforts are underway to retrieve these wastes for treatment and, ultimately, disposal. In order to conduct these operations safely, and to meet the inventory data requirements for ultimate disposal, it is essential to achieve a valid measurement of inventory in the early stages of the waste routing process. BNFL Instruments Ltd are playing an important role by providing this measurement capability to retrieval projects on the parent company's Sellafield site. Current measurement solutions are based on many years of operational experience which the company has accumulated in delivering special purpose waste monitoring instrumentation for process control as well as for inventory measurements. In spite of the difficulties of measuring the inventories of very variable retrieved wastes, in high dose rate environments, measurement systems are being provided which satisfy the demands of both the repository operators and national regulators.

INTRODUCTION

In common with the other nuclear powers, Britain has constructed and operated several generations of nuclear reprocessing facilities since the early 1950's. The Sellafield site, now operated by British Nuclear Fuels plc (BNFL), has seen the construction and operation of early fuel storage, reprocessing, product finishing and waste storage facilities which are now in varying stages of shutdown or decommissioning. Of particular concern are the very variable solid wastes which may be contaminated with a variety of fissile and radioactive nuclides. Interim storage facilities within the site boundary have contained these highly active wastes safely and with minimum environmental impact for several decades. Now as part of BNFL's overall waste management strategy, major efforts are underway to empty these stores and repackage and encapsulate

the waste prior to ultimate disposal. Once emptied the buildings themselves can be decommissioned.

In this paper the issue of measurement is placed in the context of the waste retrieval operations and the particular difficulties of adequately monitoring the radionuclide inventory and fissile content are highlighted. The solution to such problems can be shown to lie in the experience gained by close involvement throughout more recent reprocessing and waste treatment operations. BNFL Instruments Ltd, a wholly owned subsidiary company of BNFL plc, have supported the parent company's engineers and designers by developing specialized instrumentation for all new plants at Sellafield for over 25 years. In the course of this support, techniques have been developed which, for highly active waste, will enable operators to perform a non-destructive assay prior to encapsulation and to establish the detailed characterization of waste required to satisfy the disposal authority and the national regulator.

WASTE STORAGE AND RETRIEVAL

An overview of the material flow in reprocessing operations and the routes by which wastes arise at Sellafield are shown in Fig. 1.

Irradiated fuel has historically been stored in water filled ponds prior to reprocessing. Once sufficiently cooled for safe reprocessing, the fuel is taken for pre-treatment. For Magnox fuel, this consists of stripping off the outer cladding prior to dissolving the metal fuel in nitric acid. The de-cladding debris (known as swarf) is treated as intermediate level waste (ILW). Current arisings of swarf are being sent directly for cement encapsulation into 500 litre drums, at a plant commissioned in 1990. A second plant commissioned in 1992 is used to deal with the cladding residue from oxide fuel reprocessing in a similar way.

Fig. 1

Storage

Prior to the availability of the encapsulation plants, the cladding waste was sent to interim storage facilities. The first such facility was a dry silo, but from 1960 onwards new arisings were stored underwater. The silos were also often used to store scrap associated with the de-cladding process. The contents of the silo compartments filled in the early years of operation have corroded to a mixture of sludges and solids. The original fuel storage ponds, which are now redundant, also contain sludges, corroded fuel and fuel handling equipment.

Retrieval

As part of BNFL's overall waste management strategy, major efforts are underway to empty the redundant ponds and interim storage facilities and repackage and encapsulate the waste (1). Once emptied it will be possible to complete the decommissioning process by dismantling the building itself.

Safety and dose minimization are important issues in engineering a cost-effective retrieval process. Major problems to be overcome include: high dose environments, the need for containment to prevent the spread of contamination, chemical hazards, fire risks and other safety issues. Significant development work and plant trials have been undertaken to find the safest and most cost-effective means of retrieval. Purpose built machinery is being designed and constructed to facilitate retrieval and subsequent treatment of the stored waste into a form suitable for ultimate disposal. Several dedicated plants will be built to receive this waste. Sorting, screening, settling, packaging and drying prior to

encapsulation of the waste are all options either in operation or under consideration.

The retrieval of stored irradiated fuel bearing wastes has already begun on the Sellafield site and will continue well into the next century.

Measurement and Characterization

Detailed reviews including sampling, analysis and scrutiny of plant records have been carried out on the contents of the redundant storage facilities. This has enabled sufficient information on the stored materials to be determined for the selection of waste routing through to disposal.

The waste retrieval, treatment and repackaging process will generate waste packages which will require characterization on an individual basis. The sampling and plant records can often provide a good representation of the overall contents of the waste stores, however, the characterization requirements for the final waste product cannot be satisfied with this information alone. One of the major requirements is the provision of radionuclide inventory and fissile content information to satisfy disposal regulations and to ensure criticality safety.

Non-destructive assay (NDA) techniques can be used for measurement of each waste package prior to ultimate disposal. The requirements for NDA depend on the application. There are a number of general driving forces including safety, dose minimization, regulatory requirements, disposal authority requirements and process control.

The major function of waste product monitoring is the generation of an extensive radionuclide inventory including fission products, trans-uranic radionuclides and total fissile content of the waste, where applicable.

CHALLENGES FOR WASTE MEASUREMENT

In determining the optimum monitoring solution, physicists, engineers and designers have co-operated to find the most suitable waste retrieval strategy incorporating the required measurements. The technique used, location and physical arrangement of the system must fit in with the need to engineer a safe and cost-effective waste routing process through to disposal, taking into account the suitability of measurement at various possible locations.

Typically, waste is retrieved in batches, which contain a variable mixture of materials including highly active fuel residues, fuel cladding materials and miscellaneous operational scrap. Figure 2 illustrates a mock-up of typical retrieved wastes undergoing a sorting process.

Treatment, sorting and repackaging are often performed before the monitoring stage. The detailed inventories that have been prepared for the storage facilities themselves will generally provide only limited information for the waste measurement system to use (e.g. during calibration). These measurement conditions are also often complicated by the bulk quantities of waste involved and the inhomogeneous and variable nature of the materials.

Fig. 2

Further problems arise due to the high dose rates and potential for loose contamination associated with the waste treatment processes. This often leads to shielding, handling and containment requirements which makes the measurement more problematic.

Quality Assurance plays a key role throughout the entire process of providing plant operators with the solution to their measurement needs. The instrumentation must be capable of generating fully reproducible

results and must operate reliably and be capable of performing regular automated self checks.

MEASUREMENT TECHNIQUES

There are a variety of radiometric NDA techniques which can be used in waste monitoring. The basis of all such measurements is the quantification of certain properties of the sample based on the detection of some form of radiation. Due to the bulk quantities of materials encountered, the most suitable NDA techniques are based on either gamma or neutron measurements.

Gamma Ray Measurements

In order to obtain the maximum amount of information from a gamma ray measurement, High Resolution Gamma Spectrometry (HRGS), based on high purity germanium (HPGe) semiconductors, is preferred to low resolution spectrometry based on scintillation detectors. Within irradiated fuel, there are many gamma emitting nuclides. Due to the attenuation effects within the waste container, only higher energy gamma rays can be measured in large containers. In practice there are very few photopeaks in the spectrum of historic irradiated fuel bearing waste. This is mainly due to the long cooling times and Compton scatter interference effects from several high energy, high intensity gamma rays (mainly 137Cs and 60Co). Gamma ray detection can also be used by instrumentation at the retrieval facility and within the treatment and sorting facilities. Gamma ray imaging techniques have been developed which allow operators to identify `hot spots' of gamma activity on a contaminated wall, say, or at a sorting table. These instruments can include a coarse spectrometric capability which enables fuel to be discriminated from other items. This `gamma view' can be combined with the traditional optical image to assist in segregation of the waste and reduces operator dose uptake.

Passive Neutron Measurements

Passive neutron counting involves measuring the intrinsic fast neutron emission from the waste. This arises from two types of event: spontaneous fission and (,n) reactions. Typically 2 or 3 coincident neutrons will be emitted from each spontaneous fission event, whilst (,n) reactions (caused by the interactions of alpha particles with light elements such as oxygen) result in the emission of a single neutron. Maximum (,n) emission occurs when the alpha emitter is chemically linked to the light element as, for example, in corroded fuel.

Assay systems can utilize the detection of the total or coincident neutron emission. The latter may involve the detection of two time correlated neutrons (referred to as Passive Neutron Coincidence Counting PNCC) or multiple time correlated neutrons (Multiplicity Counting). Coincidence techniques allow the signal from spontaneous fission to be isolated from the (,n) signal. This is necessary when the chemical composition of the waste is poorly characterized such that the ratio of the (,n) to the total neutron emission can vary.

Active Neutron Interrogation

In contrast to passive methods, active neutron measurements rely on the detection of induced radiation. Neutrons from an interrogating source are introduced into a measurement chamber made up of moderating and shielding materials. Fast neutrons quickly slow down in the chamber by multiple elastic scattering in the moderating materials. In addition some moderation and absorption usually takes place in the measurement sample; the magnitude of which will depend on the matrix composition. The neutrons induce fission events in any fissile material present giving

rise to the emission of secondary fast neutrons and gamma rays. It is this secondary radiation that is detected to give a measurement of the total amount of fissile material present.

One method of active neutron interrogation is the differential die-away (DDA) technique. Short pulses of fast neutrons from a neutron generator are injected into the measurement chamber. This gives rise to a thermal neutron flux which persists for a few milliseconds. Fast neutrons arising from the induced fission events are then counted using fast neutron detector packages embedded in the chamber walls. These detector packages have much shorter characteristic neutron lifetimes than the chamber and this large difference in the die-away time makes the measurement possible. The measurement signal is used to quantify the mass of fissile material present.

EXPERIENCE IN WASTE MEASUREMENTS

Over the last 25 years, many special purpose waste monitoring systems have been developed to meet a wide variety of difficult measurement problems in the Sellafield reprocessing and waste treatment operations (2). As new generations of increasingly sophisticated plant have been commissioned and the demands of the regulators have become more stringent, several generations of instrumentation have been developed from concept through to commissioning, calibration and fully automated operation. Several examples are given here of such systems.

Swarf Tipping Monitors

One of the earliest measurements of highly active wastes was performed during the process of transferring the Magnox decanning waste to its interim store. A measurement of the uranium content of the waste was required in order to quantify carry-over of fuel from the decanning plant. The chosen measurement technique was based on detection of a distinctive high energy gamma ray from a short cooled gamma emitter which is only associated with irradiated fuel (the 2.18 MeV gamma ray from 144Pr was used).

Two low resolution NaI(Tl) gamma spectrometers viewed a swarf bin which was scanned past the detectors before and after tipping. Calibration was performed using fuel rods of known irradiation history and cooling time, placed in a supporting frame, at known positions in a bin full of water. The measurement enabled the reprocessing plant operators to gain confidence in the decanning process and was also of sufficient quality to contribute to the materials accountancy for the plant. A great deal of performance assessment work was carried out during and after commissioning of the instrument. This work has provided confidence in the accuracy of the declared data and provided valuable experience in the development of later generations of highly active waste assay instrumentation.

Swarf Inventory Monitor

The next generation of monitoring instrumentation was developed to improve on the original measurements carried out during swarf tipping. In 1985, the first high technology system (designed to quantify fuel carry-over), was installed in a new fuel handling plant. Subsequently a system was developed to provide improved measurements in parallel with the advent of swarf encapsulation. As previously noted, since 1990 arisings of waste from the Magnox decanning process have been encapsulated into 500 litre drums to await ultimate disposal. A requirement of the licensing and regulatory bodies is that an extensive radionuclide inventory must be provided for this waste. In addition, there is a

process control measurement requirement, to monitor for excessive quantities of uranium in the swarf prior to export to the encapsulation plant. The most appropriate measurement technique is HRGS, since the relatively short cooling time of this waste stream enables a wide variety of nuclides to be directly measured. This provides sufficient information to determine the irradiation history of the fuel from which it is possible to infer the uranium content as well as a variety of other non-measurable radionuclides.

Measurements on bulk quantities of swarf would be undesirable because of gamma absorption effects, so the swarf is measured in small batches. The Swarf Inventory Monitor (SIM), located in the decanning cell, is illustrated in Fig. 3 (a neighboring instrument for measuring the cooling time of the fuel is also shown here). For each batch, a HPGe detector views the waste and acquires a gamma ray spectrum. The activity of the gamma emitters is calculated from analysis of the spectrum. This calculation takes into account detection efficiency, background from the measurement tray and self-attenuation in pieces of uranium. The irradiation history of the fuel is derived from various ratios of the measurable gamma emitters. Uranium mass and the activity of other radionuclides can then be quantified using known relationships derived from the fuel inventory code, FISPIN (3).

Fig. 3

The HPGe detector is mounted in the cell roof on a precisely engineered movable table. The detector views the waste tray through a gamma ray collimator. The positioning of the detector is carefully monitored with an infra-red proximity sensor in order to ensure that the calibration arrangement remains valid during all subsequent measurements. An ultra high count rate capability (handling an input of up to 500,000 counts per second) is required to provide a wide dynamic range to cope with the variation in activity of monitored swarf. Advanced electronics are used to process the signal from the detector with accurate dead-time correction. Based on experience gained in using HPGe detectors in process plants, special mounting and screening is used on the detector and its electronics to overcome electrical and mechanical noise. Considerable effort was made to provide high quality diagnostics and to make the system `user friendly' for the plant operators and engineers.

The monitor has been operating since July 1990 and provides a reliable determination of uranium mass and radionuclide inventory. Detailed assessment work was performed after commissioning of the instrument in order to identify and eliminate potential biases. This has enabled the system's process parameters to be finely tuned to the actual measurement conditions that have been found to arise in the plant.

Fissile Material Detector

Current arisings of miscellaneous items of ILW from the Sellafield plant are stored in 3m3 boxes at a purpose built facility, the Miscellaneous Beta Gamma Waste Store (MBGWS). For criticality safety, it is necessary to quantify the fissile content of this waste prior to filling of the storage boxes. The technique employed for this measurement is the active neutron interrogation technique, DDA.

The Fissile Material Detector (FMD) is capable of measuring the wide variety of wastes consigned to the store. Various calibrations were performed for each type of waste classification during commissioning. The waste consigned to the plant includes miscellaneous mixtures of: steel, lead, concrete, graphite, cellulose and plastics. The total fissile

content is derived using the DDA measurement result, operator declared classification and the neutronic properties of the sample, determined during the measurement.

The measurement chamber consists of polyethylene and graphite, with neutron detectors and a pulsed neutron generator located in the walls. Lead shielding is used to reduce the gamma flux to the detectors from the waste. Regular automated checks on the system performance are performed using a source transfer system.

Hulls Monitor

The reprocessing of oxide fuel at the THORP plant generates another ILW stream. This consists primarily of the residues of fuel assemblies (hulls") which are the waste product from the dissolver after the shear/leach process. The hulls are exported from THORP to an encapsulation plant where they grouted and stored prior to ultimate disposal. Monitoring of the hulls (prior to export) is necessary in order to:-

Ensure criticality safety in subsequent handling.

Ensure that fuel retention complies with limits for interim storage and ultimate disposal or return to the customer.

Demonstrate that uneconomic fuel retention has not occurred.

Derive inventory information for customers and regulators.

Provide safeguards and materials accountancy data.

The Hulls Monitor, illustrated in Fig. 4, has been developed to satisfy these measurement requirements using a combination of neutron interrogation (DDA), passive neutron totals counting and HRGS. The application and development of these techniques to this measurement scenario represents a major measurement challenge.

Fig. 4

The residual fissile content of the leached hulls is determined from the DDA measurement. Total uranium content is derived using the fissile content, the measurement of passive neutron emission and information on the initial enrichment of the fuel batch provided by the reactor operators and from measurements (via a separate monitor) on the fuel before shearing. Additional inventory information is determined using HRGS, the passive neutron count, initial enrichment and cooling time of the fuel and FISPIN derived correlations.

The measurement is undertaken on a dissolver basket, 0.67m in diameter filled with hulls up to a depth of 2m. In addition to the hulls, the basket will contain additional fuel assembly hardware (such as end appendages) and a small amount of neutron poisoned dissolver liquor trapped within the hulls. During a measurement sequence, the basket is lowered into a re-entrant thimble in a monitoring cell below the basket handling area. Fast neutron detectors and the neutron generator are housed in a collar around the thimble. The collar is constructed of moderating materials designed for the DDA measurement. Plant ruggedised electronics have been developed to provide the detection systems with noise immunity, high count rate capabilities and fast recovery times essential for DDA measurements. The HRGS system is located outside the cell and views the basket through a collimator set into the cell wall above the neutron collar.

Comprehensive self-checking and back-up facilities have been designed into the instrument. Functionality of the neutron and gamma detection systems, and of the neutron generator is confirmed by standardization checks that are initiated by the basket handling cave control system at

regular intervals and before each measurement. Confirmation of a satisfactory standardization must be provided before a measurement can be carried out. In addition, real-time checks are continually performed by the software to confirm the absence of fault conditions.

The hulls monitor neutron system is capable of measuring residual fissile content with a lower limit of detection of 5-10 g 235U equivalent levels. Further work is underway to develop an advanced radiometric instrument to deal with the more challenging measurement of hulls from mixed oxide and higher burn-up uranium oxide fuels.

THE MEASUREMENT SOLUTION

Development work has been performed to adapt the existing waste measurement technologies to the challenge of monitoring highly active waste retrieved from interim storage facilities. One area in particular where considerable development work has been applied is in the application of neutron assay techniques. These measurements are often needed for criticality control and as supporting information for radionuclide inventory of waste packages prior to ultimate disposal.

Technique Development

The assay of trans-uranic elements in highly active waste presents a major measurement challenge. Passive techniques, in isolation, are of limited applicability. The gamma rays associated with the trans-uranic nuclides are often difficult to detect due to the low energy or low emission rates. For the measurement of irradiated fuel bearing wastes, there is usually no measurable gamma signal from any trans-uranic nuclide because of the interference from the intense gamma rays from fission and activation products. Passive neutron counting can be used on highly active wastes but this measurement can only quantify the total spontaneous fission neutron emission rate (due to emissions from even mass nuclides such as 238U, 240Pu and 244Cm). In isolation, this measurement cannot be used to quantify specific trans-uranic nuclides. To do this, additional parameters relating to the trans-uranic content of the waste need to be known or measured.

Active neutron interrogation enables the total fissile content of the waste to be measured (comprising the fissile nuclides 235U, 239Pu and 241Pu). For waste measurements, one of the most suitable forms of this technique is DDA. In the UK, as elsewhere in the world, the earliest applications of DDA have been the assay of low concentrations of fissile material in wastes that have low overall inventories of radioactive materials. In addition the materials presented for assay and the geometrical arrangements have traditionally been well characterized.

The adaptation of DDA to the highly active wastes retrieved from interim storage is complicated due to the following aspects of the measurement:-

Wastes can vary in composition. Materials that have very different neutronic properties (absorbers and moderators) are often found together. Examples include graphite, stainless steel, organics and sludges. Waste is heterogeneous. Frequently a container will contain voids and lumps of different materials.

Water is often present in the waste. For neutron measurements, the moderating effect of water is usually undesirable, particularly where the content is variable. Efforts can be made to design the waste retrieval process to present the waste in a dry state. However, safety or process limitations often lead to the presence of bound or free water in the waste stream.

Wastes can vary in density. Typical ranges that can be encountered are 0.5 to 5 g/cm3.

The wastes are poorly characterized. A methodology has been developed for measurements of mixed wastes, where the operator cannot define distinct `streams'.

The measurement container is large. Process and engineering requirements often mean that the waste can only be presented for measurement in large containers. This is undesirable from a measurement point of view as it increases the uncertainty in the measurement. Physicists have worked in close collaboration with plant designers to find the most appropriate measurement point in the process.

System Description

The conceptual measurement system for assay of highly active waste comprises a custom built chamber made up of polyethylene and graphite, suitable for both active and passive measurements (DDA and PNCC). Fast neutron detector packages are located in the walls of the chamber together with the pulsed neutron generator(s) for the active assay. Due to the gamma emissions associated with highly active materials, the chamber is lined with several centimetres of lead shielding to reduce the gamma flux to the detectors (which are sensitive to both gamma rays and neutrons).

Imaging and matrix corrections capabilities have been developed which allow accurate determination of the fissile content and spontaneous fission neutron emission rate of a variety of waste streams. It is possible to use these parameters to quantify individual trans-uranic nuclides (e.g. 239Pu) and fission products in the waste container. Sensitivities at or below gram levels of fissile material are achievable.

CONCLUSIONS

Characterizing a retrieved waste from an interim store which may have been operational thirty or more years ago represents a challenge to the instrument supplier. For a site such as the BNFL Sellafield facility, many such retrieved waste streams will be encountered as redundant facilities are cleaned out in preparation for decommissioning, each posing its own distinct set of problems. To all of this can be added the challenge of meeting the increasingly detailed requirements of the regulator and disposal site operator in terms of the degree of characterization required.

Detailed consideration of some of the needs for retrieved waste monitoring at Sellafield has demonstrated the value of utilizing instrumentation concepts and techniques originally developed in support of reprocessing and associated operations.

BNFL Instruments Ltd with its considerable experience of providing instrumentation systems for the characterization of spent fuel and freshly generated wastes, which have now been taken through several generations of increasing sophistication and reliability, has provided the parent company with the confidence to move forward in its waste retrieval program with the knowledge that tractable and economic solutions can be developed for all waste streams.

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Session 59 -- RISK ASSESSMENT: METHODS AND APPLICATIONS
Co-chairs: Tom H. Smith, Lockheed Idaho Technologies;
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59-1
RISK ASSESSMENT AS A DECISION MAKING TOOL FOR ENVIRONMENTAL RESTORATION AND WASTE MANAGEMENT

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ABSTRACT

This paper presents a human health risk assessment methodology for a large site that includes multiple facilities with solid and liquid waste disposal sites, which contribute contamination through several media such as air, groundwater, soil, and surface water. Risk-based decision making is evaluated for agricultural, residential, industrial, and recreational land uses. Also evaluated are risk-based decision making as it pertains to Native American land uses. The methodology includes a risk presentation that is useful for communicating with interested groups and valuable for decision makers. The human health risk is presented in isopleths overlaid on the region of analysis. This region of analysis may be the region within the site boundary or the region outside of the site boundary as far as necessary.

The risk assessment methodology is applied as a decision making tool. Decisions on whether to undertake a remedial action are evaluated based on parameters such as the extent of remediation and restoration achieved compared to target risk, the amount of risk reduction at varying cost levels, and the risk from various land uses before and after remediation. Several examples of the human heath risk contours are presented. The high-risk areas within an isopleth are traced back to the media, source, and the constituents that cause the high risk. Comparisons of various alternatives versus cost are presented. Finally, the feasibility of a single or multipurpose land-use strategy is evaluated based on human health risk and ecological risk for several remedial alternatives.

INTRODUCTION

Increasingly, risk assessment is being applied to pollution control and remediation decisions, particularly in the context of cost-benefit analysis and land-use planning. While there are many advantages in using risk assessment to prioritize remedial actions and risk reduction strategies, there are several inherent limitations in their ability to

addresses simultaneously the impact on human health, the ecosystem, and social culture. Some of the limitation are 1)numerous source terms and release patterns; 2) various constituents and environmental media; and 3)endpoint receptors.

Most risk assessments are limited to a single waste site or a closely integrated site grouping, such as the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) operable units. Thus, most risk assessments can be modeled using a single release term for each individual waste site to multiple receptors, such as the maximally-exposed individual (MEI), a member of the public, or a land user. For a facility with 100 waste sites and a receptor, the risk from each site to the same receptor has to be calculated individually (100 times), then added to all the risk values to evaluate the overall risk to the individual from the facility. A change in any parameters or components of the risk such as the source, location, or exposure pathway will require complete recalculation of risk. The methodology described in this paper is based on the Modular Risk Assessment (MRA). The methodology allows overall risk from widely scattered release sites throughout an installation with areas of thousand of km2 to be calculated and displayed graphically. The change in risk over time and anticipated remediation and post-remediation land uses can be easily calculated and displayed graphically. While the MRA methodology is applicable to human heath, ecological, and cultural risk, this paper discusses only the human health risk. The application of MRA methodology to the ecological risk is described in Duke et al. (1995). The application of MRA to the cultural risk is reported in Harper (1995).

Briefly, the methodology described in this paper breaks the facility into smaller, more manageable sections or cells by superimposing a map of the facility with a grid. This methodology uses a discrete and modular approach. This methodology has been discussed in several publications (Nazarali et al. 1994 and Whelan et al. 1994) and applied in two major Hanford Site Environmental Impact Statements (EIS). The contaminations, as a module, are identified and quantified in each cell for each medium (i.e., groundwater, soil, surface water, and air) using existing databases and environmental surveillance data. There lease and transport of a unit concentration, as a module, through different media to the receptor is modeled for several time periods ranging from the present to 10,000 years in the future. The unit risk factor, as a module, is calculated by using a unit of concentration to calculate the risk for several land-use receptors. The overall risk is calculated by multiplying the source term by the unit transport factor, at the selected time, to obtain the concentration of the contaminants at the selected time available to the receptor. The product of this concentration and the unit risk factor results in the risk from the particular constituent at the certain time to the receptor of concern.

The most important part of this methodology is its ability to present the results and information for environmental restoration and waste management decisions. The presentation is in contour form, which provides a great deal of information in a graphic form that is easy to assimilate. This presentation assists in thorough decision making due to the illustrated overall picture of the results. A thorough decision requires incorporating all the variables that are the integral part of the decision. The more typical tabular presentation of the risk based on single site and single receptor for a large facility with multiple sites

will not display all the decision making variables in a manageable format. The relative risk from one or all of the sources through one or all media to one or several receptors, which is essential to the decision makers, would be tedious and almost impossible if the single site single risk evaluation method were applied. In addition, presenting the results in tabular form would make the analysis much harder because of the volume of the data. The estimated risk by the MRA methodology can be communicated with the interest groups in a simple and clear manner. Decisions regarding the remediation of a site, the whole facility, or selected constituent based on the target risk, regulatory standard, or the stack holder cleanup goal can be analyzed and compared efficiently and easily.

METHODOLOGY

The methodology is divided into two primary sections, risk calculation and risk presentation. The risk calculation is based on the MRA. The risk presentation is time and spacial dependent. The risk results are presented in graphical form for a selected time and specific location. The risk calculation and presentation are discussed in the following sections.

Risk Calculation Methodology

Modular Risk Assessment Approach

The multiple waste sites with different waste forms at different locations that require assessment demand a more flexible approach, which ensures that the important aspects of the assessment are not lost through gross aggregation. The MRA approach allows for evaluating regional risk by accounting for the effects of widely scattered waste sites. The approach becomes powerful and relatively easy to apply because the system is automated. The MRA is based on the independent evaluation of each module that is the component of the risk. These independent modules are source, transport, and exposure, which are shown in Fig. 1 and discussed in the following sections.

The first approach in applying the MRA is to establish a spacial configuration that addresses the location of the waste sites. The location of any object can be addressed by Cartesian, cylindrical (polar), and spherical coordinates. In this paper, the Cartesian coordinates are used for the MRA approach. The x-axis represents the east-west coordinates, and the y-axis represents the south-north coordinates. The z-axis (perpendicular to the xy plane) represents the magnitude of the source, concentration, or risk. The size of the grid cells in xy- plane depends on the size and configuration of the facility. For a large facility such as the Hanford Site, the 1.0-km by 1.0-km grid cell would be proportionate. Smaller facilities may use as small as 100-m by 100-m or 10-m by 10-m grid cells. Figure 2 shows the 1.0-km by 1.0-km grid cells superimposed over a map of the Hanford Site.

Fig. 1

Fig. 2

Source Module

The source module represents the source term, which refers to the inventory or concentration of the pollutants that are available for release within the media such as the soil (surface and subsurface), groundwater, surface water, and air. Each cell contains a table for the source at the present time, as shown in Table I. The pollutants are divided into three groups of radionuclides, carcinogenic chemicals, and non-carcinogenic chemicals. The groundwater column represents the

inventory or concentration of each constituent within the groundwater. The soil column displays the inventory or concentration of each constituents in the surface soil (Z1), subsurface soil (Z2), and total soil (Z). The surface water and air columns represent the inventory or concentration of each constituents in surface water and air, respectively. For cells that contain several waste forms, the aggregated source term or concentration is used.

Table I

Prior to collecting data that specifically describes the waste at each site, the environmental setting and grid system must be determined. The installation is divided into regions where the geology, hydrology, hydraulics, and meteorology are considered to be representative of the region. Data describing each environmental setting are collected and verified as being representative of the area by installation engineers and scientists.

Transport Module

A unit of each constituent in each media is transported into the environment using the transport code. The Multimedia Environmental Pollutant Assessment System (MEPAS) was used for the Draft Hanford Remedial Action EIS (HRA-EIS) risk assessment analyses. MEPAS was chosen because it includes the following elements. MEPAS 1) addresses radioactive, organic, and inorganic wastes; 2) provides user flexibility in describing the geology, hydrology, hydraulic, and meteorology at an installation by allowing the use of site-specific data; 3)performs calculations within the installation boundary and offsite; 4) is largely based on the solutions to the advective-dispersive equations for solute transfer; 5)includes atmospheric complex terrain, channeling of wind, fugitive dust emissions, wet and dry deposition, and gaseous and particulate releases; 6) addresses both active and inactive sites and releases; 7)allows for user or code specific time-varying (i.e., transient) source-term emission rates; and 8) addresses contaminated soils, pond sites, liquid discharges, injection wells, and point, line, and area sources.

The unit source term for each media from each cell that contains the source are transported to other cells using transport code. The comprehensive methodology and application of unit environmental transport assessment of contaminants is described in Whelan et al. (1995). For example, in the groundwater transport the source from one cell in some cases was transported to several hundred cells. Therefore, a fraction of the unit source from one cell may appear in one or several cells. The contribution of all the sources from one cell or several cells to a cell at a selected time are gathered in a file (table) as the unit transport factor (UTF). The data have the same format as the source term, as shown in Table I. The values in this table are the fraction of unit source or concentration of the constituent of concern within the media of concern at a selected time for the particular cell. Each cell contains its own table with the specific UTF.

Exposure Module

This module calculates and represents the risk to a certain receptor (life style or land use) from a unit of concentration in groundwater, soil, surface water, and air for each constituents. Four land-use receptors are analyzed in the Hanford Site Risk Assessment Methodology (HSRAM) document (DOE 1995). Each of these receptors' unit risk factor is calculated based on their activity and life style. The four receptors

(agricultural, residential, industrial, and recreational) unit risk factors for radionuclides, carcinogenic chemical, and noncarcinogenic chemicals for all the media (groundwater, soil, surface water, and air) are described and calculated (Strenge and Chamberlain 1995). For example, the agricultural receptor uses all the media. This receptor uses the groundwater for showering, drinking, watering the garden, crops, and animals. This receptor uses the soil (surface and sub-surface) for raising a garden, crops, and animals. The surface water and air are part of this receptor's pathway. The risk from a unit of the concentration within each medium are gathered in a file (table) as the unit risk factor (URF). The URF data have the same format as the source term and UTF, as shown in Table I. The URF values have unit of risk (cancer incidence or fatalities) per concentration.

Risk Result

Each cell will have one risk table for every exposure scenario and time. The product of the source term and the UTF results in the concentration of the pollutant in that medium. The product of this concentration and the URF results in the risk to the selected receptors at the selected time for the constituent of concern at the particular cell. The resultant data are gathered in a file (table) as risk. These data have the same format as the source term, UTF, and URF. An example of the risk output is shown in Table II. The consistency in the source, UTF, URF, and risk tables format is for computation purposes. The separation hazard categories (radionuclides, chemicals carcinogenic, and chemical noncarcinogenics) are for risk summation. The resultant risk can be summed for each media, each constituents, and each hazard category. For each receptor (land user) at the Hanford Site, there are total of 2,700 (53 km by 53 km) risk tables generated (a table per cell).

The mathematical risk calculation can be expressed by Eq. 1.

Eq. 1

Where R, S, U, and E represent the total risk, source term, unit transport factor, and unit risk (exposure) factor, respectively. The subscripts i, j, k, and t represent the constituent, medium, receptor, and time.

Risk Presentation Methodology

In most analytical calculations, the computation results are presented in tables or some type of graphical form such as a pie or bar chart. For the Hanford Site, a comprehensive risk assessment would include 2,700 risk tables for a selected time. The analyses usually evaluate the risk for several time frames, such as T0, T1, T2, and T3 for each receptor, which results in 10,800 individual risk tables (2,700 4) for each receptor and a total of 43,200 tables (10,8004) for all four receptors.

In this methodology, the value of the risk for each cell is plotted on the z-axis. Therefore, for the Hanford Site all the 2,700 risk tables for a selected receptor at a selected time can be presented on one graph. The four land-use receptors and the four time frames of interest will result in a total of 16(4by4) contour maps. Figure 3 presents the risk to a residential farmer for 4 selected time frames (T0, T1, T2, and T3,)

ANALYSES

The methodology can be applied to the baseline, remediation, and post-remediation human health and ecological risk analyses. The baseline risk represents the risk to a selected receptor from all the sources prior to any remediation or restoration or for the no action alternative for the National Environmental Policy Act (NEPA) EISs. The remediation risk

refers to the risk that corresponds to the remedial activities. The post-remediation or residual risk refers to the risk from the pollutants that are left in the environment after completing the remediation activities. The flexibility and capability of this methodology makes it possible to conduct analyses such as comparing the baseline, remediation, and post-remediation risk for a several land-use scenarios. By subtracting the post-remediation risk from the baseline risk, the reduction in risk to the receptors can be analyzed. This reduction in risk can be compared against the risk from the remediation itself for further analyses. The risk at different times may be reduced as the time goes by so that remediation may not be required at all or may be required only for a certain waste form or land-use scenario. The combination of several land uses over the region of interest is a process of analyzing that is possible by using this methodology.

Fig. 3

Table II

Target Risk Analysis

The target risk mainly refers to the end point risk that is desired after completing remediation. For a selected target risk, a risk reduction factor (RRF) can be calculated by this methodology by comparing post-remediation risk with the baseline risk. For example, a typical tolerated exposure level for carcinogenic substance may increase the upper bound lifetime cancer risk from 1.0E-04 to 1.0E-06. This factor can be expressed as the ratio of the target risk to baseline risk as follows:

Eq. 2

The level of contamination remaining after remedial actions compared to the baseline can be expressed mathematically as a concentration reduction factor (CRF). This factor can be evaluated using target risk or target concentration. The RRF is directly proportional to the CRF, as follows:

Eq. 3

Decision Making

This methodology is a powerful tool for risk-based decision making. There are several ways to use this methodology as decision making tool. First, the processes of radioactive decay and biological degradation reduce the initial inventory of a contaminant over time. Therefore, presenting risk for different times in the future can communicate the need for remediation. Second, presenting the risk for the same time but different land uses such as unrestricted (agricultural and residential scenario), partially restricted (industrial scenario), and restricted (recreational scenario), can give enough information to the decision maker for selecting the remediation and extent of remediation that is necessary. Third, the combination of time and land use can assist the decision making processes, as this methodology demonstrates.

APPLICATION

This methodology has been applied in two major EISs at the Hanford Site.

Hanford Remedial Action Draft Environmental Impact Statement

The Hanford Remedial Action EIS evaluated and analyzed several alternatives regarding the remediation of over 1,100 past practice waste sites at the Hanford Site. In this EIS, a comprehensive baseline human health and ecological risk analysis was conducted. The risk assessment was performed using the methodology described in this paper. However, the remediation risk and the post-remediation risk were not the same as the methodology in this paper. The remediation assumed the complete removal of the all the sources and/or securing the waste sites by mean of

capping. This forced the analysis to assume that the post-remediation risk was nominal.

Tank Waste Remediation System Draft Environmental Impact Statement

The Tank Waste Remediation System EIS analyzed several alternatives for remediating 177 single and double-shell tanks. These tanks contain 56,000,000 gallons of high-level waste with total activity of 176,700,000 curies that have been generated from nuclear defense production activities. The post-remediation risk for this EIS uses this risk assessment methodology. The baseline risk was analyzed under the no action risk. This risk assessment methodology assisted in selecting a combination of alternatives for the Tank Waste Remediation System EIS.

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ACCIDENT ANALYSIS FOR THE LOW-LEVEL MIXED WASTE "NO-FLAME" OPTION IN THE U.S. DEPARTMENT OF ENERGY WASTE MANAGEMENT PROGRAMMATIC ENVIRONMENTAL IMPACT STATEMENT

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ABSTRACT

This paper outlines the various steps pursued in performing a generic safety assessment of the various technologies considered for the low-level mixed waste (LLMW) "No-Flame" option in the U.S. Department of Energy (DOE) Waste Management Programmatic Environmental Impact Statement (WM PEIS). The treatment technologies for the "No-Flame" option differ from previous LLMW technologies analyzed in the WM PEIS in that the incineration and thermal desorption technologies are replaced by sludge washing, soil washing, debris washing, and organic destruction. A set of dominant waste treatment processes and accident scenarios were selected for analysis by means of a screening process. A subset of results (release source terms) from this analysis is presented.

INTRODUCTION

This paper presents a preliminary assessment of potential accidents for the "No-Flame" option leading to airborne releases at U.S. Department of Energy (DOE) sites. The assessment is being developed in support of the Programmatic Environmental Impact Statement (PEIS) for management of low-level mixed waste by the Environmental Management (EM) Office of DOE. An important consideration in the WM PEIS is the risk to human health of potential radiological releases from facility accidents. An evaluation of facility accidents is a necessary first step in evaluating the risk of accidents to the on-site and off-site populations at each of the sites. This risk evaluation is part of the process of comparing alternative management strategies in the WM PEIS. These strategies include decentralization, regionalization, and centralization of waste treatment activities.

Low-level mixed waste contains both radioactive and Resource Conservation and Recovery Act (RCRA)-controlled substances. LLMW is generated, projected to be generated, or stored, at 37 DOE sites as a result of research, development, and production of nuclear weapons. It is projected that waste management activities will require management of an estimated 226,000 m3 of LLMW over the next 20 years.

A variety of treatment methods and processes for LLMW were considered in the WM PEIS. For difficult-to-treat LLMW containing organic material, two thermal treatment methods were analyzed: incineration, which EPA considers the best demonstrated available technology for organic waste, and thermal desorption, which bakes the waste at temperatures lower than those used in incineration. A "No-Flame" treatment process is being considered that replaces thermal treatment (incineration and thermal desorption) with sludge washing, soil washing, debris washing, and organic destruction technologies.

The safety documentation that exists for the washing and organic destruction technologies were reviewed to establish which technology may significantly contribute to the overall risk of waste treatment. The technologies were also examined to determine if one or more of the following accident conditions could exist:

1) Conditions which could result in large-scale damage or overpressurization of the various pieces of equipment, tanks, or vessels for each technology;

2) Ignition of flammable gases (including liquids and aerosols) that are always present or ignition after release of retained flammable gas/liquid/aerosol;

3) Process equipment failures which could result in an energetic release of radioactive material;
4) Suspension of radioactive materials by sprays, etc.

Based on the above criterion, it was determined that the accident analysis would focus on Organic Destruction (ORD), due to the potential of overpressurization (point 1), combustibility of the input waste stream (point 2), and energetic releases upon reactor rupture (point 3).

The organic destruction technology is similar to wet-air oxidation except that the organic concentration in the waste feed is significantly higher (greater than 50 percent). Organic destruction is the aqueous-phase oxidation of concentrated organic and inorganic wastes in the presence of oxygen at elevated temperature and pressure. Pressure in the range of 300 to 3,000 psi is used to maintain water in its liquid state, which allows oxidation to progress at lower temperatures than would be required for open-flame combustion. Water serves to moderate the oxidation rate by absorbing excess heat of reaction. Reactor temperatures typically range from 350o to 610oF (4,6). The layout of an ORD conceptual facility is presented in Fig. 1, based on (6)

Fig. 1

OVERVIEW OF FACILITY ACCIDENT ANALYSIS

The source term associated with an accident is the amount of radioactive material that is released to the immediate environment and is the product of four factors that vary for each radionuclide within the inventory affected by the accident:

Eq. 1

The material-at-risk (MAR) is defined as the inventory of waste impacted by an accident. The damage fraction is defined as the volumetric fraction of the MAR actually susceptible to airborne release. The RARF is the fraction of the total available radioactive material that is released and rendered airborne from primary confinement in a readily dispersible form. The LPF accounts for the reduction of the amount of airborne material due to containment, high-efficiency particulate air (HEPA) filtration, deposition, etc.

Determination of the Material-at-Risk (MAR)

The material-at-risk for the conceptual ORD facility is given by the summation of the major equipment and process piping:

Eq. 2

The various pieces of process equipment and their operating conditions were reviewed to establish which accident conditions would result in the largest airborne release, based on the present state of knowledge concerning the operation of the technology, potential failure modes, and radionuclide quantity present at the presumed time of failure. The material-at-risk associated with the process lines, heat exchangers, and other pieces of process equipment is neglected in this analysis, due to the presumed low volume of material associated with these items. The same argument is applied to the material-at-risk in the separator, due to the low-temperatures and pressures employed in this unit, as well as low volatility of the treated wastes.

The calculation of material-at-risk for the reactor and feed mix tank takes the following form:

Eq. 3

where TR is the treatment throughput rate (m3/yr) of the ORD facility, CONCi is the concentration of radionuclide "i" in the feed (Ci/m3), and t is the space time (residence time) of the reactor and the feed mix tank.

The treatment throughput rate and radionuclide concentration of the feed are obtained from the WASTE_MGMT computational model (1) and are a function of DOE site, treatment technology, and alternative site configuration.

LLNL (4) indicates that an organic destruction reactor would require a capacity of 540 gallons for a throughput of 5,177 kg feed per week, resulting in a space time of 33 hours (treactor ~ 33 hour). The space time for the feed mix tank is estimated based on equipment size data for a similar wet-air oxidation system (3) which indicates that comparable volumes of waste are contained within the reactor and feed mix tank. To assure a continuous flow of waste as feed to the reactor, the space time of the feed mix tank must be similar to that of the reactor (tfeed mix tanks ~ 33 hour) and therefore the material-at-risk for the feed mix tank is equivalent to that postulated for the reactor. The material-at-risk for the ORD facility is given by:

Eq. 4

based on 4,032 hours of operation per year.

DEVELOPMENT OF ACCIDENT SEQUENCES

A spectrum of accidents that occur during treatment were developed based on the waste's physical and radiological characteristics in conjunction with the technology specifications. They range from operational events (i.e., an overpressurization in the reactor chamber) to facility fires to external events (i.e., natural phenomenon events and airplane crashes). The accidents considered are discussed below.

Rupture of a Single ORD Reactor (accident sequence WAX)

Due to the similarity in processes between organic destruction and wet-air oxidation (WAO), the limited safety literature for the WAO process was reviewed to determine which accident sequences have been postulated to be risk-dominant. The worst-case internally-generated accident generally involved the rupture of the WAO reactor resulting from overpressurization and/or equipment failure (2). The ORD reactor operates at a pressure and temperature of approximately 250 psi and 260oC, respectively (4). If the reactor fails, the solution will flash to steam. The steam will be assumed to condense into particles of less than respirable size (10 micron AED) and be transported out of the ORD facility. The release to the atmosphere will be limited as the release is not energetic enough to breach the facility containment.

A review of the literature for tank and/or connecting pipe failures indicate that the failure rates depend primarily on 1) the design standard or basis when considering specific damage mechanisms, and 2) the inherent conservatism involved. As an example, large high-pressure vessels have a lower failure rate than low-pressure storage tanks. Failure rates between 1 x 10-4 to 1.3 x 10-3 / yr have been reported for various pressure vessels with a median value of 1 x 10-3 / yr. In this analysis, it is assumed that a failure rate of 1 x 10-3 / yr applies to breaches of the ORD reactor that could result in significant releases. The damage fraction for this sequence is based on the contents of a single ORD reactor. The contents of the three ORD reactors constitute 50% of the facility MAR, so that the contents of a single ORD reactor is about 16.7% of the facility MAR (one-third of 50%).

The respirable airborne release fraction (RARF) is the product of the airborne release fraction (ARF) and the respirable fraction (RF). The RARF for free-fall spill of the superheated aqueous solution in the ORD reactor is determined assuming isentropic expansion. The amount of the

ORD reactor contents that will flash to steam upon release was determined from the following:

Eq. 5

where HL1 is the enthalpy of the feed stream (1,265 Btu/lb at 250 psi and 260oC), HL2 is the enthalpy of the liquid (water) after release (180 Btu/lb at 1 atm and 100oC) and DHvap is the heat of vaporization at the release temperature and pressure (970 Btu/lb at 1 atm and 100oC). The mole fraction of vapor flashed is estimated from the above equation to be approximately 100%. Thus, all of the solution would flash (evaporate) to steam. The release factor from pressurized releases of superheated aqueous liquid solutions is given by (7):

Eq. 6

Based on a vapor mole fraction of 100%, the RARF is estimated to be 0.33. It should however be noted that this relatively high release fraction would be mitigated by the presence of double banks of HEPA filtration and moisture-condensing systems such as demisters, condensers, etc.

The characteristics of the WAX accident sequence are given in Table I. The value of the RARF shown in the above table only applies to nonvolatile particulate solid radionuclides (such as U-235, Pu-238 and other transuranics, etc.); a release fraction of unity is applied to noble gases and halogens.

Table I

Facility Fire (accident sequence WAF)

The ORD facility is designed to process organic and inorganic semi-solid and adsorbed materials and to destroy soluble materials such as heavy organic oils and emulsions (including chlorosolvents) produced during washing of sludges, soil, and debris. The high organic feed streams are diluted to 5 percent organic or less prior to injection into the reaction vessel. A low-order detonation of the organic waste and oxidant has been postulated to occur; however, the reaction chamber would be designed to avoid detonation (4). In this analysis, it is postulated that a fire occurs outside the ORD feed mix tank following leakage. A fire caused by ignition of combustible solvent would disperse radioactive particulates in the immediate area of the fire and would last for a short period because the amount of combustible material is limited. Due to the high structural integrity of the ORD reactor as well as the dilute aqueous nature of its contents, it is assumed that its contents would be unaffected (i.e., not released in significant quantities) by this accident sequence. (However it may be expected that its continued operation would be impaired.) The accident is presumed to be initiated by failure of the feed mix tank resulting in a large pool of organic liquid, fine particulates, etc. on the ground. This leakage from the feed mix tank is ignited by an electrical short, etc. It is conservatively assumed that all of the tank contents are spilled and burn. The release to the atmosphere will be limited due to the fire protection capabilities of the facility and the assumption that the release would not be energetic enough to breach the facility containment.

A wide range of initiating fire frequencies has been reported in recent NEPA literature, ranging from 7 x 10-4 to 2.0 x 10-2 / yr with a median value of 5 x 10-3 / yr. In this analysis, it is assumed that an initiating frequency of 5 x 10-3 / yr applies to a fire in the ORD feed mix tanks that could result in significant releases.

The damage fraction for this sequence is based on the contents of a single ORD feed mix tank. The contents of the ORD feed mix tank constitute 50% of the facility MAR, leading to a damage fraction of 50%. The input feed to the ORD reactor is stated to be diluted to 5 percent organic or less. The feed to the ORD facility has been categorized in the WM PEIS as an Organic Combustible Solution in an Aqueous Solution. Because a large percentage of the input liquid is aqueous in nature, the behavior of the feed mix tank contents upon application of a thermal stress was considered to be consistent with that of a boiling Aqueous Solution with droplet formation, for a RARF of 1 x 10-2 (7). The characteristics of the WAF accident sequence are given in Table II.

Table II

External Events

External challenges to the ORD facility include airplane impacts and natural phenomenon. The representative natural phenomenon analyzed is a seismic event because of its potential to affect the entire facility. A seismic event is postulated to rupture fittings/connections to the ORD reactors, resulting in aerosol formation. It was however not assumed that this would result in a small fire affecting the facility MAR. The contribution associated with the feed mix tank was neglected as a beyond-design basis seismic event could fracture the concrete footings under the holding tanks, allowing any spilled material to be absorbed by the soil, with negligible atmospheric releases. The accident frequency for seismic events is estimated on the performance goal for a Moderate Hazard facility, as defined in DOE guidelines.

Aircraft impacts were also analyzed as potential man-made external events. Aircraft accident frequencies are site dependent and were obtained from aviation statistics and the locations of DOE sites with respect to major airports and aviation routes.

Functional event trees specific to the organic destruction technology were developed to track the progression of the external accident initiators out to the point of airborne release. Initiating accident frequencies and conditional probabilities of the various event tree branches were determined from applicable safety literature where possible. Further information on development of the external event sequences is available in (5). The assumed characteristics of the various external accident sequences are given in Table III.

Table III

RESULTS AND DISCUSSION

The results of the accident analysis were obtained in the form of a detailed source term and an associated estimated annual frequency. The accidents have been grouped into four categories on the basis of their estimated frequency, with the categories ranging from anticipated (frequency higher than 10-2 per year) to extremely unlikely (frequency less than 10-6 per year) events.

Table IV provides a sample results with detailed information about the risk-dominant accidents summed over all radionuclides released, including the volume of the material-at-risk (VMAR., in m3), the material-at-risk (MAR, in Ci), total release fraction (TRF), source term (in Ci), accident frequency, and frequency class. The total release fraction is the product of the leak path factor(LPF), damage fraction (DF), and the respirable airborne release fraction (RARF). Only one WM PEIS alternative, number 36, which involves treatment at 12 sites, is shown because of space restrictions.

Table IV

The results in the table suggest that, in general, the risk of releases from the ORD facilities due to accidental causes would be low. Preliminary screening estimates confirmed that the risks to human health involved in LLMW management for the "No-Flame" option would be relatively low. Generally, releases of large amounts of radioactivity are associated with a very low estimated frequency, while more frequent events potentially result in small releases. The relatively low health impacts are the result of a number of factors including less severe operating conditions, absence of a fuel source such as natural gas used in incineration, and dilution with water of the product stream from organic destruction. All these factors may be expected to contribute to a lower health impact in comparison with incineration.

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59-3

WASTE MANAGEMENT FACILITY ACCIDENT ANALYSIS (WASTE_ACC) System: SOFTWARE FOR ANALYSIS OF WASTE MANAGEMENT ALTERNATIVES*

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ABSTRACT

This paper describes the Waste Management Facility Accident Analysis (WASTE_ACC) software, which was developed at Argonne National Laboratory (ANL) to support the U.S. Department of Energy's (DOE's) Waste Management (WM) Programmatic Environmental Impact Statement (PEIS). WASTE_ACC is a decision support and database system that is compatible with Microsoft Windows. It assesses potential atmospheric releases from accidents at waste management facilities. The software provides the user with an easy-to-use tool to determine the risk-dominant accident sequences for the many possible combinations of process technologies, waste and facility types, and alternative cases described in the WM PEIS. In addition, its structure will allow additional alternative cases and assumptions to be tested as part of the future DOE programmatic decision-making process. The WASTE_ACC system demonstrates one approach to performing a generic, systemwide evaluation of accident risks at waste management facilities. The advantages of WASTE_ACC are threefold. First, the software gets waste volume and radiological profile data that were used to perform other WM PEIS-related analyses directly from the WASTE_MGMT system. Second, the system allows for a consistent analysis across all sites and waste streams, which enables decision makers to understand more fully the trade-offs among various policy options and scenarios. Third, the system is easy to operate; even complex scenario runs are completed within minutes.

INTRODUCTION

This paper builds on earlier work that developed the computational framework for identifying risk-dominant accident sequences (1,2). Only atmospheric releases are considered, and risk is assessed in terms of the airborne source terms, measured in Ci, along with the frequency of the associated accident sequence. The framework employs a probabilistic analysis for potential radiological accidents at DOE facilities that manage low-level waste (LLW), low-level mixed waste (LLMW), transuranic waste (TRUW), and high-level waste (HLW). The framework considers both waste storage and waste treatment, as well as treatment technology options including incineration, vitrification, and wet-air oxidation. In addition, it considers internally initiated industrial-type accidents that may occur during waste management activities (e.g., breaching a waste drum during handling and rendering a portion of the contents airborne) and externally initiated accidents (e.g., earthquakes or airplane strikes). The WASTE_ACC system implements this framework to provide policy analysts and researchers with an easy-to-use tool.

The WASTE_ACC system generates a consistent evaluation of accident risks across all 52 potential waste management sites within the purview of the WM PEIS and across the various waste streams (i.e., LLW, LLMW, HLW, and TRUW). Many of the input and process parameters used by WASTE_ACC were developed specifically for the WM PEIS, and these data reside in comprehensive generic databases that are an integral part of the WASTE_ACC system. The WASTE_ACC software uses a state-of-the-art database that incorporates the latest accident information from DOE safety documents. To date, no other system can provide the consistent analysis across sites and waste streams required in the WM PEIS. Furthermore, no other model utilizes site-specific waste volumes and radiological profiles, as developed through ANL's WASTE_MGMT system.

Although WASTE_ACC was developed to support the WM PEIS, any site contemplating processing radioactively contaminated waste could use the system to obtain at least a first-cut analysis. All that would be required to perform site-specific analyses is further development of supplemental site-specific databases. Besides being used for waste treatment facilities, WASTE_ACC's probabilistic approach could be used for other types of facilities, such as hazardous waste incinerators or chemical processing plants, and for other types of facility accidents.

WASTE_ACC CAPABILITIES

WASTE_ACC can be used to:

- Determine the risk-dominant accident sequence as a function of treatment site, waste management alternative, and waste type;
- Perform preliminary calculations of the health effects of the postulated accident sequence for four receptors: off-site maximally exposed individual (MEI), off-site population, on-site MEI, and on-site population;
- Develop the progression of accident sequences for external initiators (such as seismic events or airplane crashes) and calculate the probabilities of accident progression along various event tree branches;
- Print records associated with the risk-dominant accident sequences to standard or laser printers; and
- Generate ASCII text output files containing airborne release data by nuclide for risk-dominant accident sequences.

The sections below describe the WASTE_ACC software.

WASTE_ACC DEVELOPMENT

The programming challenge was to construct an easy-to-use PC-based system with the capacity to process large amounts of data and the flexibility to accommodate various waste management alternatives, waste streams, and site-specific information. The solution implemented an application shell (based on the Microsoft FoxPro for Windows database system) around the framework that had been developed earlier. This solution has two main advantages. First, FoxPro provided the software tools necessary to develop a Windows-compatible graphical user interface, thereby enabling a user to manipulate the model by means of a familiar metaphor. Second, the development of an application shell around the framework integrated several stand-alone modules that previously constituted the accident analysis system into a single application. The system can now be run from a single screen with a few mouse clicks, and run-time has been reduced by more than 75%.

WASTE_ACC REQUIREMENTS

The system requires a DOS-compatible computer running Microsoft Windows 3.1, a microprocessor equivalent to a 33 MHz 486SX or greater, and at least 4 MB of RAM. The software itself requires about 2 MB of disk space for installation (beyond that required for FoxPro). The WASTE_MGMT input data files can take up to 500 MB for all alternatives and waste types. Furthermore, the system can require as much as 300 MB of additional free disk space to run. Scenario runs generally take between 1 and 10 minutes on a PC powered by a 90-MHz Pentium, depending on the waste type and the complexity of the alternative being considered. Decentralized cases require more computational time and disk space than centralized alternatives because they have more sites to be analyzed. LLMW alternatives require longer execution times because they have more waste subtypes.

WASTE_ACC OPERATION

Figure 1 presents WASTE_ACC's welcome screen. After clicking on the Continue button, the system's main screen appears, as presented in Fig. 2. These screens illustrate the latest improved user interface. Buttons, list boxes, and text fields are conveniently and logically arranged on the screen. These input mechanisms address the information requirements for running an alternative clearly and in the English language. All information is requested before a model run begins. In contrast, earlier versions used menus that popped up on the screen while the program was running. These menus required the user to have considerable knowledge of the executing module to select the correct option.

Fig. 1

Fig. 2

Besides the user interface, the model's structure has also been updated. The updated version of WASTE_ACC integrates numerous stand-alone modules into a single application that runs to completion at the click of a button. Stand-alone modules had been necessary to cope with the extremely large data files and complex data manipulation steps inherent in the model. However, improved computer resources and streamlined code enabled the system to be integrated into a single application. The integration greatly reduced processing turn-around time and improved the system's integrity and reliability. A run that used to require four separate programs and some human intervention can now be completed with a few mouse clicks in a fraction of the time. The next few paragraphs describe the main screen in more detail, and discuss how to run WASTE_ACC.

User Input

The screen displayed in Fig. 2 is divided roughly into three parts top, middle, and bottom. In the top part of the screen, the user selects which waste stream and alternative to run and the desired output options. The middle of the screen contains an area that displays which module is running so the user can monitor a run's progress. It also contains buttons to start model execution or to exit the program. A check box labeled "Batch" enables the user to set up a series of runs all at once. In the bottom part of the screen, users can tell WASTE_ACC where to find input files and where to locate working (scratch) disk space. The top part is described first.

In the first field, under the phrase "Waste Stream:," the user can select a waste stream to be analyzed from a list box. This box provides a complete list of the possible choices, from which the user selects one. The waste stream choices are: High Level Waste, Low Level Waste, ER Low Level Waste, Low Level Mixed Waste, ER Low Level Mixed Waste, Transuranic Waste, and ER Transuranic Waste. In the field below, labeled "Alternative:," the user selects from another list box containing the possible alternatives for the selected waste stream.

After the alternative has been selected, the user can choose output options. Regardless of the options selected, WASTE_ACC always produces a summary data file that is sent to the output directory. If the "Summary Reports" check box is selected, as it is in Fig. 2, the system sends a formatted summary report to the printer and to the screen, as shown in Fig. 3. If the "Site/Nuclide Reports" check box is selected, the program prints reports that detail source terms by site and nuclide for each risk-dominant accident sequence. Finally, if the "Source Term Files" check box is selected, a source terms data file is sent to the output directory.

Fig. 3

The user provides file location information by clicking on the appropriate button in the lower part of the screen. When the user clicks on the "Data" button, a standard Windows file-open dialog box appears. The user selects a directory to tell WASTE_ACC where to find various support and input files, except for the WASTE_MGMT data (as of Version 4.3 in October 1995). Similarly, by clicking on the "Output" button, the user can tell the program where to put the output files. Files can be directed to any available drive, and not all the drives need to be the same. The user can select the "Scratch" or temporary drive, where temporary files will be placed. In most cases, this should be the user's own hard drive to minimize network traffic and increase processing speed, but the extremely large capacity needed to run the model (more than 300 MB free space) may prevent this. As an added feature, the user does not have to select a button to specify a drive but may instead type the directory name directly into the field. Finally, the user tells the system where to find the WASTE_MGMT files by typing a drive letter next to the phrase "Network Drive for WASTE_MGMT files:". This drive may be the user's own hard drive (i.e., C), but because the size of the WASTE_MGMT files needed by WASTE_ACC is so large (more than 500 MB), the drive is most likely to be that of a file server.

Performing Scenario Runs

WASTE_ACC can perform runs one at a time, sequentially, or grouped together in a batch. To run a single alternative, the user simply makes the appropriate waste stream, alternative, and directory choices and then clicks on the "Run" button. As the model runs, small message windows appear on the screen to indicate WASTE_ACC's progress. Besides messages, WASTE_ACC displays the name of the executing module in the main screen next to the word "Module:" to tell the user exactly which stage the model is executing. Then, when an alternative has finished processing, WASTE_ACC alerts the user with a beep and a message. The user has the option to either quit the program or make additional (sequential) model runs.

When several alternatives must be run, the user may elect to group them together in a "batch" run to significantly reduce processing time. To initiate a batch run, the user clicks on the "Batch" check box, and a screen similar to that shown in Fig. 4 appears. The user chooses which alternatives to run by typing a "T" in the "T/F" (last) column of the screen. Then, when the model runs, WASTE_ACC processes all of the designated alternatives and places the resulting source term files in the output directory as specified on the main screen.

Fig. 4

After the user has clicked on the Run button, WASTE_ACC sets up the work space, which in this context means it closes any files that may be open from earlier model runs. Next, the system locates necessary input files for the waste stream of the scenario being processed. When all the files are located, the computations begin. First, the quantity and characteristics of the material at risk (MAR) are calculated. After the MAR file is built, the system assigns unit dose conversion factors (DCFs), provided by Oak Ridge National Laboratory, to the MAR on the basis of its characteristics. Next, the system develops the accident parameters (such as accident frequency,damage fraction, and conditional probability) for each site, accident initiator, and sequence. These accident parameters are then linked to the MAR, and the model computes releases, doses, risks, and consequences for all of the accident

sequences. Finally, the system ranks accident sequences by risk and then constructs a file to contain radiological source terms for sequences with the largest impacts. The system writes this file to the output directory, and the computations are complete.

If the model is processing a single run, the system displays the results in a series of report screens (see Fig. 3). After all of the results have been presented, the application returns to the main screen so the user can choose the next action. For batch runs, the system does not present results on the screen since this would interrupt the program's operation. Instead, as the program loops though each of the designated alternatives, it writes the results to the output file described above. When all of the alternatives are done, the main screen appears, and the user regains control.

Output

The final outputs of WASTE_ACC are site-specific atmospheric source term results and accident summaries. One type of source term result is on the amount of radioactivity released to the atmosphere per radionuclide of each accident initiator's risk-dominant sequence. The accident summaries (partially shown in Fig. 3) provide detailed information about the risk-dominant accidents, summed over all radionuclides released, which include data on the:

- Volume of material at risk (m3),
- Material at risk (Ci),
- Air-release source term (Ci),
- Dose to the off-site MEI (person-rem),
- Cancer risk of the off-site MEI (incidents/yr),
- Risk to the off-site MEI (rem/yr),
- Doses to off-site population (person-rem),
- Number of excess latent cancer fatalities in the off-site population,
- Accident frequency (incidents/yr), and
- Frequency class.

Future Development

Although WASTE_ACC has come a long way toward becoming an easy system to use, much work remains. The system focuses primarily on DOE Waste Management issues and capabilities, but with some data development, it could perform accident analyses for other systems, such as hazardous waste incinerators or chemical processing plants. Moreover, the large and complex databases that support WASTE_ACC could have tools, such as input screens and preprocessing routines, that would make data collection and entry easier and less error-prone. The system could be adapted to perform detailed analyses on sites by using even more site-specific data than are available or appropriate for a nationwide analysis. Finally, the system could be adapted to quantify the uncertainty for the computed risk values. The model would account for uncertainties in the input data and process parameters, so instead of computing a point estimate, the model would produce a probability distribution reflecting the uncertainty of the system.

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59-4

A PROPOSED ALTERNATIVE APPROACH FOR PROTECTION OF INADVERTENT HUMAN INTRUDERS FROM BURIED DEPARTMENT OF ENERGY LOW LEVEL RADIOACTIVE WASTES*

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ABSTRACT

The burial of radioactive wastes creates a legacy. To limit the impact of this legacy on future generations, we establish and comply with performance objectives. The most controversial of these performance objectives is the requirement to protect possible future inadvertent human intruders (IHIs). This paper summarizes the regulatory approaches designed to protect future IHIs from buried radioactive wastes and proposes an alternative approach for protecting the IHI.

Standard exposure scenarios for the burial of U.S. Department of Energy (DOE) low-level radioactive wastes (LLW) assume that inadvertent human intrusion will occur, i.e., the probability of intrusion is one. A number of events must proceed inadvertent exposure to buried waste, including loss of active site control, loss of knowledge of the site, loss of site characteristics (the burial site must not resemble a burial site), loss of waste characteristics (the waste must not resemble waste), and inadvertent intrusion must occur.

Assuming that the probability of inadvertent intrusion is one is conservative, but does not account for important site characteristics. Site characteristics are included in the proposed alternative approach which is based on the "expected dose" to the IHI. The expected dose is the product of the consequence of intrusion and the probability of intrusion. For example, if the consequence of intrusion is calculated to be 125 millirem per year chronic dose and the probability of intrusion is assessed at 0.65, the expected chronic dose to the IHI is 81 millirem per year. All aspects of calculating the consequence of human intrusion are identical to calculating the consequence under DOE 5820.2A; the only new element is the derivation of the probability of intrusion.

Consequence is calculated using the long-standing exposure scenarios for intrusion. The probability of intrusion, where probability is a reasoned belief, can be assessed a number of ways and the expert elicitation process is recommended. A site-specific assessment of the probability of inadvertent intrusion allows credit to be taken for favorable site characteristics, such as site aridity, depth to water, and past site use, that are not explicitly accountable when the probability of intrusion is assumed to be one.

INTRODUCTION

The burial of radioactive wastes creates a legacy. To limit the impact of this legacy on future generations, we establish and comply with performance objectives. This paper reviews performance objectives for the long-term isolation of buried radioactive wastes; identifies regulatorly-defined performance objectives for protecting the inadvertent human intruder (IHI) from buried low-level radioactive waste (LLW); (3) discusses a shortcoming of the current approach; and (4) offers an alternative approach for protecting the IHI. This alternative approach is written specifically for the burial of U.S. Department of Energy (DOE)

wastes at the Nevada Test Site (NTS), although the approach might be applied at other DOE burial sites.

PERFORMANCE OBJECTIVES FOR BURIAL OF LLW

There are typically three performance objectives for the isolation of buried radioactive wastes:

- the long-term protection of human health from wastes that might migrate from the disposal site;
- the long-term protection of individuals that might inadvertently intrude into the disposal site (e.g., water well drillers);
- the long-term stability of the disposal site after closure to eliminate the need for active maintenance.

A possible fourth objective, protection of deliberate intruders, is typically considered infeasible (15).

In summary, performance objectives are defined to keep the wastes from getting to people and keep people from inadvertently getting to the wastes. These goals are expressed as two types of performance objectives: protection of the Member of Public (MOP) and protection of the IHI. This paper addresses protection of the IHI.

There are three categories of measures to protect the IHI from buried LLW: 1) active institutional controls, such as fences and guards; 2) passive institutional controls, such as deed restrictions, long-term government ownership, and long-lived markers; and 3) engineered barriers, such as special waste forms, burial depth, and special materials incorporated into the site closure system (15).

There are many who believe that DOE will maintain control of DOE disposal sites forever, thereby eliminating the possibility of an inadvertent human intrusion. The intent of the U.S. government is to maintain control of its high-level radioactive waste (HLW) burial sites forever (35 FR 17533) and there are formal discussions of maintaining control of the LLW burial and cleanup sites in perpetuity. However, there is significant concern that over the next few thousand years, control will be (at least temporarily) lost. For example:

- The U.S. Environmental Protection Agency (EPA) "... believes that passive institutional controls can never be assumed to eliminate the chance of inadvertent and intermittent human intrusion ..." (40 CFR 191, Appendix B);
- From guidance developed for the U.S. DOE "...survival of engineered systems may not be credible over time frames beyond 1 or 2 millennia." (15, p. 10);
- The National Research Council stated in 1995 that: "... it is not reasonable to assume that institutional controls can be maintained for more than a few centuries, we also conclude that there is no scientific basis for assuming that human activity can be prevented from occurring in an exclusion zone ..." (7, p. 122);
- The U.S. Nuclear Regulatory Commission (NRC) stated that: "... inadvertent human intrusion into a closed disposal facility at some point after closure of the disposal facility is likely." (9, p. 4-53).

With the failure of these protective mechanisms, the future may include IHIs receiving high doses of radiation from buried radioactive wastes. If future conditions could be anticipated, it would be possible to protect future generations. However, future conditions are uncertain. Despite this uncertainty, it is still possible to make intelligent decisions to protect the IHI. Regulations governing the protection of IHI typically define a set of hypothetical conditions. As a society (i.e., by

rulemaking), this country has established that compliance with these hypothetical conditions is protective of the IHI.

REGULATORY BACKGROUND

Several standards govern the disposal of radioactive waste in this country, and each of these standards establishes a different set of requirements that address the IHI. However, each standard has the same underlying assumption; that future generations maintain current living habits. This assumption is probably incorrect for the next 10,000 years; in just the last 60 years, our children have migrated from the yards and creeks to the video arcades and malls.

In the U.S., three major standards set requirements for the disposal of radioactive waste:

Commercial LLW is governed by 10 CFR 61; DOE-titled LLW, and other DOE radioactive wastes, are governed by DOE Order 5820.2A; and

40 CFR 191 sets requirements for the disposal of transuranic (TRU) wastes, HLW, and spent nuclear fuel.

The disposal of TRU wastes at the Waste Isolation Pilot Plant (WIPP) is governed by 40 CFR 191 with site-specific requirements set by 40 CFR 194. For the disposal of spent nuclear fuel and HLW, the WIPP Land Withdrawal Act exempted Yucca Mountain from the requirements of 40 CFR 191, and the Energy Policy Act of 1992 directed the U.S. EPA to establish site-specific standards for Yucca Mountain. These site-specific standards will define a fourth major regulation and are currently under development (60 FR 47172). These three standards (10 CFR 61, DOE Order 5820.2A, and 40 CFR 191) are briefly discussed as below.

Standards for Commercial LLW Burial

The NRC's 10 CFR 61 sets licensing requirements for the shallow-land burial of commercial LLW. In developing 10 CFR 61, the NRC recognized that the greatest risk of inadvertent exposure to buried LLW will be far in the future. If the nature of possible future exposures were knowable, specific measures could be taken to protect future intruders. However, the nature of any future exposure is uncertain. To proceed in the face of this uncertainty, the NRC chose to define surrogate, or hypothetical, IHI exposure scenarios. Protection of the IHI under the conditions defined in these hypothetical exposure scenarios would, by legal definition, be protective of future IHIs.

Defining the characteristics of the surrogate IHI exposure scenarios cannot be accomplished by science alone and requires policy judgments. NRC formalized these policy judgments through the rulemaking process. NRC held a series of four regional workshops, worked with the EPA and public input, developed draft and final Environmental Impact Statements (EIS) (9,10), and corresponding proposed rule (8) and final rule (10 CFR 61). Because the NRC was setting disposal standards for LLW burial sites not yet defined, it chose to protect the IHI by developing a generic set of waste classes. These classes are termed generic, because they were developed independent of disposal site characteristics and are based on a standard set of exposure scenarios defined through the rulemaking process. A similar approach is currently being taken by the EPA to set cleanup standards for federally-owned sites contaminated with radioactive materials (13).

The NRC's generic analysis was based on the following set of beliefs or concerns:

The potential for inadvertent human intrusion is likely. Extensive intrusion activities (such as major apartment construction) are unlikely.

There is a limited time (e.g., 500 years) in which natural and engineered barriers can be expected to last.

As long as the waste is structurally stable (i.e., is recognizable as waste), extensive inadvertent human intrusion is not considered credible. Based on these beliefs, the NRC established performance objectives to protect the MOP from contaminated groundwater and to protect the IHI. Establishment of these performance objectives was complicated and required consideration of five elements:

The length of institutional control;

The time at which the intrusion would occur;

An appropriate dose standard for the IHI;

The probability of the intrusion occurring; and

The specific exposure scenario(s) for the IHI.

Performance objectives are an interrelated "package deal." Altering one of the components may change the overall protectiveness of the rule. A short summary of each of these five objectives is provided below.

Length of Institutional ControlThe NRC selected 100 years. "...the Commission believes that it is not a question of how long the government can survive, but how long they should be expected to provide custodial care." (10, vol. 1, p. 5-27)

The Time at Which the Accidental Intrusion Would OccurNRC made the conservative assumption that knowledge of the site would be lost when the cognizant government control was relinquished (100 years after closure). This loss of knowledge was assumed to be a temporary, bureaucratic error.

An Appropriate Dose Standard for the IHIThe NRC selected 25 millirem (mrem) per year as the dose standard for the MOP and 500 mrem/year as the dose standard for the IHI. Setting an IHI standard 20 times greater the MOP dose standard was justified, because the MOP standard assumes continuous exposure to radionuclides by populations, whereas the IHI would not be routine and would only involve a few people.

The Probability of Intrusion OccurringIn the draft of 10 CFR 61, the NRC assumed that the probability of IHI is one (IHI will occur). The assumption was conservative (i.e., easy to defend) and not costly (i.e., the final 10 CFR 61 allows the shallow land burial of 99% of the commercial LLW (52 FR 5999)).

In response to the draft rule, many reviewers were concerned that the probability of an intrusion occurring was too high (i.e., not reflective of what the future would probably bring), and that the probability of intrusion should be incorporated into the calculation. The NRC had several responses. First, it is very difficult to set a numerical value on the probability that an intrusion event will occur, and to estimate the extensiveness of the intrusion. Second, in the final EIS, the NRC responded by increasing (liberalizing) the concentration-based limits on Class C wastes by an order of magnitude (from 10 to 100 nanocuries per gram) to "eliminate unnecessarily conservative assumptions." Additionally, the IHI is based on reasonably conservative scenarios (discussed below), and finally, the dose standard for the IHI is 20 times higher than that for the MOP. So, rather than attempt to assign a probability to the IHI, the NRC "loosened" other components of the regulatory package.

Specific Exposure ScenariosThe NRC developed three generic waste classes, Classes A, B, and C. The NRC states "[t]he (IHI) events are conservatively assumed to occur based on consideration of typical human

activities. NRC has assumed reasonably conservative (but not overly conservative) actions on the part of the intruder" (10, Vol. 1, p. 4-13). The exposure scenarios included the intruder-construction scenario and the intruder-agriculture scenario. The intruder-construction scenario involves the hypothetical exposure of workman involved in the construction of a house, with a basement, directly on the disposal facility, contacting and dispersing the waste. The intruder-agriculture scenario involves individuals that live in the house constructed in the intruder-construction scenario, who consume food grown in a small on-site garden which contains some of the soil from the basement excavation. These scenarios are described in great detail (cubic yards excavated,percentage of excavated soil used in the garden, etc.) in the Draft EIS, Vol. 4, beginning on page G-57 (9).

For each of the two intruder scenarios, for each of the four hypothetical, regional disposal sites, three sets of assumptions were made, corresponding to the three classes of wastes. Analysis of these exposure scenarios does not consider site-specific factors that might reduce the probability of the intrusion occurring and/or might increase the probability of the waste being recognized as waste.

In summary, 10 CFR 61 regulations apply to the burial of commercial LLWs and define generic classes of wastes where the waste classification is defined to protect the hypothetical IHI. The regulations not take into consideration site-specific burial site characteristics.

DOE Order 5820. 2A

DOE Order 5820.2A sets policy for DOE's management of radioactive wastes. Like the NRC's 10 CFR 61, DOE chose to establish separate dose standards for protection of the MOP and the IHI from buried LLW.

To protect the IHI, the original versions of 5820.2A had a classification system similar to the NRC's Classes A, B, and C, with some exceptions. However, DOE did not adopt the NRC system, because 1) the DOE waste streams were different from the NRC's commercial LLW streams; 2) in some cases, the DOE standard would be inconsistent with the NRC standard; and 3) the NRC standard was generic, not site-specific. Because DOE already knew the locations of its active, LLW disposal sites, DOE chose to require site-specific performance assessments to demonstrate protection of the IHI, as well as the MOP.

The boundary between standardized assumptions and site-specific considerations is not defined in DOE Order 5820.2A. General guidance is provided by the DOE Performance Assessment Task Team (14,15) and the DOE Performance Assessment Peer Review Panel. The Performance Assessment Task team recommends that "standardized" DOE exposure scenarios include (15):

An acute construction scenario and a chronic agricultural (homesteader) scenario involving excavation into disposal units, mixing exhumed waste in an intruder's vegetable garden, and permanent residence in a home on top of disposal units;

An acute discovery scenario and a chronic residential scenario involving an attempted excavation into disposal units, which is assumed to be precluded by the presence of intact engineered barriers, and

An acute drilling scenario and a chronic post-drilling scenario involving drilling a water well through a disposal units and mixing the drilling wastes in an intruder's vegetable garden.

Like the NRC's generic scenarios, DOE Performance Assessments (PAs) have not taken site-specific credit for the probability of IHI and have assumed the probability of intrusion is one. In summary, DOE's Order

5820.2A governs the disposal of DOE radioactive wastes, and the specific performance objectives for disposal of LLW are similar to the NRC's 10 CFR 61, with some important exceptions.

Regulations for Disposal of Spent Nuclear Fuel, HLW, and TRU Wastes

The EPA's 40 CFR 191 sets standards for the management and disposal of spent nuclear fuel, HLW, and TRU radioactive wastes. Subpart B of 40 CFR 191 contains one requirement (assurance) and three performance objectives: protection of the individual, protection of groundwater, and containment.

Protection of the individual and protection of groundwater are both "undisturbed performance" standards and specifically exclude the IHI. The containment standard (40 CFR 191.13) requires the assessment of all events and processes that may disturb the disposal system, including inadvertent human intrusion.

The containment standard is a 10,000-year, probabilistic standard. The containment standard is based on an abstracted ratio between the original inventory of buried TRU radionuclides and the inventory of radionuclides that might move beyond the controlled area (a legally defined term). The containment standard does not consider the dose to the IHI; only the abstracted ratio is important. A repository might pass the containment standard even if the IHI hypothetically received a lethal dose from an "industrial accident."

In a PA for TRU wastes, a "scenario" is a mutually exclusive combination of features, events, and processes, which is very different from an "exposure scenario" as used in an LLW PA. The EPA requires the assessment of site-specific scenarios for disposal sites governed by 40 CFR 191. The EPA recognized there would be uncertainty in assessing features, events, and processes for the next 10,000 years, and offered significant guidance in Appendix B of 40 CFR 191.

" ... inadvertent and intermittent intrusion by exploratory drilling for resources ... can be the most severe intrusion scenario assumed by the implementing agencies...the Agency assumes that the likelihood of such inadvertent and intermittent human drilling need not be taken to be greater than 30 boreholes per square kilometer of repository area per 10,000 years for geologic repositories in proximity to sedimentary formations, or 3 boreholes per square kilometer in other geologic formations."

In summary, 40 CFR 191: 1) applies to TRU, HLW, and spent nuclear fuel wastes; 2) does not specifically attempt to protect an intruder; 3) requires the assessment of the probabilities of all future events and processes; and (4) offers significant EPA guidance concerning the probability of future inadvertent human intrusion.

ALTERNATIVE APPROACH FOR PROTECTION OF INADVERTENT HUMAN INTRUDERS

The previous sections of this paper highlighted the development of the IHI standard. In this section, important assumptions are reviewed, consequences of these assumptions are discussed, and an alternative approach is proposed.

The NRC and the DOE make the generic assumption that inadvertent intrusion will occur. This assumption is predicated on the simultaneous occurrence of at least five events:

- institutional control is lost;
- knowledge of the site's history is lost;
- the disposal site is unrecognizable as a disposal site;
- the waste is indistinguishable from soil; and

the population and well drilling densities are high enough to result in a random drilling into a burial cell.

Despite the conservative nature of these generic assumptions, most DOE LLW meet the IHI criteria. However, some long-lived wastes (e.g., thorium) fail the IHI standard, independent of the location and the depth of burial. For example, for a given disposal configuration, burial of LLW in metropolitan Washington, DC and burial at the NTS result in the same (calculated) dose to the IHI, using the well drilling scenario. The calculated dose using the well drilling scenario is also independent of the depth of burial. Burial under 3 m. of cover and burial under 30 m. of cover results in the same calculated dose to the IHI. Clearly, the probability of intrusion is greater in a large metropolitan area than at the NTS, and burial under 30 m. of soil, as opposed to 3 m., should reduce the probability of intrusion.

For DOE's LLWs that fail the existing IHI standard, independent of the depth of burial, an alternative approach is proposed which assumes that the probability of IHI varies with disposal facility. This alternative approach is within the flexibility allowed by DOE Order 5820.2A and takes credit for the NTS's arid environment and the fact that wastes will be recognizable for long time periods.

Alternative Approach

The alternative approach is based on several premises. The first is that protection of the IHI is an appropriate goal for the burial of LLW. The approach also accepts the IHI standard in DOE's 5820.2A as appropriate for most LLW. It presumes that LLW with characteristics of HLW (wastes that fail the IHI exposure scenarios, independent of the depth of burial) require an alternative approach, and finally, that the probability of IHI is different for different locations.

The alternative approach continues using the IHI standard found in DOE Order 5820.2A, as interpreted by Wood et. al. (15) and the DOE Peer Review Panel. For wastes that fail the IHI standard, independent of the depth of burial, the "expected dose" should be calculated. The expected dose is the product of the consequence of the intrusion times the probability of the intrusion. For example, if the consequence of intrusion is calculated to be 125 mrem/year chronic dose and the probability of intrusion is assessed at 0.65, the expected chronic dose to the IHI is 81 mrem/year. All aspects of calculating the consequence of human intrusion are identical to calculating the consequence under DOE 5820.2A; the only new element is the derivation of the probability of intrusion.

The expected dose could then be compared to (a) the IHI dose standard (100 mrem/year chronic and 500 mrem acute), or to (b) the MOP dose standard of 25 mrem/year. To compare the IHI expected dose to the 25 mrem/year MOP standard would be inconsistent with the existing DOE Order. However, there is concern that comparing the expected dose to the 100 mrem/year standard may not be appropriate, because the 100 mrem standard was set higher than the 25 mrem/year standard to compensate for the intrusion being "...not...routine" and involving only a few people. The higher IHI standard was set to partially compensate for the low probability of occurrence. With the formal inclusion of the probability of inadvertent intrusion, the higher dose standard may not be appropriate. This portion of the alternative approach is not currently resolved.

The proposed approach of assessing the expected dose to the IHI will allow the inclusion of important, site-specific characteristics in the decision making process. For burial at Area 5 of the NTS, these include:

Consideration of the very slow rate of waste decomposition (the long time required for the wastes to become unrecognizable as wastes) caused by the aridity of the site, which receives approximately 10 cm of precipitation annually;

Consideration of the depth to the water table, requiring the drilling of water wells over 230 m. deep;

Consideration of the past use of the NTS for testing nuclear weapons (knowledge of the past use of the NTS will be very difficult to lose because of the hundreds of large craters which will probably survive tens of millennia), and

Consideration of the very low population densities (and possibly low future population densities).

The question is how to derive a probability of intrusions which considers the site-specific characteristics of the disposal site. The NRC stated that it is very difficult to set a numerical value on the probability that an intrusion event will occur and to estimate the extensiveness of the intrusion. However, to develop 10 CFR 61, the NRC did estimate the extensiveness of intrusion by developing the standard exposure scenarios. This paper advocates using those same long-standing exposure scenarios, coupled with the site-specific assessment of the probability of occurrence, which is addressed in the following sections.

Probability

The use of this alternative approach requires assessing the site-specific probability of an individual inadvertently drilling a water well through buried LLW. Of importance is the definition of probability. There are two schools of thought concerning the definition of probability (6). The frequentists believe that probability is the result of repetitive experiments or observations. In this paper, we call this frequency. Subjectivists believe that probability is a state of knowledge or state of confidence. In this paper, we call this probability.

When there are insufficient data, the subjective probability is the only probability. Often, a numerical value can be assigned to this state of belief, e.g., "there's a 75% probability that inflation will return to double digits by the year 2000." Using this definition, probability is a "soft" number and may vary with the individual.

The assessment of probability is a key component of risk analysis, where risk is typically defined as RISK = PROBABILITY x CONSEQUENCE. Mathematically, RISK can also be thought of as the "expected value." With this definition of probability, the next question is how to derive a probability of inadvertently drilling at a specific location at the NTS over the next 10,000 years.

Determining the Probability of IHIs

There are a number of ways to assess probability (where probability is a state of knowledge or reasoned belief). A formal, defensible approach is the expert elicitation process. The process typically requires a trained facilitator, panelists (experts and/or stakeholders), some preliminary exchange of information, the formal elicitation process, and follow up activities. Bonano et al. (2) discusses many of the advantages and disadvantages of the formal elicitation process. The advantages include improved accuracy of judgment, well-thought-through design of elicitation, consistency of procedures, scrutability, communication, and

fewer delays. The cited disadvantages can include high cost, time, and flexibility.

There are seven standardized steps for a formal probabilistic elicitation. They are:

- Development of an issue statement (define the event of concern);
- Panel identification and selection;
- Presentation of the issue statement and process definition;
- Organization of the panel;
- Review of literature pertinent to the elicitation;
- Elicitation; and
- Documentation of elicitation.

The event would be divided into components and the panelists would assign probabilities to each component. For example, the issue statement might be "What is the probability of an individual inadvertently drilling a water well through a buried LLW cell at the NTS's Area 5?" The event could be divided into components such as:

- the probability that institutional control will be lost;
- the probability that knowledge of the past use of the NTS will be lost;
- the probability that wastes will become unrecognizable as wastes;
- the probability of drilling a water well in Area 5 of the NTS; and
- the probability of drilling though a buried waste cell.

Panelists would be asked to provide the probability of each component occurring and the reasoning for each numerical estimation.

The elicitation process is subjective. However, the assessment of probability is used daily for decisions of far greater consequence than assessing the probability of an IHI. For example, medical opinions typically employ probabilities, and U.S. economic policy (e.g., to raise or lower the prime interest rate) is based on indices and subjective probability.

SUMMARY

For the burial of LLW, we want to keep the wastes from getting to people and we want to keep people from inadvertently getting to the wastes (i.e., to protect the IHI). This paper addresses protection of the IHI. If the future could be anticipated, current society could define the specific measures to protect future intruders. However, the future is uncertain and society has established that demonstrating compliance using specific exposure scenarios is, by definition, protective of the IHI. For buried LLW wastes, the probability of inadvertent intrusion has been assumed to be one: intrusion will occur. This paper proposes incorporation of the site-specific probability of intrusion. For burial of LLW at the NTS, incorporation of this probability will allow the consideration of favorable site characteristics such as aridity, the slow decay of wastes, the great depth to water, and (possible) low future population densities. This proposed alternative is identical to the existing interpretation of DOE Order 5820.2A, except that the probability of intrusion is included as a site-specific characteristic.

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NEVADA RISK ASSESSMENT / MANAGEMENT PROGRAM (NRAMP)*

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ABSTRACT
The team of the Harry Reid Center for Environmental Studies (HRC) at the University of Nevada Las Vegas and the firm of E. J. Bentz & Associates (EJB&A) of Springfield, VA are conducting a risk assessment and risk management evaluation of environmental management activities in Nevada. The NRAMP is part of a national effort by the DOE Office of Science and Technology Policy (EM-52) to develop new sources of information and approaches with public involvement to risk assessment, risk management, risk communication and public outreach as these objectives relate to the ecological and human health effects of radioactive and hazardous waste management and site remediation activities. Implementation of this process has not yet been demonstrated at any site.
This program is currently eight months into a 21 month schedule. Conclusions to date are primarily based on the development of a public working group, literature surveys, collection of DOE national and site-specific data and a state-wide public opinion survey. Members of the public have been successfully organized into a "working group" to augment public involvement in coordination with the NTS Community Advisory Board. The scope of NRAMP activities differs from past DOE risk assessments in Nevada by integrating risks from all major contaminants, consideration of geologic time periods and by incorporating the project goal of "making a difference".

INTRODUCTION
The team of the Harry Reid Center for Environmental Studies (HRC) at the University of Nevada Las Vegas and the firm of E. J. Bentz & Associates (EJB&A) of Springfield, VA was awarded in April 1995 a cooperative agreement by the US Department of Energy Environmental Management Program Office of Science and Technology Policy (EM-52) to conduct a $2.8 million, 21 month risk assessment and risk management evaluation of environmental management activities in Nevada and to transfer this experience to other sites. The NRAMP is part of a national effort by the DOE to develop new sources of information and approaches with public involvement to risk assessment, risk management, risk communication and public outreach as these objectives relate to the ecological and human health effects of radioactive and hazardous waste management and site remediation activities.
Specific objectives of the NRAMP are to be accomplished by; 1) Working closely with stakeholders to develop credible independent assessments of risks at radioactive and hazardous waste sites in Nevada and other states; 2) Evaluating and prioritizing options in Nevada for waste site restoration and future land use from the perspective of stakeholders, and; 3) Transferring risk assessment/management results and methods from Nevada to stakeholder groups in other states as well as utilizing, from

other states, appropriate methods, tools and results for the evaluation of Nevada sites.

APPROACH

Specific tasks of the NRAMP implement a risk assessment process with public involvement as described in the National Research Council's report, Building Consensus Through Risk Assessment and Management of the Department of Energy's Environmental Remediation Program. Implementation of this process has not yet been demonstrated at any site. This process, if successful, holds great promise to reaching clearly defined, defendable and lasting agreements between the DOE, regulatory agencies and the general public on appropriate and sustainable strategies for site restoration, future site operations and waste management.

DOE operations in Nevada are an appropriate place for the development of risk analysis and risk management techniques because of the timing of this program and ongoing DOE needs for the NTS. NRAMP is timely to EM activities in Nevada because:

The NRAMP activities will generate greater public interest in ongoing NEPA activities in Nevada

The Community Advisory Board (EM's Site Specific Advisory Board in Nevada) is developing in Nevada and can benefit from access to the independent technical resources available through the NRAMP

There has been limited public involvement in risk-related work conducted in Nevada to date

The NTS is a multi-program site. The Nevada Test Site will continue to play an important role for the US DOE in the completion of defense, environmental and alternative energy missions

The Congress and the DOE have advocated the development and use of a risk-based programmatic approach to achieve human health goals for EM activities in a cost effective manner.

The NRAMP is demonstrating the application of risk assessment and risk management techniques with direct stakeholder involvement. The methods to be applied under this program are graphical to facilitate direct active participation of stakeholders in the application of the risk assessment model, assumptions and data. Value-impact assessment techniques developed by the US Nuclear Regulatory Commission will be used in the NRAMP to systematically consider risk, cost and judgmental factors in the stakeholder identification and prioritization of options for site restoration.

Five principle means will be used to accomplish the objectives of developing credible, broadly-accepted risk assessments and improving decision-making processes for Nevada and other states such that they reflect the priorities and concerns of all stakeholders:

First, using the principles of consensus building, develop a Working Group consisting of stakeholders to work towards a consensual agreement.

Second, supplement the meetings of the Working Group with systematic and extensive opportunities for stakeholder participation in developing a detailed scope of work and conducting the risk assessment/management evaluations.

Third, development of quality products through the adaptation of risk management, value/impact processes and technical computer tools.

Fourth, include extensive independent technical reviews.

Fifth, collaborate with other national sites to transfer results and include applicable results from other sites in the NRAMP.

RESULTS TO DATE

This program is currently ten months into an 21 month schedule. Task 1, Development of a working group has been completed. The NRAMP working group consists of 70 individuals who began meeting in May 1995 to learn about the Nevada Test Site and identify issues for the NRAMP team. The NRAMP team also established relationships with the NTS Community Advisory Board and the Nevada Operations Office. Results of these discussions and relationships included the development of a risk assessment scoping matrix which is the basis for the development of the technical portion of the NRAMP, a state-wide public opinion poll and a summary of risk related activities by site-specific advisory boards at all DOE sites. A work plan for the development of a preliminary risk assessment of DOE sites in Nevada is under development.

NRAMP Working Group Development

Primary methodologies used to develop the NRAMP working group included working closely with the Nevada Test Site Community Advisory Board (CAB) - whose composition carefully mirrors public demographics, local governments, and involved agencies - and forming a Working Group with much broader public involvement to ensure inclusion of all stakeholders. The NRAMP Working Group has several responsibilities: providing a list of their priorities for future NTS land uses, reviewing this Work Plan to ensure stakeholder issues have been appropriately included, providing input and reaction to Preliminary Risk Assessment data, and accepting or rejecting the conclusions of the Baseline Risk Assessment and prioritization of site restoration activities. By analyzing insights gained from interactions with this Working Group, the NRAMP team will construct a stakeholder model for use at similar DOE sites across the nation.

Stakeholder identification was Milestone I for the NRAMP and several methods were applied to reach it. The project was first presented to the NTS CAB. Although the CAB was initially concerned about duplication of effort, it was strongly swayed toward cooperation by the concept that NRAMP provided an independent research team to accomplish studies the CAB did not have resources to complete. As a result, two members were appointed to attend the Working Group meetings and to coordinate CAB and NRAMP activities. This kind of cooperation with existing groups is an important component of the developing interaction model.

Using the CAB mailing list and a similar list from DOE/NV Environmental Management, letters of personal invitation were mailed to persons who had shown previous interest in the NTS. General public announcements were aired on local radio and television stations, and large advertisements were run in local newspapers. Everyone who responded was included in the NRAMP Working Group, and no attempt was made at representative demographic sampling.

However, at the same time, a statewide telephone survey* was conducted to validate Working Group input by comparing it with general Nevada interests and attitudes toward the NTS. The survey questions also provided insight into the general level of public knowledge about the NTS and its possible future uses. By design, the survey deliberately over sampled Nevada's rural areas. Four hundred miles separate the state's two major population centers of Las Vegas and Reno/Carson City. Nye County, which includes most of the NTS, lies between these two centers and is almost totally rural in nature. The NRAMP Survey collected 400 opinions from the Las Vegas area, 400 from Reno/Carson City, and 400 from rural Nevada.

Survey results show that the public perceives very high risk from transportation of high-level radioactive waste (71.9%), storage of radioactive waste (67.7%), transportation of hazardous waste (60.8%), worker exposure to radioactivity (55.3%), groundwater contamination (54.8%), ecosystem damage as a result of soil contamination (53.3%), atmospheric releases of radioactive materials (53.0%), and, in general, to their personal safety and well-being (52.1%). The same public supports the following future uses in descending order of popularity: alternative energy research and development (79.6%), environmental restoration research (78.2%), nuclear emergency response and safety technology development (65.6%), weapons disassembly (52.9%), hazardous waste management (50.5%), nuclear materials management and technology development (49.5%), open air nuclear age museum (48.5%), maintenance of current uses (33.4%), return to public land inventory (29.1%), and nuclear fuel storage (21.3%).

Initial surveys among the Working Group show generally similar results; however, survey issues are currently being studied by the group and interesting comparisons will be drawn at the conclusion of the NRAMP education effort (see below).

Working Group Interactions

Because of the state's geography, two separate groups have been formed: the Las Vegas or Southern Working Group and the Reno or Northern Working Group. (A third group, the Nye County Working Group, is still being organized.) The Southern Working Group met for the first time in May 1995 and collected a good representation of 45 local citizens, agency representatives, and pre-existing groups who appeared willing to work together in learning and using the risk-based process. Seven persons attended the first meeting of the Northern Working Group one week later in June. Attendees at both meetings were interested in knowing why they should devote time and effort to still another public meeting, questioned how their input would be used, and wondered whether they could truly have a significant impact on DOE decisions.

To answer these questions, the NRAMP team chose a meeting model based on three focus areas: 1) response to recorded stakeholder issues from previous meetings, 2) educational presentations, and 3) group input activities. As an example, the second meeting was planned to begin with an education segment, a presentation on plutonium contaminated soils. A response to stakeholders was prepared to address group issues such as reasons for supporting the NRAMP and its opportunities for input. And, to close, an input activity, a group reading and revision of the public opinion survey, was conducted.

Stakeholder issues identified in the June meetings included plutonium contamination inside and outside the NTS, the extent of environmental damage, mutations in animals, the State's cancer rates, contaminated ground water, the Ruby Valley Treaty, and the DOE decision not to attempt clean-up of underground contamination. In addition, both groups requested more information about the NRAMP, including its time line and its procedures.

In an attempt to bring the two Working Groups together, the NRAMP team moved to videoconferencing for the third, or August, meeting. The technology proved very effective and provided a venue for presenting the NRAMP Risk Assessment Matrix, the near-term opportunities for Working Group input to DOE, and a review of all issues raised by the group, as well as the NRAMP responses to those issues.

The Risk Assessment Matrix identifies five general categories of hazards (underground contamination, surface contamination, industrial sites, waste disposal, and transportation-related issues), maps them across their possible receptors based on land use, determines the time period under consideration, identifies subjects at risk, and explores the trade-offs in costs and opportunities. The NRAMP will provide stakeholders with an in-depth look at the identified categories and, in the Baseline Risk Assessment, ask the Working Group to prioritize future land uses based on what has been learned.

At the same time, the Working Group will provide input to DOE documents currently in preparation and on DOE actions in progress such as the Federal Facility Agreement and Consent Order, the NTS Resource Management Plan, the EM Risk Report, the Waste Management PEIS, the NTS EIS, and the Federal Facility Compliance Act.

The NRAMP is carefully tracking and responding to all issues raised in Working Group meetings. Major issues include

Short- and long-term risks to the public and the environment from above-ground test residuals (Response: include in risk assessment)

Describe the various forms of potential waste for storage (Response: schedule for Working Group presentation)

Land ownership is an open dispute with Native Americans (Response: acknowledge in NRAMP reports and document input)

The NTS is a potential museum to the atomic age (Response: include in list of future land uses)

Cultural resources should be considered in the risk assessment (Response: develop maps of NTS locations and review current status during risk assessment)

NRAMP should provide input to the NTS EIS, Resource Management Plan and Compliance Agreement (Response: include in risk assessment, write letters requesting formal participation, invite presentations by DOE officials, and develop group comments in writing)

A finding of "no risk" based on current low population density is unacceptable (Response: include future potential populations and workers in the risk assessment)

Consider cumulative risks from storage, waste management, and past testing (Response: include in Matrix to extent possible)

The working group will decide about decision making and facilitation processes(Response: allow groups to develop most effective process for individual circumstances)

Consider the effect of teleconferencing on group decision making (Response: encourage group to experiment with new technologies)

Include more rural and Native American representatives (Response: hold Nye County meeting, create new press releases, present Native American contributions included in NTS EIS)

Working Group should control of NRAMP agenda (Response: agree)

Several general problems remain to be overcome. Some group members are still distrustful of NRAMP motivations and purposes because of project links to DOE. Many are still struggling to grasp the whole NRAMP concept and exactly how it can make a difference. Frictions and differences in opinion and interest exist between working group members. However, several NRAMP successes with group interactions can be plainly seen even at this point. A dedicated group of stakeholders has been formed without resort to the lengthy and complicated "keystone" process. Interesting uses of technology have been applied: the telephone poll has helped to

compensate for the vast distances of a sparsely populated state, and videoconferencing has begun to coordinate the efforts of two geographically separated groups. In addition, the Working Group has committed itself to an action-oriented approach which appears more aggressive than that of site advisory boards in general. The NRAMP Working Group is clearly geared toward making a difference.

Working Group Education

The NRAMP team believes that stakeholder input must go hand-in-hand with education, and team efforts will include training in group processes, instruction in basic scientific data collection and analysis, and information about current risk management theory and practice.

In an effort to learn from experience with other stakeholders, the NRAMP Team Member, E. J. Bentz and Associates, Inc., conducted a study of existing DOE advisory boards. Their report, Risk-Related Activities Currently Being Performed by DOE Site Specific Citizens' Advisory Boards, provides narrative profiles of advisory boards at Fernald, Hanford, Idaho National Engineering Laboratory, Monticello, Pantex, Rocky Flats, and Savannah River. Specifically, it addresses each site's tools for consensus building and communication, future land use options, management and remediation options, risk approaches/methodologies, and perceived cost benefits. Data from the report are being used to suggest new policies and approaches for NRAMP participants. The Working Group's most pressing need in the process area is decision-making, and the facilitator will develop several alternatives for selection by the group.

The crucial point to be made above and beyond all of these activities is that forging "groups" into "teams" appears to be the pivotal component for success with stakeholders. The Working Group need to develop enough confidence in themselves and in the NRAMP team to work through difficult, but necessary, decisions. They also need enough information about risk management to judge the efforts of the NRAMP team and to understand the basics of the scientific data which underlie the risk management process. One of many insights gleaned from the NRAMP process is that formation of such stakeholder teams should begin long before attempting consensus on such complex choices as future land uses of the NTS.

Working Group Participation

Successful stakeholder participation is occurring through the Working Group in several identifiable areas. Comments in behalf of the group have been submitted to the EM-6 Risk Report to Congress and the group has requested information about or inclusion in NTS activities such as the NTS EIS, the Resource Management Plan, the Federal Facility Agreement and Consent Order, the Yucca Mountain Project, and the Nevada Department of Environmental Protection. Areas of study have been identified, and a matrix for understanding the risk process has been prepared.

Although the Working Group have not yet fully formulated their position on future land uses, they will review the general modeling procedures described in this Work Plan. The NRAMP team intends to continue consensus building while conducting the Preliminary Risk Assessment. The Working Group will study the issues being modeled, and it is hoped that stakeholders will be ready to examine and state their priorities in relation to the models of both the Preliminary and the Baseline Risk Assessments.

Specifically, then, stakeholders will be expected to play the following important roles through participation in the Working Group:

1. input to major working documents

2. identification of future NTS land uses 3. acceptance or rejection of the preliminary integrated risk analysis
4. addition of specific areas of concern for concentrated modeling
5. evaluation of cost effectiveness of alternatives based on the Baseline Risk Assessment
6. endorsement of specific future land uses with full knowledge of risk and cost analysis.

Preliminary Risk Assessment Work Plan

The Preliminary Risk Assessment (PRA) that is currently under development is intended to be a screening assessment of present and future risks to the public, workers and the environment from existing, and anticipated environmental management program activities in Nevada. This plan is currently under review by the NRAMP Working Group and the NRAMP peer review team. The PRA has the primary objective to "identify the holes" in understanding the NTS relative to the scope of a risk assessment discussed with the working group, to facilitate further discussions with the working group on acceptable risk levels and to obtain comment from the NRAMP peer review group on the general approach to the assessment of NTS risks. The PRA will form the basis for evaluating the value of planned site remediation activities as part of the baseline risk assessment, development of additional risk management concepts during the risk management task, and the identification of data and model needs to better understand and manage risks from DOE activities in Nevada.

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MEETING NEW AIR STANDARDS WITH A VOLATILE ORGANIC TREATMENT TRAIN

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ABSTRACT

The U.S. Environmental Protection Agency (EPA) issued the second phase of the organic air emission standards for hazardous waste treatment, storage, and disposal facilities (TSDFs) and hazardous waste generators in December 1994. These standards (referred to as the Subpart CC standards) are designed to further reduce organic air emissions from hazardous waste management activities. Lawrence Livermore National Laboratory (LLNL) is evaluating a modification of its existing Waste Water Treatment Tank Farm (hereafter called Tank Farm) with a volatile organic removal and destruction treatment train in order to comply with these new air standards.

LLNL's Tank Farm consists of six, 7,000-L open-top tanks used to store and treat aqueous low-level radioactive, mixed, and hazardous waste. The

aqueous waste has elevated volatile organic constituent (VOC) concentrations while stored and treated in the tanks. According to the Subpart CC standards, tanks handling waste with similar VOC concentrations must be retrofitted with a cover and an emission control device for cover openings; the installed emission control devices must achieve at least a 95% reduction in the total organic content of the vented gas stream. LLNL concluded that the removal and destruction of VOCs from waste before they enter the Tank Farm would demonstrate compliance with the Subpart CC standards more effectively and be more cost effective than the installation of air emission control devices on the Tank Farm. LLNL has designed this removal and destruction technique to consist of an air stripper, high-efficiency particulate air (HEPA) filter, catalytic oxidizer, scrubber, and mist eliminator.

INTRODUCTION

LLNL operates a waste water treatment tank farm for the treatment of aqueous low-level radioactive, mixed and hazardous waste. Federal air emission standards promulgated under RCRA will restrict LLNL from treating waste exceeding 100 mg/kg VOC concentration in the open-top design of the tank farm. LLNL has evaluated several compliance options and selected a preferred option in order to continue treating its aqueous waste streams.

LEGAL REQUIREMENTS

The Resource Conservation and Recovery Act (RCRA) directs the EPA to promulgate regulations for monitoring and controlling air emissions from TSDFs as may be necessary to protect human health and the environment. Two phases of the congressional directive are complete. The first phase includes the promulgation of standards that control organic emissions vented from certain hazardous waste treatment processes (i.e., distillation, fractionation, thin-film evaporation, solvent extraction, steam stripping, and air stripping) as well as from leaks in certain ancillary equipment used for hazardous waste management processes. The second phase includes the promulgation of emission standards designed to further reduce organic emissions from hazardous waste management activities associated with tanks, surface impoundments, containers, and miscellaneous units operated at TSDFs. This second phase is referred to by EPA as the Subpart CC standards.

In reference to tanks managing hazardous waste, the Subpart CC standards apply when the waste exceeds a VOC concentration of 100 mg/kg; the VOC concentration of a waste is determined at the point of waste origination. The blending of VOC-ladened wastes with other aqueous wastes to lower the VOC concentration is not permitted as a means to avoid subjection to the standards. If a tank manages waste exceeding the regulatory VOC concentration, the regulations require the installation and operation of one of the following air emission control systems on the tank:

- A cover that is connected through a closed-vent system to a control device
- A fixed-roof type cover with an internal floating roof
- An external floating roof
- A pressure system that allows the tanks to operate as a closed system.

In specific cases, the Subpart CC standards allow the use of a fixed-roof-type cover without additional controls. The specific cases include tanks which do none of the following:

- Mix, stir, agitate, or circulate waste using a process that results in splashing, frothing, or visible turbulent flow on the waste surface

Heat the waste except to prevent the waste from freezing or to maintain adequate waste flow conditions

Treat waste using a waste stabilization process or a process that produces an the exothermic reaction

Exceed a defined maximum organic vapor pressure.

The type of air emission control device incorporated into a tank design modification is not specified by the performance-based standards of Subpart CC. TSDFs have the flexibility of choosing an air emission control device best suited for the characteristics of the organic emissions. However, the installed air emission control device must achieve at least a 95% reduction in the total organic content of the vapor stream vented to the device or, in the case of an enclosed combustion device, a reduction of the total organic content of the vapor stream to a level less than or equal to 20 mg/kg on a dry basis corrected to 3% oxygen.

The Subpart CC standards provide an alternative to the installation and operation of air emission control systems. TSDFs are allowed to remove or destroy organics in the waste by a treatment process that significantly reduces the volatile organic concentration so that storage and treatment units operated downstream of the treatment process in the waste management sequence do not have to use air emission controls. This alternative does not require the owner or operator to perform volatile organic concentration waste determinations for the hazardous wastes prior to treatment, yet still accommodates the blending of wastes that have different volatile organic concentrations.

The Subpart CC standards are temporarily deferred for systems solely managing mixed wastes (waste that contains both hazardous waste and source, special nuclear, or byproduct material subject to the Atomic Energy Act of 1954). EPA provided the deferral because the air emission control equipment required by the Subpart CC standards may conflict with certain radioactive waste management requirements under the Nuclear Regulatory Commission. The Subpart CC standards will be effective in December of 1995.

TANK FARM

LLNL's onsite activities generate a variety of aqueous hazardous and mixed waste. Many of the waste streams contain VOCs such as tetrachloroethylene (PCE), methyl chloroform, and methylene chloride. The organic concentration of each waste stream ranges between 0 mg/kg and 10,000 mg/kg. The VOC-containing waste streams are generated at a rate of approximately 586,000 L/y.

The aqueous waste streams are initially accumulated at the point of generation in various size containers ranging from 19 L carboys to 3785 L portable tanks. The containers are transported from the onsite activities to the treatment facility and placed in storage as destined for treatment. Treatment of the waste begins with the transfer of waste from a select number of containers in a predetermined sequence to one of the tanks of the Tank Farm. The sequence of blending is established to moderate reactions and dissipate heat of reactions. The blended waste within a tank is subsequently treated using one or more of several different treatment techniques. The treatment techniques are selected to achieve a desired treatment objective. The types of treatments potentially performed on a batch of blended waste within a tank include:

Neutralization/pH adjustment

Oxidation/reduction

Cyanide destruction
Precipitation
Chelation/flocculation
Ion exchange
Adsorption
Separation.

Treated waste may be shipped offsite for disposal or discharged to the City of Livermore Water Reclamation Plant via the sanitary sewer in accordance with established discharge limits.

The Tank Farm consists of six, 7,000-L tanks with ancillary equipment such as piping and pumps. Piping system of the six tanks allow for the transfer of wastes to and from each tank, to and from a filtration system, and to and from portable tanks and containers. Each tank is a vertical, open-top, cylindrical tank. A roof, constructed of metal sheeting on steel supports, partially protects the tanks from precipitation and weathering. The tanks are equipped with high-level alarms, high-level interlocks, pH probe, treatment reagent lines, and mixers. A diagram of a 7,000-L tank is presented in Fig. 1.

Prior to the blending process, wastes to be treated in the Tank Farm have average volatile organic concentrations equal to or greater than 100 mg/kg. The treatment methods employed in LLNL's Tank Farm create exothermic reactions and prevent LLNL from qualifying for the special case variance. Consequently, the Subpart CC standards will require either the open-top design of the tanks in the Tank Farm to be modified or a change in LLNL's hazardous waste management operations.

The simplest and least costly modification of the Tank Farm design to comply with the Subpart CC standards is the installation of a fixed-roof cover connected through a closed-vent system to an air emission control device. The organic emissions from the Tank Farm result from the evaporation of VOCs at the liquid surface of the waste and their dispersal into the atmosphere by displacement during tank filling, diffusion, or wind. The rate of the organic emissions depends on the physical and chemical characteristics of the waste, temperature of the waste, tank design, tank condition, and operational characteristics. Although mixing and exothermic reactions within the tanks increase mass transfer rates, the blending of the waste reduces both the VOC concentrations and the overall organic emission rate while the waste remains in the tank. The low organic emission rates create difficulty in the identification of an air emission control device for LLNL's Tank Farm that achieves the Subpart CC performance-based standards. The Subpart CC standards suggest two air emission control systems to achieve the performance standards: adsorption systems and thermal destruction. Adsorption systems work best on air emissions that are below the economic threshold for thermal oxidation. Adsorption systems include adsorbents that may be either polar or nonpolar. Polar sorbents have a high affinity for water vapor and are ineffective in air streams that have any appreciable humidity, but most air streams associated with waste treatment are humid. Nonpolar adsorption, like activated carbon, is effective at removing all but the most volatile compounds associated with waste treatment systems. Adsorption capacity of an adsorbent is directly proportional to the concentration of the adsorbate. Thus, all else being equal, more adsorbent is required to remove a lower concentration contaminant with equal efficiency than to remove the same contaminant at higher concentrations. EPA suggests that the minimum air stream VOC

concentration for carbon adsorption to economically achieve a 95% removal efficiency of VOCs is 1,000 mg/kg by volume.
Fig. 1
If LLNL were to use carbon adsorption technology on a Tank Farm design modification to achieve the performance-based standards, the fixed-roof cover would require an actively vented system connection to the carbon adsorption air emission control device. A passively vented system on the Tank Farm may maintain low organic emissions in the air stream, but the amount of carbon necessary to meet the performance-based standard would be impractical. However even with an actively vented system, it is difficult to achieve the 95% removal efficiency using a carbon adsorption system. Adsorbers that operate periodically or where the concentration of the contaminate varies greatly, such as in batch operations with varying waste streams, can be quickly impaired. In addition, an actively vented system on a tank containing aqueous waste will create a high humidity gas stream. Excessive humidity in the gas stream can reduce the effectiveness of adsorption systems. Gas streams close to saturation can cause capillary condensation, which occupies potential adsorption sites and blocks access to the carbon pores.
Thermal destruction of vented organics can achieve the Subpart CC performance standards using flares or other thermal oxidation units. However, LLNL has had extreme difficulty in obtaining a permit with the State of California for direct-fire or flame techniques. LLNL speculates that this difficulty is a result of poor public opinion on these techniques. Use of other thermal oxidation units as air emission control devices are economically impractical for the low organic emission rates associated with LLNL's Tank Farm operations.
The difficulties in using available technologies for air control emission devices on the tank system forced LLNL to evaluate the alternative provided by the Subpart CC. In particular, LLNL investigated the removal of VOCs from the aqueous wastes before the wastes are introduced into the Tank Farm. LLNL's preferred option to remove VOCs from the waste is air stripping, but the volatile organic removal or destruction method following the air stripping poses similar problems for an air emission control device installed with a fixed-roof cover.
Using effective air stripping of LLNL's aqueous wastes, the worst case VOC concentrations (typically methyl chloroform, PCE, and methylene chloride) in the gas stream exiting the air stripper will be 1-10% organics. Condensation following air stripping has potential rewards, and LLNL will evaluate them further; however, the condensed organics will require treatment prior to disposal. Because many of the wastes have radionuclides, solvent recovery was not considered. Carbon adsorption systems are prohibitively expensive for removal of VOCs in the gas stream from the air stripper because of the high organic concentrations. LLNL did not evaluate direct-fired or flame destruction techniques because of the difficulty in obtaining a permit with the State of California these techniques.
The methods LLNL investigated to destroy the VOCs in the gas stream exiting the air stripper included ozonation and catalytic oxidation. Ozonation uses ozone to oxidize organic contaminants in two ways:

- By direct oxidation with ozone gas
- By the generation of free radical intermediates, such as hydroxyl radicals.

Contaminants most amenable to direct oxidation by ozone include aromatics such as polycyclic aromatic hydrocarbons and chlorinated ethenes. The major concerns with using ozone are that it is not good with low-molecular-weight chlorinated organics and the concentrations of the gas stream expected by LLNL are extremely high for cost-effective systems. Catalytic oxidation is well suited for the destruction of VOCs in vent gas streams, especially when the hydrocarbon concentrations are 25% or less of their lower explosive limit with flow rates ranging from 14 standard m3/min to over 2,800 m3/min. Catalytic oxidation systems are normally designed for destruction efficiencies that range from 90-98%. The destruction efficiency for chlorinated hydrocarbons is typically quite low; however, recent catalyst developments have produced products that can effectively destroy chlorinated hydrocarbons. For example, in a joint venture of King, Buck & Associates Inc. (KBA) in San Diego, California and Catalytic Combustion Corp. in Bloomer, Wisconsin, a catalytic oxidizer system demonstrated the capability of treating 5.7 m3/min of chlorinated VOCs at a concentration of more than 2,000 mg/kg.

Proposed Modifications

Based on the potential of catalytic oxidation, LLNL's selected preferred option for compliance with the Subpart CC standards includes an air stripper followed by nonflame catalytic oxidation using a halocarbon destruction catalyst as a removal and destruction system. The specific design of the system consists of an air stripper followed by a heat exchanger, preheater, catalytic oxidizer, and scrubber. A diagram of the unit is provided in Fig. 2.

The catalytic oxidizer system will be operated on a batch basis as needed. Typical waste management activities will require the system to operate six hours per week. As waste streams are transported from the onsite activities in various containers, the containers will be incorporated into a blending sequence. The blending sequence will be developed to optimize the volatile content of a blended waste stream for air stripping. In addition, the blending sequence will be designed to limit reactions and dissipate reaction heat.

According to the determined blending sequence, up to four containers will be emptied using a vacuum pump. The pumped waste will be blended via a pipe manifold system. The blended waste stream will then be piped to an air stripper at controlled flow rates up to 57 L/min based on the volatile content of the blended waste stream. As containers are emptied, they will be replaced manually with containers containing waste according to the blending sequence schedule.

The air stripper will be designed to remove at least 95% of the VOC content from the blended waste streams. An air stream of approximately 14 m3/min will strip the VOCs from the blended waste streams. The stripped aqueous waste stream will be pumped to a 4,200-L portable tank with a condenser installed on the vent line or to one of the open-top tanks of the Tank Farm. The actual location to which the aqueous waste stream will be pumped will depend on the precautions necessary to ascertain that the system has adequately stripped the blended aqueous waste stream. The stripped aqueous waste stream will have a concentration less than 50 mg/kg in order to comply with the Subpart CC standards.

Fig. 2

The VOC concentration of the gas stream exiting the air stripper will range between 300-50,000 mg/kg. The gas stream will be directed to a duct heater to elevate the temperature of the air stream from ambient

temperature to 38C. The gas stream will then enter a blower followed by a HEPA filter to remove radionuclides associated with the mixed wastes. After filtration, the gas stream will enter a hot catalytic oxidizer to oxidize the organics. The catalytic oxidizer will contain approximately 0.28 m3 (10 ft3) of a halogenated hydrocarbon destruction catalyst capable of achieving greater than 99% destruction efficiency. The air stream exiting the oxidizer, typically at a temperature between 370-540C, will then be quenched and neutralized using a sodium carbonate solution in a quench column. The quench column will scrub the acid gas generated during oxidation. The neutralizing solution can be recycled or treated in LLNL's Tank Farm. The gas exiting the quench column will be demisted and exhausted to the atmosphere.

The capital cost of the system is approximately $350,000. Electrical power requirements are largely required for the pumps, duct heater, and hot catalytic oxidizer. The power costs will be approximately $15 per hour of operation. The catalyst will have to be replaced, but since the system is only operated periodically, catalyst replacement should only be necessary once every six years at an approximate cost of $85,000.

STATUS AND PLANS

The VOC removal and destruction system will require permitting under RCRA and the Clean Air Act. Regulators from both the Bay Area Air Quality Management District and California Department of Toxic Substances Control support catalytic oxidizer systems which are not direct fired. Based on preliminary discussions, the regulatory agencies support LLNL preferred option for compliance with the Subpart CC standards.

LLNL is currently completing a specification package on the catalytic oxidizer system design. The system will not be on-line before December, 1995; therefore, an implementation schedule for installation of the equipment will be developed and placed in the facility operating records in order to comply with the Subpart CC standards.

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ACKNOWLEDGEMENT

Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract W-7405-ENG-48.

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ABSTRACT
The Clean Air Act (CAA) and its associated regulatory requirements represent one of the most complex environmental programs in the U.S. The program involves consideration not only of specific pollutants and processes associated with the generation of those pollutants, but also the application of mathematical modeling, periodic and possibly continuous emissions monitoring, analytical meteorology, source-receptor relationships, human health and ecological risk assessments, control techniques and management plans, reporting, and recordkeeping. These considerations can be compounded by the complexity of facility processes and the geographic siting of a facility. Meeting the applicable CAA requirements can be a complex task even for the most experienced DOE facility manager.
Using Technical Area 53 (TA-53), the Los Alamos Neutron Scattering Center (LANSCE), within the U.S. Department of Energy's (DOE) Los Alamos National Laboratory (LANL) as an example, this paper summarizes steps that were successfully taken to bring a complex facility into compliance with the radioactive air emissions requirements of the Clean Air Act (CAA).
Additionally, several other issues related to the Radionuclide NESHAP Memorandum of Understanding (MOU) between the USEPA and DOE for complex-wide application, as well as the USEPA's self-policing policy, will be covered briefly.
OBJECTIVES AND APPROACH
The primary objective of this paper is to examine how the LANSCE facility located at LANL successfully implemented new management techniques to demonstrate compliance with radioactive air emissions requirements after receiving U.S. Environmental Protection Agency (USEPA) citations for nonattainment. Secondarily, it suggests several additional air quality management program factors that might be considered by DOE and other facilities in seeking to address issues and opportunities associated with the management of facility air emissions. These represent factors that were not in place when the Notice of Noncompliance for radioactive air emissions was received.
INTRODUCTION
At the outset, it is important to recognize that three major portions of the Clean Air Act (CAA) Amendments of 1990 (P.L. 95-95) are most applicable to facilities managed by DOE: Titles I, V, and VI. Title I contains requirements associated with Attainment and Nonattainment Areas (107); National Ambient Air Quality Standards, or NAAQS (109); State Implementation Plans, or SIPs (110; Hazardous Air Pollutants, or HAPs (112); and the Prevention of Significant Deterioration, or PSD (160-169). Title V contains requirements to fulfill the Operating Permit Program (501-507), and Title VI contains requirements associated with Stratospheric Ozone Protection (601-618).Some of the specific CAA regulatory drivers that relate to DOE sites include the following recordkeeping and monitoring requirements:
  Requirements applicable to stationary sources that emit air pollutants for which a NAAQS is in effect to maintain records concerning the nature

and amount of these emission(s), along with air sampling data and other information deemed necessary to determine compliance with applicable emission limitations listed on the site's permit. These records must be kept a minimum of two years (40 CFR 51).

Requirements for stationary sources that emit HAPs involve the maintenance of monitoring data records, monitoring system calibration checks, and the occurrence and duration of any period during which the monitoring system is malfunctioning or inoperative; records of emissions test results from stack sampling that are used to determine total emissions; and records of concentrations at all sampling sites and other data needed to determine such concentrations. All these records must be held a minimum of two years (40 CFR 61).

Historically, most States and the Air Quality Management Districts (AQMDs) within them have issued air permits for individual non-radioactive emissions units under the applicable State Implementation Plan (SIP). Accurate and documented monitoring is the key to accountability and compliance under this regulatory approach. Continuous Emissions Monitoring (CEM) systems are referenced in 504 under Title V of the CAA. The entire CEM system must be installed and certified before it can be used for compliance, and a written quality assurance/quality control (QA/QC) plan must accompany the permit application. The CEM system utilized by a particular site would be dependent on the compounds of interest and the monitoring requirements based on the CAA.

DISCUSSION

Los Alamos National Laboratory (LANL, or "Lab") is a nuclear research lab located in northern New Mexico, immediately adjacent to Bandelier National Monument (Fig. 1) operated by the Board of Regents of the University of California. The Laboratory must meet the requirements of over 20 major environmental laws. Under the CAA, LANL is concerned with the National Emission Standards for Hazardous Air Pollutants (NESHAP) for radionuclides, asbestos, and beryllium; the NAAQS; and the New Mexico Air Quality Control Regulations. In addition, because Bandelier National Monument is adjacent to LANL, the air quality requirements associated with PSD also must be considered. The Air Quality and Meteorology Section of LANL's Environmental Protection Group (ESH-17) has responsibility for air quality services to LANL, including monitoring, surveillance, meteorology service, emergency response, and environmental assessment for ambient and facility need.

Fig. 1

New Mexico State ambient air quality standards are more stringent than the National standard for all items, and the Laboratory has been in compliance with all nonradiological ambient air quality standards for the past several years. LANL has a site-wide Air Quality permit application for non-radioactive emissions submitted to the New Mexico Environmental Department (NMED) under the Title V program. There is no permit, as such, for radioactive emissions; however, these are covered specifically under 40 CFR 61 Subpart H, which sets 10 mrem/year as the dose limit for DOE facilities.

TA-53, the Meson Physics Facility (Fig. 2) is used for physics research, and is the site for a linear particle accelerator for research in areas of basic physics, materials studies, and isotope production, the Ground Test Accelerator, the Proton Storage Ring, and the Los Alamos Neutron Scattering Center (LANSCE), formerly known as the Los Alamos Meson Physics Facility (LAMPF). By accelerating protons to an energy of 800 Mev

into graphite and tungsten targets, this facility produces a variety of sub-atomic particles. Over 25 radioisotopes are currently produced at LANSCE, primarily for a variety of medical and research procedures, and are distributed world-wide. A variety of radioactive materials are therefore generated at LANSCE by radioactivation, among them radioactive air emissions. The air that passes through the proton beams, as well as material used in the beam stops and experimental area, becomes activated while the beams are on. The radionuclides thus produced include Ar-41, Be-7, Br-82, C-10, C-11, N-13, N-16, O-14, O-15, Ta-182, and H-3.
Fig. 2
LANSCE operates approximately 3000 hrs/yr, or about 29% of the total number of available annual hours. Although LANSCE personnel would like to operate 7000 hrs/yr (80%), the overriding concern is that LANSCE contributes over 98% of the total allowed radionuclide emissions for the entire Lab.
As indicated earlier, accurate and effective monitoring is the primary method of assessing accountability and compliance, and CEMs are EPA's stated method of choice. At LANL, several systems are in place, designed to monitor both point and non-point sources as well as both radioactive and non-radioactive emissions. Radioactive air emissions at the Laboratory are evaluated primarily against 40 CFR Part 61, Subpart H, National Emission Standards for Emissions of Radionuclides Other Than Radon From Department of Energy Facilities, as well as DOE/EH-0173T, Environmental Regulatory Guide for Radiological Effluent Monitoring and Environmental Surveillance.
The Laboratory operates monitors to routinely measure primary pollutants, beryllium, acid precipitation, and visibility.
Radioactive Air Emissions. Based on self-reporting, in 1991 DOE notified EPA that it could not demonstrate full compliance with certain requirements under 40 CFR 61. As a result, the USEPA Region VI issued two Notices of Noncompliance (NONs). The first NON documented deficiencies in LANL's identification and evaluation of release sources, lack of stack monitoring equipment on all point release sources, inadequate quality assurance programs, and incomplete reporting. The second NON was issued for using a method that was not approved by EPA for calculating the emissions dose that resulted in exceeding the 10 mrem standard for calendar year 1990. The issuance of the NON immediately presented concerns to DOE and the University of California. DOE and EPA commenced negotiations for a Federal Facility Compliance Agreement to establish the framework for demonstration of site-wide compliance with NESHAPs requirements
Working with the EPA Region VI office in Dallas, Texas, DOE and the University of California-operated Lab made a concerted effort to correct these air program deficiencies. Procedures to more closely monitor the stack emissions were implemented. Four additional ambient air monitors co-located with LANL monitors in Los Alamos and nearby White Rock are used to verify LANL measurements, and to help quantify nonstack-type emissions.
As indicated, LANSCE contributes in excess of 98% of the Lab's total radioactive emission allowance. These products are very short-lived, as indicated in the isopleth (Fig. 3) which also illustrates the locations of LANSCE, monitoring sites, and the potentially Maximally Exposed Individuals (MEIs).
Fig. 3

Following receipt of the NON, measures were implemented at LANSCE to bring radioactive air emissions into compliance. In May 1995, the "LANL Radioactive Air Emissions Management Plan for LAMPF, TA-53" modified a 1993 plan ("LAMPF Compliance with Laboratory Radioactive Air Emissions Limits") in several ways. Two important changes of the new Air Emissions Management Plan: 1) although annual emissions reports to USEPA Region VI, as required by regulation, will continue, these reports will be filed internally monthly, making it easier to maintain running 12-month mrem totals; and 2) graded frequencies of data analyses and reporting of environmental exposure were replaced by more real-time management decision making, increasing assurances and administrative controls to achieve NESHAP compliance and providing more useful, defensible data. Other specified actions were also outlined in the new Air Emissions Management Plan. One of these, an Air Quality Group procedure, the "Monthly Curie Limit Projection for LAMPF", was developed to construct the monthly curie release limit used in tracking LANSCE releases, using CAP-88 modeling. This limit is imposed for any month following a month in which the running 12-month dose-to-date from LANSCE exceeds 7.0 mrem. This information is crucial, in that LANL needs to know the total of the previous 12 months' dose is as it approaches the 10 mrem/yr limit. A comparison of the total discharged Curies vs. the allowed total is graphed, such as this actual example from the month of November 1995 (Fig. 4).

Another change in the Air Emissions Management Plan was that the original agreement had called for automatic cessation of radionuclide-emitting Lab operations at 8.5 mrem; now the LANL/DOE management team meet when the accumulated total is 8.0 mrem, and agree on action to be taken when the total reach 8.5 mrem, dependent on current and pending conditions of activity. One other important change was the use of meteorological data during actual operating period, rather than using annual average meteorology data, as allowed by NESHAP regulations (by implication). Monthly projections will be modeled, using realistic but conservative meteorological data for the site. The intent is for projections to be conservative.

Fig. 4

Table I illustrates the new process for managing and reporting radioactive air emissions at LANSCE, from the inception of operations resulting in radioactive air emissions through actions to be taken should levels be projected to reach 8.5 mrem within the next month. Table II enumerates the steps taken to estimate emissions from LANSCE.

Table I

Table II

The appropriateness and accuracy of these procedures has been confirmed through off-site environmental monitoring results and doses calculated from measured stack emissions, as off-site doses have now been determined to be less than 10 mrem/yr, the Federal limit.

Additional Air Program Considerations. The above summarizes how LANL successfully implemented a management strategy for demonstrating compliance with radioactive air emissions requirements. It indicates not only how radionuclide emissions can be estimated and monitoring procedures implemented, but also how various operational, compliance, and management personnel can work together to demonstrate compliance at a complex facility. By nature, however, the steps that were summarized are retrospective. It is also useful to explore several program

considerations that have been put in place following issuance of the NON. These are summarized in the following with regard to the DOE/EPA MOU concerning NESHAPs for radionuclide emissions at DOE sites across the country, and the final USEPA policy on self-policing. In suggesting these approaches, the intent is to assist future compliance efforts by offering for consideration additional elements for meeting regulatory requirements. Not considered in the following are such approaches as the Risk Management Plan under 112(r) of the CAA, and the Environmental Management System under the voluntary ISO 14000 standard; while useful, they may be somewhat outside the scope of this paper.

Subsequent to the development of an air emissions management plan for LANSCE, the USEPA and DOE finalized a Radionuclide NESHAP MOU. This MOU serves to clarify provisions of 40 CFR Part 61, Subpart H, I, Q and T as they apply to DOE facilities. The MOU notes that in some instances where DOE facilities cannot demonstrate compliance that CEM requirements may present technical and procedural difficulties which necessitate significant effort and resources to resolve. Faced with these circumstances, the DOE facility and appropriate USEPA Regional Office can determine the most efficient compliance measures including consideration of alternative monitoring methods under 40 CFR 61.93(b)(3).

In addition to alternative monitoring methods, the MOU also indicates that engineering calculations and/or representative measurements may be used to comply with periodic confirmatory measurement requirements. This option may be applicable to those DOE facilities which have large numbers of minor release points and that have similar emissions and controls. DOE facility managers can use their best professional judgment, knowledge of the radionuclides and quantities being used in plant operations, and their potential for release to determine when representative measurements should be made and/or engineering calculations utilized.Under this MOU, DOE facilities (with prior USEPA approval) can also implement continuous monitoring procedures that differ from 40 CFR 61.93(b) reference methods; reasons for utilizing different reference methods include site-specific conditions as well as engineering, economic, health and safety considerations. In addition, the MOU allows for the use of environmental monitoring as an alternative to air dispersion calculations at critical receptor locations in order to demonstrate compliance with the radionuclide NESHAP; this assumes that the criteria contained in 6193(b)(5) are met and that the USEPA grants prior approval.

In addition to the USEPA/DOE Radionuclide NESHAP MOU, the USEPA issued on 22 December 1995 (60 FR 66706) a final self-policing policy, "Incentives for Self-Policing: Discovery, Disclosure, Correction and Prevention of Violations", aimed at encouraging the adoption of a voluntary approach to protecting the environment and public health. The policy which became effective on 22 January 1996 arose for several reasons including the realization that government enforcement efforts alone cannot achieve environmental compliance.

In implementing this self-policing policy, the USEPA has identified nine conditions that must be met in order to avoid gravity-based penalties (viz., the punitive portion of the penalty and not that portion which represents the economic gain from non-compliance). If all nine conditions are met, gravity-based penalties will not be sought. If all conditions are met except for the voluntary discovery of the violation through a formal environmental audit or due diligence, then the USEPA will reduce the gravity penalty by 75%.

The other conditions are:

Voluntary discovery other than through a legally mandated monitoring and sampling requirement that is required by law, regulation, permit, judicial or administrative order, or consent agreement. For example, the policy does not apply to:

- Violations discovered through CEM where such monitoring is required;
- Violations of NPDES discharge limits detected through sampling and monitoring; and
- Violations discovered through a compliance audit required by a consent order or settlement agreement.

Prompt disclosure in writing to the USEPA within 10 days of the specific violation.

Discovery and disclosure independent of government or third party plaintiff.

Correction of the violation with 60 days of disclosure.

Written agreement to prevent the future recurrence of the violation.

The USEPA self-policing policy indicates that criteria for determining due diligence can be met by a variety of compliance management programs. According to the policy, due diligence can be demonstrated through the systematic application of all of the following:

Identify how compliance policies, standards and procedures can meet the requirements of laws, regulations and permits.

Assignment of overall and specific responsibility for overseeing compliance with policies, standards and procedures.

Mechanisms for systematically assuring that compliance activities are being carried out.

Communication efforts with employees and others.

Compliance performance incentives for managers and employees.

Procedures for prompt and appropriate correction of violations.

The USEPA notes that this is a policy (not a regulation) which they expect USEPA employees to follow; however, the agency reserves the option of issuing a rulemaking at some later date if greater consistency and predictability are needed. Finally, within three years the USEPA will conduct a study to determine the effectiveness of this self-policing policy.

Both of these measures - the USEPA/DOE Radionuclide NESHAP MOU and the USEPA self-policing policy - add important new considerations to the facility management of air emissions control. Through these steps as well as others the complex and difficult task of achieving and maintaining facility compliance with the requirements of the CAA will be made more flexible and cost-effective.

CONCLUSIONS

The CAA is a very complex legal and regulatory framework, with interlocking and rigorous requirements. When LANL received two NONs for radioactive air emissions in 1991 from USEPA, DOE and LANL reacted by implementing a management plan to gain control of the Lab's radioactive air emissions program through a series of steps, including increased monitoring and reporting and actions to be taken at appropriate milestones. As the LANSCE facility within TA-53 contributes over 98% of the Lab's total radioactive air emissions, the majority of these steps were aimed specifically at LANSCE's emissions. This "Success Story" can serve as a guideline for other sites with similar problems, and other solutions/considerations that were not available at the time of TA-53's solution were covered in brief.

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THE ROLE OF RADIOLOGICAL PERFORMANCE ASSESSMENTS IN LLW DISPOSAL FACILITY DEVELOPMENT

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ABSTRACT
This paper presents a generalized approach to conducting performance assessments for low-level radioactive waste disposal facilities. It defines the components of the disposal system (the natural site, the engineered facility, and the waste disposed) and identifies characteristics of the components that are important to performance assessments. It presents and explains the sequence of performance assessment activities and also discusses the application of the performance assessment process in various stages of disposal facility development.

OVERVIEW

The extent of radiological performance assessment undertaken in the development of a low-level radioactive waste (LLW) disposal facility depends heavily upon the regulatory framework within which the facility must be developed. Radiological performance assessment is prescribed by regulations that control LLW disposal facility licensing, namely Title 10

of the Code of Federal Regulations, Part 61, (1,2) and compatible state regulations.

Regulations for LLW disposal facilities state general performance objectives, list technical requirements, and require that members of the general public, potential inadvertent intruders, and workers at the disposal facility be protected from radiation. Design features and operating procedures must be prepared that support these general performance objectives and the more detailed technical requirements. However, advance assurance that potential radiation exposures to members of the general public, inadvertent intruders, and facility workers will meet the performance objectives can only be provided through assessment of the facility's radiological performance.

Performance assessment involves systematically evaluating a disposal system to determine the likely extent of compliance with specific performance objectives.(3,4) The process consists of a sequence of technical activities, shown in Fig. 1, that are coordinated to generate estimates of the potential impacts of the disposed waste on human health and safety. The following are the major steps involved in assessing the performance of an LLW disposal facility:

Characterize the disposal system (i.e., waste, natural site, and disposal facility).

Identify potential radiation exposures (i.e., potential receptors and exposure scenarios).

Develop conceptual exposure models (i.e., contaminant release mechanisms, transport pathways, and uptake modes).

Develop calculational tools (i.e., mathematical models and computer codes).

Estimate and evaluate potential radiation exposures (i.e., baseline results, sensitivity analyses, uncertainty analyses, and review of results).

Revise inputs, models, and assumptions as appropriate.

Fig. 1

Radiological performance assessment should be an ongoing and iterative process, as depicted in Fig. 1. Early in the process, assumptions must be made about some characteristics that may not be known with confidence. However, as the development process continues, reliance on assumptions should decline while confidence in actual characteristics should strengthen. Furthermore, as the process continues, the results of previous assessments should be reviewed to ascertain whether key components of the assessment should be revised to reflect the improved knowledge of the disposal system characteristics. The six major steps in the performance assessment process are described below.

CHARACTERIZE THE DISPOSAL SYSTEM

The principal elements of the disposal system (the waste to be disposed, natural site, and disposal facility) should be as fully characterized as possible each time performance is assessed. Characterizing the waste element of the disposal system involves describing the types and quantities of waste to be disposed of in the LLW disposal facility.(2,4,5) Important components of this element include the annual disposal volume, radionuclide concentrations, physical and chemical characteristics, container characteristics, leaching characteristics, gas generation characteristics, and external gamma radiation levels.

Actual waste stream characteristics will be available only after disposal facility operations begin. Prior to that time, information must be

collected from the expected waste generators to develop the best possible estimates of the types and quantities of waste that will require disposal. These estimates should be updated as appropriate to maintain an accurate understanding of the waste. Figure 2 shows a sample format for presenting these data, although other formats may also be suitable.
Fig. 2
To characterize the natural site element, the abiotic and biotic conditions in the region selected for development of the LLW disposal facility must be described.(2,4,5) Key components of the natural site include the following:

- Prominent manmade and natural features in the area.
- Population distribution around the site.
- Meteorological and climatological characteristics.
- Geologic characteristics and resources.
- Seismic characteristics.
- Surface water and groundwater hydrologic characteristics.
- Geotechnical characteristics.
- Geochemical characteristics.
- Water resource utilization patterns.
- Biotic features.

Finally, characterizing the disposal facility element involves describing the disposal units, temporary holding facilities, buildings, equipment, and other engineered features (2,4,5) within the disposal (natural) site which are used for disposing of LLW. The descriptions of principal design features should include textual descriptions, engineering drawings, design calculations, construction procedures, and operating procedures.

IDENTIFY POTENTIAL RADIATION EXPOSURES

The next major step is the identification of potential receptors and definition of applicable exposure scenarios, as shown in Fig. 1. Potential receptors include

- Persons residing or who may reside at or near the boundary of the disposal facility during and following disposal operations.
- Persons who may inadvertently intrude into the stabilized disposal facility (whether or not into the waste itself) following the assumed loss of institutional control, nominally 100 years after facility closure.

Because of human persistence and creativity, protecting against intentional intrusion through passive means is extremely difficult if not impossible. Therefore, assessment of intentional intrusion should be considered optional.
Once the potential receptors have been identified, situations in which persons may be exposed to the waste or to releases from the disposal facility should be characterized. These situations, or exposure scenarios, should be broad enough in scope to provide reasonable assurance that no individual will receive radiation exposures greater than those received by the individuals posited in the assessment. They should also address the potential for both acute and chronic exposures. Exposure scenarios considered in a performance assessment should include offsite resident, intruder-drilling, intruder-construction, intruder-discovery, and intruder-agriculture scenarios.
The exposure scenarios developed should be internally consistent and reasonable, that is, their assumed events and activities should not involve conditions that could not logically occur.

DEVELOP CONCEPTUAL EXPOSURE MODELS

A conceptual model should be developed for each exposure scenario. The conceptual models define the release mechanisms, transport pathways, and modes of uptake that result in the postulated exposures.

All phenomena that control or influence the release of radionuclides from the disposal unit should be identified, described, and characterized. These may include water infiltration, container degradation, leaching, gas generation, and intrusion into the waste.

The pathways by which contaminants released from the disposal unit may be transported away from the disposal unit should be identified, described, and characterized.(2) These include groundwater, surface water, atmospheric diffusion and dispersion, gaseous releases through cover system, and food chain transport.

All phenomena that control or influence the exposure of humans to external radiation or the uptake of contaminants by humans should be identified, described, and characterized. These include water ingestion, food ingestion, inhalation, external radiation, and dermal absorption.

DEVELOP CALCULATIONAL TOOLS

Once a conceptual exposure model is developed, the Applicant should determine appropriate mathematical representations of each model component, as shown in Fig. 1. This process involves identifying models available for each phenomenon, evaluating model characteristics and abilities, comparing model characteristics and abilities to represent the most important aspects of the phenomena, and selecting the preferred model.

In general, several mathematical representations or models exist for each release mechanism, transport pathway, and uptake mode. Some models of a given phenomenon are simpler than others, and each has its own data requirements. The ability of each model to represent the actual phenomenon should be evaluated considering the limitations and strengths of each.

Most models of release mechanisms, transport pathways, and uptake modes have been implemented as computer codes and are widely available. Existing codes may require modification to adequately represent the phenomena at the disposal facility. When such modifications are necessary or preferable, care should be taken to ensure that the resulting code is properly implemented and documented. The code should be benchmarked and validated to the extent practical.

As a general rule, the model(s) used should not be more sophisticated than is necessary to adequately represent the actual phenomenon. Furthermore, the sophistication of the model(s) used in the performance assessment should be consistent with the level of detail of data available for input.

ESTIMATE AND EVALUATE POTENTIAL RADIATION EXPOSURES

As depicted in Fig. 1, after the first four major steps of the performance assessment have been completed, the performance of the disposal facility can be assessed. The assessment will demonstrate the extent to which the disposal facility satisfies radiological performance objectives, identify parameters significant to the analysis, and examine the effects of uncertainty on the projected results. Following the assessment, the results should be reviewed for reasonability and consistency with data collected during characterization efforts.

The base-case performance assessment indicates the extent of compliance with the performance objectives and technical requirements using the actual or expected conditions at the facility. Under many circumstances,

the actual conditions will be represented by making simplifying assumptions. When simplifying assumptions are necessary, they should be conservative (i.e., that the exposure results will not be understated), but also realistic. Extreme levels of conservatism may not be useful.

A significant element of the performance assessment is the analysis of sensitivities and uncertainties in the assessment.(6,7) Sensitivity analyses are conducted to identify important elements and parameters of the system, and involve estimates of changes in performance measures produced by changes in the variables of the system. Uncertainty analyses are conducted to identify uncertainties associated with the important elements and parameters of the system, and to understand the impacts these uncertainties have on the conclusions drawn from the calculated performance measures.

Sensitivity analyses typically involve perturbing each model parameter while keeping all other parameters at their nominal values. The relative effect of the perturbation on the model prediction is quantified; the parameters having the greatest influence on the model projections are designated as the most sensitive parameters in the model. Sensitivity analyses, then, focus on identifying the model parameters of greatest consequence to the projected results.

The need for uncertainty analyses arises from the fact that all environmental and dose assessment models are inherently uncertain. Uncertainties have been categorized as a)uncertainty in conceptual and mathematical models (model uncertainty); b)uncertainty about the future state of the site (scenario uncertainty); and c)uncertainty in the input data used in the models (parameter uncertainty).(7)

The results of all performance assessment calculations should be reviewed to ensure that they are reasonable and internally consistent. The results should be compared with assumed conditions to ensure that no logical contradiction exists. If calculated results infer a condition that is different from an assumed condition, the contradiction should be pursued to determine whether the assumed condition should be revised, or whether more fundamental aspects of the modeling process should be questioned. The process of judging the reasonability of calculated results requires knowledge of the sensitivities and uncertainties, as well as the perspective they provide.

REVISE INPUTS, MODELS, ASSUMPTIONS

Upon reviewing the results of the performance assessment, the need for revisions to the modeling process should be determined, as shown in Fig. 1. The extent of revisions may be as superficial as revising selected input values. However, the review of the results may also reveal that the entire performance assessment process should be revised. This may be the case, for example, if additional site-specific data indicate that the site conceptual model used in the performance assessment was inappropriate.

The potential effects that any data revisions may have on performance assessment results should be considered. If the revisions are expected to have important effects, the assessment should be revised to make it consistent with the data revisions. The revised performance assessment results should receive scrutiny similar to that received by the original results to ensure it is reasonable and internally consistent.

Revisions in the performance assessment process may be made within a particular stage of the facility life or as the facility evolves from one stage to the next. If the revisions are made within a single facility

stage, they may be made to virtually any step in the performance assessment process, as shown in Fig. 1. If the facility is progressing from one stage to the next, consideration should be given to reviewing all steps in the process and evaluating previous decisions according to the most recent and complete information available.

CUSTOMIZING PERFORMANCE ASSESSMENT TO EVOLVING NEEDS

The important stages of the disposal facility development process during which the radiological performance should be assessed are the following:

- Site screening.
- Site characterization and selection.
- Facility design and licensing.
- Facility operations.
- Facility closure and decommissioning.
- Facility postclosure and monitoring.
- Long-term care and license termination.

The objectives of assessing radiological performance and the nature of such assessments change significantly throughout the facility life. Early on, the extent of knowledge about the waste, site, and facility is relatively limited and superficial compared to that existing following facility closure. Differences in the extent of knowledge that exists over time should influence the methodology used to assess radiological performance and the emphasis placed on the results of such assessments. In early stages of the facility development process, the results of performance assessments should be used to influence (but not dictate) decision making. Later, performance assessments should be used to confirm earlier projections of facility performance and to identify appropriate monitoring and management approaches.

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WORKSHOP -- HOW CLEAN IS CLEAN

Co-Chairs: J. S. Devgun, Consultant; D. LeMone, UTEP

WORKSHOP SUMMARY
"HOW CLEAN IS CLEAN?"
D&D AND RELEASE CRITERIA
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This workshop was the fourth in the "How Clean is Clean?" series. The workshop focused on the important topic of release criteria for decontamination and decommissioning. The number of facilities and sites requiring D&D and cleanup runs into thousands, and the cleanup costs are projected to be in hundreds of billions of dollars. In the DOE complex alone about 6,000 facilities have been declared surplus and targeted for D&D. A number of nuclear power reactors have also been decommissioned worldwide and many other projects are currently underway. With D&D gaining momentum both in the federal and the private sector, the objective this year was to focus discussion specifically on the release criteria relevant to D&D projects.

A panel of experts was convened to discuss and debate the issues related to the theme of the workshop. The panel included representatives from the U.S. Department of Energy (DOE), U.S. Nuclear Regulatory Commission (NRC), Atomic Energy of Canada Limited (AECL), Los Alamos National Laboratory (LANL), Scientific Ecology Group (SEG), and the two workshop Co-Chairs.

Workshop format included presentations by the panelists followed by breaking out into two focus groups and then reconvening for a general discussion. Each focus group was led by two discussion leaders. The topics for group discussion were:

1) Risk-based criteria vs.generic concentration limits (site-specific criteria, pathways modeling, ALARA)
2) Criteria, cost and the need for BRC or de minimus.

Jas Devgun provided an overview of the D&D in the nuclear power industry and in the federal sector along with a very brief summary of the international projects. He also summarized the relevant national regulations and the international guides. NRC's proposed D&D criteria of 15 mrem/yr TEDE was also discussed.

Anthony Kluk of DOE discussed the DOE Order 5400.5 which lists the release guidelines used by DOE for decontamination of equipment, structures, and sites, implementation of these guidelines presented in DOE/CH 8901 report, application of RESRAD, and field examples from the Santa Susana Field Laboratory D&D project. Recent guidance from DOE/EH is that limit for soil should be selected such that ALARAis used and documented and that individual dose to a member of the public is less than 25 mrem/yr. Margaret Federline represented the NRC and provided an overview of the existing NRC criteria and guidance. Summary of the proposed rule was presented along with a detailed discussion of the public comments. The proposed rule provides a limit of 15 mrem/yr plus ALARA for unrestricted release.

The EPA representative or alternate were not able to attend. Notes presented by the EPA representative at the second workshop (Anthony Wolbarst) were considered relevant to D&D and were made available to the participants.

Robert Pollock of AECL presented the Canadian perspective on D&D and the release criteria. Regulatory Guide R-85 provides the policy statement.

The document provides radiation protection prerequisites for the exemption of certain radioactive materials from further licensing upon transferral for disposal.

Patrick LaFrate discussed the decontamination and decommissioning of surplus facilities at LANL. Lessons learned from these projects were discussed. The metal from D&D operations was recycled at SEG's facilities in Oak Ridge. James Nicolosi of SEG describes a processor's perspective and discussed the metal recycle facilities now available in Oak Ridge. David LeMone (UTEP) summarized the international developments in the D&D and the BRC area. Decommissioning plans recommended by the International Atomic Energy Agency were discussed as were the activities categories for radioactive scrap metal.

The discussion groups were led by Matthew Kozak, Mahmoud Haghighi, and the two Co- Chairs. The panelists also participated in the discussion groups. Each group presented their key points when all participants reconvened for a general discussion. Risk-based cleanup criteria were generally favored by the participants rather than the generic limits. Release criteria remain in flux. There is a need on the part of the federal agencies to come to a consensus on the acceptable dose criteria. Two recent events in this area are significant. One is the formation of Interagency Steering Committee on Radiation Standards (ISCORS) in April 1995. The other is the Multi-Agency Radiological Site Survey and Investigation Manual (MARSSIM) that is expected to be issued in mid-1996 in the draft form.

Approximately 70 people participated in the workshop. A package of papers, notes, and view graphs was provided to each participant. In the evaluations, a majority of the participants rated the workshop in the "good" to "very good" category.

EXHIBITORS

3M
ACZ Laboratories, Inc.
AECL Technologies, Inc.
AFFTREX, Ltd.
ALARON Corporation
ASTM
ATG, Inc. Allied Technology Group
Adtechs Corporation/JGC Corporation
Advanced Sciences, Inc.
Advanced Systems Technology, Inc.
Alternative Remedial Technologies, Inc.
American Operations Corporation
American Technologies/Diversified Envir. Services
Analytical Technologies, Inc.
Applied Radiological Control, Inc. (ARC)
Aptec Nuclear Inc.
Aqua Tech Environmental Laboratories,Inc.
B&W Nuclear Environmental Services , Inc.
BNFL plc
BNFL Inc.
BNFL Instruments/Pajarito Scientific Corporation
BROOKS
Bartlett Nuclear, Inc.
Beckman Instruments, Inc.

Benchmark Environmental Corporation
Bio-Imaging Research, Inc.
Brown & Caldwell
Brown & Root Environmental
CH2M Hill
CORE Laboratories, Inc.
CORPEX Technologies, Inc.
Canberra Industries
Chem-Nuclear Systems, Inc.
Chemrad Tennessee Corporation
Clemson Univerisity Research Laboratory
Comtronics, Inc.
Container Products Corporation
DIAL (Diagnostic Instrumentation & Analysis Lab.)
DataChem Laboratories
Desco Mfg. Co. Inc.
DynCorp
EET Corporation
EG&G Nuclear Instruments
Eberline Instruments
Energy Laboratories, Inc.
Envirocare of Utah, Inc.
Environmental Science &Engineering, Inc.
Environmental Systems Engineering Dept.
Envitco, Inc.
Euclid Garment Mfg Co
Exchange Monitor Publications
Fluid Tech, Inc.
Fluor Daniel, Inc.
Fontijne Holland BV
Foster Wheeler Environmental
Framatome Technologies
Frham Safety Products, Inc.
GPU Nuclear
GTS Duratek
Gamma Products
General Engineering Laboratories, Inc.
Geosafe Corporation
Genden Engineering Services and Construction Company
Geraghty & Miller Inc.
GPI
Gutierrez-Palmenberg, Inc.
HI-Q Environmental Products Company
Helgeson Scientific Services
Hot Cell Services Corporation
IDM Environmental Corp.
IceSolv, Inc.
Intergraph Corporation
International Atomic Energy Agency
International Technology Corporation (IT)
Interstate Nuclear Services Corp.
Isotope products Laboratories
J & R Engineering Company, Inc.
Jacobs Engineering Group Inc.

LND Inc.
LTC Americas Inc.
Lockheed Martin Environmental Systems & Tech.
Los Alamos National Laboratory
Ludlum Measurements, Inc.
M4 Environmental L.P.
MJW Corporation Inc.
MGP Instruments
MSE, Inc.
Manufacturing Sciences Corporation
Maxim Technologies Inc.
Modec/HAZDECON, Inc.
Molten Metal Technology
Morikawa Industries Corp.
NES, Inc.
NFT Incorporated
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Call For Papers
March 2-7, 1997
"HLW, LLW, Mixed Wastes and Environmental Restoration-Working Towards a Cleaner Environment"

WM'97, will be held March 2-7, 1997 at the Tucson Convention Center, Tucson, Arizona. Organized by WM Symposia, Inc., an Arizona non-profit corporation, the conference is hosted by the University of Arizona. Sponsoring organizations include the University of Arizona, the American Nuclear Society, the U.S. Department of Energy, New Mexico State University with the Waste-management Education and Research Consortium (WERC), and the American Society of Mechanical Engineers. The conference is organized in cooperation with the International Atomic Energy Agency. To assist international participation, we have added the names of contact people outside the US for as many topics as possible. The topics selected for WM'97 will have invited and contributed papers involving research, development and operational experience in nuclear waste, mixed waste, mill tailings, environmental restoration, waste management, and decommissioning. Papers concerning national and international agreements and regulations governing these topics as well as the impact of these activities on the environment are also solicited. Interested contributors to the meeting are invited to submit extended summaries on a 3 1/2 inch diskette and three hard copies. A single copy by fax will be accepted but the 3 1/2 diskette and three hard copies must follow by overnight mail. The disk is necessary for publishing to the World Wide Web (WWW), which we are considering in order to make the abstracts available to the Program Advisory Committee before the Paper Review. The Program Advisory Committee will be provided with an access code so that only they will be able to access this information two weeks prior to the Paper Review. The authors will be required to approve placing their summary on the WWW. Such approval has no influence on acceptance of the summary.

The summaries must show clearly 1) Topic (what paper is about), 2) Basis for originality, 3) or audience interest, 4) Conclusions. Summaries not meeting the above criteria will not be considered by the Program Advisory Committee (PAC). The fully completed attached form must accompany the extended summary for it to be considered. If you do not receive an acknowledgment within 10 days of your mailing, call (520) 624-8573. The approved papers will be assigned to either oral, poster or workshop sessions by the Technical Program Chairman with the advice of the PAC. The criteria for assignment is to assure that each paper is presented in the method of presentation best suited for its content, format and anticipated interaction with its audience. The publication of both the poster and oral papers will be identical.

The summaries will undergo critical technical review by the PAC to determine if they meet the criteria of technical content, significance and subject. The summary should be long enough to convey to the committee the substance of your proposed paper and its meeting of the stated criteria. Summaries submitted after due date may not be considered. Concerns about the quality of some papers which could be attributed to inadequate preparation led to our new policy of requiring that full papers be written and reviewed before the presentation. This process will also permit much earlier publication.

The full papers will be published on CD's and on the WWW. We also plan to put each full paper on the WWW as it is completed. This will provide much earlier distribution of your work, limited only by your submission schedule. WM'95 was our first use of CD's and some special considerations were granted which will not be possible for WM'97. We expect that many papers will be on the net before the conference but they will be

available only to the early registrants who have paid. Wouldn't you like for your paper to be one of the first available?
Please follow the schedule listed below:

Summary due by	August 16, 1996
Paper Review	September 16, 1996
Draft Full Papers	November 15, 1996
Reviewer Comments to Author	December 15, 1996
Final Approved Papers Due	February 3, 1997

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WM'97 Topics
Topics are listed under "Tracks" designed to highlight specific interests. There is no longer a need for the words US or International since all sessions are open to papers from all countries and are limited only by technical content and relevance. In the event that an entire session deals with a given International group then we will identify the group doing the organization of the session. All topics include Regulatory, Standards, Technology, QA, Risk Assessment, Fate and Transport Modeling and Monitoring where appropriate.

GENERAL SESSIONS

1.0 Nuclear Waste Policies and Programs-Lawrence Harmon, MACTEC, V-(301)353-9444, F-(301)353-9447; Edward J. Bentz, Jr., E.J. Bentz & Assoc., Inc., V-(703)455-7469, F-(703)-912-6578

1.1 Plenary-Roy G. Post, WM Symposia, Inc., V-(520)624-8573, F-(520)792-3993, E-mail-wmsym@basix.com (*rgp@wmsym.org; Morton E. Wacks, U of AZ, V-(520)624-8573, F-(520)792-3993, E-mail-wmsym@basix.com (*mew@wmsym.org)
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1.2 Programs and Progress-Lawrence Harmon, MACTEC, V-(301)353-9444, F-(301)353-9447; Pierre Barber, ANDRA, V-33-1-46-11-8068, F-33-1-46-11-8268

1.3 Regulatory Compliance-Edward J. Bentz, Jr., E.J. Bentz & Assoc., Inc., V-(703)455-7469, F-(703)-912-6578; Donald E. Wood, WHC, V-(509)376-7832, F-(509)372-1033, E-mail-Donald_E_Wood@rl.gov

1.4 Health and Safety Issues-Carol A. Peabody, USDOE, (202)586-0201, F-(202)586-0916

2.0 High-Level-(Including Spent Fuel), Transuranic- and Long-Lived-Waste-Marshall J. Anderson, BDM Federal, Inc., V-(301)601-5382, F-(301)601-5426, E-mail- manderso@bdm.com; Jim W. Voss, Golder Associates International, V-(206)556-5590, F-(206)556-5595; Leif G. Erikkson, Advanced Sciences, Inc.,V-(505)887-1079, F-(505)887-5494; Joachim Fleisch, WAK, V-49-7247-88-2230, F-49-7247-88-2144

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2.2 Concepts under Investigation (Including Environmental Studies)-Harry Babad, Westinghouse Hanford Co., V-(509)373-2897, F-(509)373-3198, E-mail-harry_babad@rl.gov; John Mathieson, UK Nirex, V-44-1235-825606, F-44-1235-825459, E-mail-100661.3164@compuserve.com

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3.0 Low/Intermediate Level (Including Very Low Level Waste) and Mixed Waste Treatment Stabilization & Disposal-Lloyd McClure, Lockheed-Martin IdahoTechnologies Co., V-(208)526-1170, F-(208)526-5142; M. Jorda, ANDRA, V-33-1-46-118367, F-33-1-46-11-83-48; Leon C. Borduin, LANL, V-(505)667-3150, F-(505)665-2897; Radovan Kohout, Kohout & Associates, V-(416)488-9466, F-(416)488-2007

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3.2 New Stabilization and Disposal Concepts under Investigation (Including Vitrification)-Paul Kalb, Brookhaven National Lab., V-(516)344-7644, F-(516)344-4486, E-mail-kalb@bnl.gov; A. Jouan, CEA, Marcoule, V-33-66-79-63-76, F-33-66-79-60-30, E-mail-milhes@cea.fr

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4.0 Transportation and Packaging (Including Control)-H. Richard Yoshimura, Sandia National Labs., V-(505)845-8181, F-(505)844-0244, E-mail-hryoshi@ttd.sandia.gov; Carole B. Bentz,
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4.3 New Concepts under Investigation-Richard Rawl, IAEA, V-43-1-2060-21260, F-43-1-20607; Lawrence Harmon, MACTEC, V-(301)353-9444, F-(301)353-9447

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6.0 Utility Waste Management-C. Clint Miller, Pacific Gas & Electric Co., V-(805)545-4582, F-(805)545-3459

6.1 At Reactor and Away From Reactor Fuel Storage-C. Bonnet, SGN, V-33-1-30-587498, F-33-1-30-587628; Ramesh Dayal, Ontario Hydro, V-(416)207-6049, F-(416)207-6094, E-mail-dayalr@rd.hydro.on.ca

6.2 Low/Intermediate Level Waste Processing-C. Clint Miller, Pacifc Gas & Electric Co., V-(805)545-4582, F-(805)545-3459; A. Scott Dam, BNFL Inc., V-(703)385-7100, F-(703)385-7128

6.3 Low/Intermediate Level Waste Storage-Larry C. Oyen, Sargent & Lundy, V-(312)269-6750, E-mail-Larry.c.oyen@slchicago.infonet.com A. Rex Meeden, APS, V-(602)393-6582, F-(602)393-5285

6.4 New Concepts in Utility Waste Management under Investigation-A. Rex Meeden, AZ Public Service, V-(602)393-6582, F-(602)393-5285; Thomas L. Nauman, ComEd, V-(815)942-2920 ext. 2841, F-(815)942-0579

6.5 Reactor Decontamination and Decommissioning-A. Freitag, Bechtel Hanford, Inc., V-(509)375-9687, F-(509)372-9447; Jas S. Devgun, Consultant, V-(708)985-9386, F-Same; Michele Laraia, IAEA, V-43-1-2060-26105, F-43-1-20607, E-mail-Laraia@NEP01.IAEA.OR.AT

6.6 NPP Waste Minimization Techniques-C. Clint Miller, Pacific Gas & Electric Co., V-(805)545-4582, F-(805)545-3459; Jas S. Devgun, Consultant, V-(708)985-9386, F-Same; E. Tarnuzzer, Consultant, V-(508)897-8512

6.7 Toxic & Hazardous Waste Management AT NPP-Jas S. Devgun, Consultant, V-(708)985-9386, F-Same; M. Mauzy, Roy F. Weston, Inc., V-(505)884-5050, F-(505)837-6870

7.0 Public Communication, Participation, Education and Training-Linda Ulland, Minnesota Pollution Control Agency, V-(218)828-6115, F-(218)828-2594; Ginger King, OCI Inc., (703)616-5307, F-(703)416-0007; Carol Worth, Words Worth Publishing & Communications, V-(703)742-0017, F-(703)742-0059, E-mail-Carolworth@aol.com

7.1 Impact of Information Technology on Community Outreach, Environmental Sciences and Public Acceptability-Ginger King, OCI Inc., V-(703)616-5307, F-(703)416-0007

7.2 Indigenous People-Merv L. Tano, Council of Energy Resource Tribes, V-(303)297-2378, F-(303)296-5690, E-mail-mervtano@ix.netcom.com;Michelle Rehmann, Energy Fuels Nuclear Inc., V-(303)899-5647, F-(303)595-0930

7.3 New Concepts under Investigation-Connie Callan, The University of New Mexico, V-(505)277-7750, F-(505)277-7833; Ron Bhada, WERC, V-(505)646-1510, F-(505)646-4149

7.4 Spent Fuel Storage and MRS Concepts*- Merv. L. Tano, Council of Energy Resource Tribes, V-(303)297-2378, F-(303)296-5690, E-mail-mervtano@ix.netcom.com

8.0 Infrastructure in Waste Management-Lowell Snow, Consultant, V-(703)690-3545, F-Same

8.1 Cost Effectiveness (Cost/Benefit) Analysis-Lloyd McClure, LIMTCO, V-(208)526-1170, F-(208)526-5142; John Mathieson, UK Nirex, V-44-1235-825606, F-44-1235-825459, E-mail-100661.3164@compuserve.com

8.2 Data Management-Jorma Heinonen, Lappeenranta Univ. Of Technology, V-358-53-621-2602, F-358-53-621-2699; Marshall J. Anderson, BDM Federal, V-(301)601-5382, F-(301)601-5426; H-J. Engelmann, DBE, V-49-5171-917710, F-49-5171-917740; Carlos Torres, IAEA, V-431-2060-21428, F-431-20607

8.3 Project Privatization and Outsourcing of Waste Services-J. Ed Day, ADTECHS, V-(703)713-9000, F-(703)713-9101; William Greenman, GTS-Duratek, V-(410)290-7078, F-(301)621-8211; Edward L. Helminski, Exchange Monitor Publications, Inc., V-(202)296-2814, F-(202)296-2805

9.0 Other Waste Problems, e.g. Mill Tailings-Morton E. Wacks, U of AZ, V-(520)624-8573, F-(520)792-3993, E-mail-wmsym@basix.com (*mew@wmsym.org); Werner Lutze, The University of New Mexico, V -(505)277-7964, F-(505)277-5433

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